ORNL/TM-6036

Preliminary Analysis of Alternative Fuel Cycles for Proliferation Evaluation

OAK RIDGE NATIONAL LABORATORY
OPERATED BY UNION CARBIDE CORPORATION FOR THE ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION

Printed in the United States of America. Available from
National Technical Information Service
U.S. Department of Commerce
5285 Port Royal Road, Springfield, Virginia 22161
Price: Printed Copy ~~$8.00~~ 9.50; Microfiche $3.00

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ORNL/TM-6036

Contract No. W-7405-eng-26

PRELIMINARY ANALYSIS OF ALTERNATIVE FUEL CYCLES FOR PROLIFERATION EVALUATION

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Foreword

On April 7, 1977, President Carter made the following nuclear power policy statements:

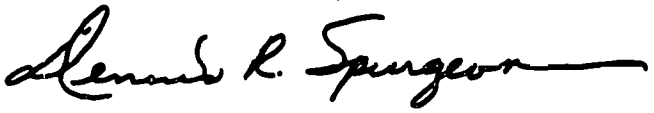
"The benefits of nuclear power are thus very real and practical. But a serious risk accompanies world-wide use of nuclear power--the risk that components of the nuclear power process will be turned to providing atomic weapons."

"The U.S. is deeply concerned about the consequences for all nations of a further spread of nuclear weapons or explosive capabilities. We believe that these risks would be vastly increased by the further spread of sensitive technologies which entail direct access to plutonium, highly enriched uranium or other weapons useable material."

"We will redirect funding of U.S. nuclear research and development programs to accelerate our research into alternative nuclear fuel cycles which do not involve direct access to materials useable in nuclear weapons."

As a result of this policy statement, ERDA embarked on a vigorous effort aimed at evaluating various alternative fuel cycle systems having possible non-proliferation advantages. Over 60 possible candidates were identified for preliminary evaluation. Contractors for the Division of Waste Management, Production and Reprocessing, ERDA, proceeded to characterize and define the candidate alternatives from a fuel cycle processes and operations standpoint. This document contains these preliminary definitions in the format of functional flow diagrams and tabulated evaluations of the operations shown in these diagrams. In addition to the definitions presented herein, process streams containing fissionable material were identified; and the state-of-the-art for each process function has been included.

WPR is continuing to assess these nuclear fuel cycle alternatives from the viewpoint of costs, materials flows and inventories, and proliferation potential. A follow-on document will be issued when the work is completed.



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Production and Reprocessing

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CONTENTS

1. INTRODUCTION

The United States Energy Research and Development Administration Division of Nuclear Research and Applications (ERDA-NRA) proposed 67 nuclear fuel cycles[1] for assessment as to their nonproliferation potential. Although the candidate alternative systems present distinct cases with respect to reactor (or combinations of reactors) operations, certain reprocessing and refabrication operations are common to several cases, making it expedient to treat the cases generically rather than individually. The object of the assessment is to determine which fuel cycles pose inherently low risk for nuclear weapon proliferation while retaining the major benefits of nuclear energy. Economics, resource use, and timeliness are also factors in the evaluation. Argonne National Laboratory (ANL), Hanford Engineering Dvelopment Laboratory (HEDL), Oak Ridge National Laboratory (ORNL), and Savannah River Laboratory (SRL) participated in studying the back cycles of the candidate systems. This is a preliminary analysis of these fuel cycles to develop the fuel recycle data that will complement reactor data, environmental data, and political considerations, which must be included in the overall evaluation. Back cycle evaluations *per se* are inadequate for eliminating certain fuel cycles from further consideration. This report presents the preliminary evaluations from the four Laboratories and is the basis for a continuing in-depth study.

The proposed fuel cycles are identified in Table 1.1 and correlated with the responsible Laboratory and the location in this report where the evaluation can be found. Case assignments among the several participants were such that presentation of results on a case-by-case basis was impractical. Hence the evaluations are presented in alphabetical order by participating institution. ORNL assumed the responsibility for all thorium cycles plus high-temperature gas-cooled reactor (HTGR) cycles; SRL for uranium-plutonium fuels in light-water reactor (LWR), heavy-water reactor (HWR), and spectral shift systems; ANL/HEDL for uranium-plutonium fuels in the liquid-metal fast breeder reactor (LMFBR) and the gas-cooled fast breeder reactor (GCFR). Advanced concepts (Group 6) and certain

Table 1.1. Index for Correlating ERDA-NRA Identification of Fuel Cycles with Participants in Study and Reprocessing and Refabrication Evaluations

ERDA-NRA Identification	Fuel Cycle Description	Responsible Laboratory	Index to Fuel Cycle Evaluation
Group 1.1	LWR (U-Pu)	SRL	Chap. 5
1.1.1	Standard recycle (reference)		
1.1.2	Spiked recycle		
1.1.3	Coprocessed recycle		
1.1.4	Uranium recycle, Pu stowaway		
1.1.5	Once-through oxide fuel		
1.1.6	Once-through metallic fuel		
Group 1.2	LWR (Th-U)	ORNL	Fig. 4.2
1.2.1	Full recycle		
1.2.2	Partial recycle, Pu stowaway		
Group 2.1	Gas-Cooled Reactors (U-Pu cycle)	ORNL/SRL	Fig. 4.3
2.1.1	Standard recycle in HTGR (reference)	ORNL	
2.1.2	Spiked recycle in HTGR		
2.1.3	Coprocessed recycle in HTGR		
2.1.4	Partial recycle in HTGR, Pu stowaway		
2.1.5	HTGR once-through oxide fuel		
2.1.6	GCR once-through metallic fuel	SRL	Chap. 5
2.1.7	Standard recycle in AGR (reference)	SRL	Chap. 5
Group 2.2	HTGR (Th-U cycle)	ORNL	Fig. 4.4
2.2.1	Full recycle		
2.2.2	Partial recycle, Pu stowaway		
2.2.3	HEU-Th (carbide) HTGR full recycle (reference)		
2.2.4	HEU-Th in pebble bed gas reactor with full recycle (reference)		
Group 3.1	HWR (U-Pu cycle)	SRL	Chap. 5
3.1.1	Standard recycle		
3.1.2	Spiked recycle		
3.1.3	Once-through oxide fuel		

Table 1.1. (Continued)

ERDA-NRA Identification	Fuel Cycle Description	Responsible Laboratory	Index to Fuel Cycle Evaluation
Group 3.2	HWR (Th-U cycle)	ORNL	Fig. 4.2
3.2.1	Full recycle		
3.2.2	Partial recycle, Pu stowaway		
Group 3.3	LWR/HWR tandem cycle	SRL	Chap. 5
3.3.1	Tandem cycle with reconstituted fuel		
3.3.2	Tandem cycle without reconstituted fuel		
Group 3.4	Spectral shift reactors	SRL/ORNL	
3.4.1	Once-through uranium cycle	SRL	Chap. 5
3.4.2	Th-U full recycle	ORNL	Fig. 4.2
Group 4.1	LMFBR (U-Pu cycle)	ANL/HEDL	Chap. 2,3
4.1.1	LMFBR/LWR standard recycle (reference)		
4.1.2	LMFBR/LWR coprocessed recycle oxide fuel		
4.1.3	LMFBR/LWR coprocessed recycle carbide fuel		
Group 4.2	LMFBR (Th-U cycle)	ORNL	Fig. 4.5
4.2.1	U-Pu-Th full recycle LMFBR/LWR		
4.2.2	$^{233}U/^{238}U$/Th LMFBR/LWR, U recycle, Pu stowaway		
4.2.3	^{233}U/Th LMFBR/LWR full recycle		
Group 5.1	GCFR (U-Pu cycle)	ANL/HEDL	Chap. 2
5.1.1	GCFR/LWR standard recycle (reference)		
5.1.2	GCFR/LWR coprocessed recycle oxide fuel		
5.1.3	GCFR/LWR coprocessed recycle carbide fuel		
Group 5.2	GCFR (Th-U cycle)	ORNL	Fig. 4.6
5.2.1	U-Pu-Th full recycle GCFR/LWR		
5.2.2	LEU-Th GCFR/LWR U recycle, Pu stowaway		
5.2.3	^{233}U/Th GCFR/LWR full recycle		

Table 1.1. (Continued)

ERDA-NRA Identification	Fuel Cycle Description	Responsible Laboratory	Index to Fuel Cycle Evaluation
Group 6	Advanced concepts		Not evaluated
6.1	MSBR (Th-U cycle)		
6.2	Direct reenrichment		
6.3	Electronuclear fuel and power production		
6.4	Gasecus core reactor		
6.5	Fusicn hybrid breeder reactor (FHBR)		
Group 7	Energy centers with ccnverters and recycle inside, converters outside	SRL/ORNL	Not evaluated
7.1	Energy center containing U-Pu converters, dispersed U-Pu converters; recycling within center, no mixed oxide fuel to dispersed reactors		
7.1.1	U-Pu LWR inside, U-Pu LWR outside	SRL	Chap. 5
7.1.2	U-Pu HTGR inside, U-Pu LWR outsice	ORNL	Not evaluated
7.2	Energy center containing Pu-Th converters, dispersed LEU-Th converters; full recycle within center	ORNL	Not evaluated
7.2.1	Pu-Th LWR inside, LEU-Th LWR outside		
7.2.2	Pu-Th HTGR inside, LEU-Th LWR outsice		
7.2.3	Pu-Th HWR inside, LEU-Th LWR outsice		
7.3	Energy center containing U-Th converters, dispersed LEU-Th converters; U recycle, Pu stowaway in center	ORNL	Not evaluated
7.3.1	LWR inside, SSCR outside		
7.3.2	HTGR inside, SSCR outside		
7.3.3	HWR inside, SSCR outside		

Table 1.1. (Continued)

ERDA-NRA Identification	Fuel Cycle Description	Responsible Laboratory	Index to Fuel Cycle Evaluation
Group 8	Energy centers containing U-Pu breeders and U-Pu-FP fuel sent to dispersed converters		
8.1	LMFBR inside	ANL	Chap. 2
8.1.1	LWR outside	SRL	Not evaluated
8.1.2	HTGR outside	ORNL	Not evaluated
8.1.3	HWR outside	SRL	Not evaluated
8.2	GCFR inside	ANL	Chap. 2
8.2.1	LWR outside	SRL	Not evaluated
8.2.2	HTGR outside	ORNL	Not evaluated
8.2.3	HWR outside	SRL	Not evaluated
Group 9	Energy center with breeders with U-Pu core, Th blanket inside; LEU-Th fuel sent to dispersed converters	ORNL	
9.1	LMFBR inside		Not evaluated
	LWR outside		
9.1.2	HTGR outside		
9.1.3	HWR outside		
9.2	GCFR inside	ORNL	Not evaluated
9.2.1	LWR outside		
9.2.2	HTGR outside		
9.2.3	HWR outside		
Group 10	Energy center containing breeders with U-Pu core, Th blanket inside; LEU-Th breeder (LMFBR) outside	ORNL/HEDL	
10.1	LMFBR inside	HEDL	
10.2	GCFR inside		Not evaluated

combinations of cycles involving energy centers were not evaluated in this preliminary study. This study might not include all interesting cases, and other fuel cycles may be added for future study.

The relationship of the back cycle to the overall fuel cycle is shown in Fig. 1.1.

1.1 ELEMENTS OF THE EVALUATION

Level 0 and Level 1 functional flow diagrams were prepared to define principal operations of reprocessing and refabrication. Each operation was characterized by five elements, which were chosen to provide a cursory technical assessment of the required fuel cycle and relate the nuclear material in process to its attractiveness for diversion. These elements are needed development, material location, material description, convertibility, and radiation hazard. The ratings assigned to these elements allow comparison among fuel cycles and broadly characterize process streams containing fissionable material but are insufficiently detailed to permit exclusion of certain fuel cycles; however, this method of rating does bring out advantages of some cycles over others.

Needed development relates to the state of the art of the several operations on the functional flow diagram to provide an appraisal of these operations with respect to the particular fuel cycle. The rating defines the research phase to which development of the process step has progressed in the sequence: cold laboratory, hot laboratory, cold engineering, hot engineering, prototype, demonstration facility (hot pilot plant or small commercial facility), and developed. For example, hot engineering would indicate that development has progressed through cold laboratory, hot laboratory, and cold engineering stages but needs hot engineering and all stages beyond.

Material location identifies the facility that is needed to house the process step, such as hot cell, glove box, shielded alpha facility, or hands-on facility. Material location is probably not a serious deterrent to proliferation but could impede diversion from safeguards considerations. The importance of material location to proliferation can be significantly influenced by equipment design, operating procedures, and administrative controls.

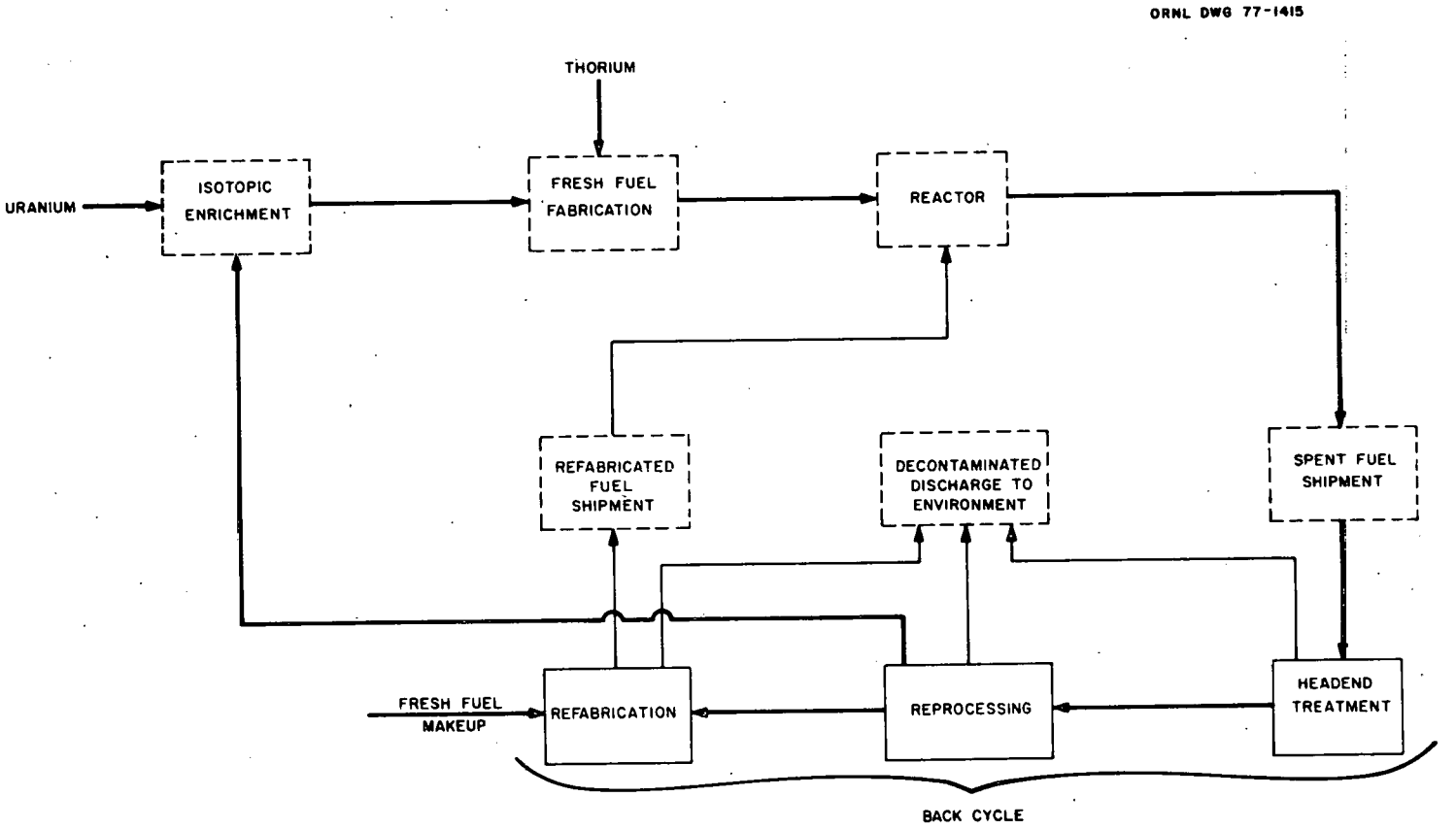


Fig. 1.1. Level 0 Functional Flow Diagram for Recycle of Fission Reactor Fuel.

Material description identifies the material and/or stream in the process operation. Chemical and isotopic concentrations were not considered in this preliminary evaluation. However, these quantities need to be assessed in an in-depth study.

Convertibility refers to the usefulness of diverted material for making a weapon. An arbitrary scale, using *nonfissionable*, *a*, *b*, *c*, and *d*, is used to give a qualitative rating to diverted material where

- *nonfissionable* identifies material that cannot be used for a weapon;
- *a* identifies radioactive fissionable material that requires a shielded isotope separation facility for upgrading to weapon quality (e.g., ^{233}U denatured with ^{238}U). This material may be less attractive as a starting point for fabricating a weapon than natural uranium;
- *b* identifies material that requires an isotope separation facility such as < 20% ^{235}U in ^{238}U. A country with such separation equipment could make a weapon without diverting reactor fuel;
- *c* refers to highly radioactive material requiring remotely operated engineering equipment for chemically separating weapon material from impurities;
- *d* identifies weapon material that can be separated from impurities through hands-on operations, or material that is in a form suitable for a weapon without additional treatment.

Radiation hazard of a process step is a measure of the danger in handling the fissile material if it were removed from its containment for conversion to a weapon. Also, it can be considered as a measure of the danger associated with the alteration of a process step (piping changes, installation of new equipment, modification of existing equipment, etc. in a confinement area) for diversion of nuclear material. Only a rudimental rating is given in this study since computed data for the candidate fuel cycles has not been completed. The ratings used in this study are necessarily broad in context and are defined as follows:

- *high* identifies a radiation level equivalent to LD_{50} at 30 cm in a few minutes, nominally > 10^4 R/hr;

- *medium* identifies a radiation level capable of producing harmful physiological effects in one day (10 to 10^4 R/hr);
- *low* identifies a radiation level resulting in severe exposure in several days but insufficient to prevent fabrication of a weapon (< 10 R/hr); such exposure could lead eventually to death;
- *negligible* identifies no harmful radiation effects from the material being handled.

In each definition it is tacitly assumed that the radiation level is that associated with the quantity of material required for a weapon or, for in-cell alterations, the total amount of material in the cell. The presence of a highly radioactive isotope of the fissile material, such as ^{232}U in ^{233}U, was assumed to make the fissile material less attractive than fissile material containing radioactive nuclides that can be chemically removed.

1.2 EVALUATION FORMAT

The evaluations of the nuclear fuel cycles named in Table 1.1 are tabulated in the following sections accompanied by the appropriate functional flow diagrams. Separate tables and diagrams are provided for reprocessing and refabrication operations. Items in the tables are keyed to operations shown on the diagrams and are assessed according to the elements discussed in Sect. 1.1. The appraisals of the operations are preliminary diagnoses of factors to be considered in evaluating the relative attractiveness for weapon production of nuclear material diverted from the process step. This study does not rate relative attractiveness since it is premature to do so without also considering the portions of the nuclear fuel cycle that are outside reprocessing and refabrication.

1.3 FUTURE STUDIES

This study has identified a need to develop additional information on the back cycles of the candidate systems to facilitate the choice of the system that has the least proliferation risk. The initial list[1] of fuel cycles did not include fast breeder systems that operate with a thorium-plutonium core and a thorium blanket. Such systems have high

breeding gain, good material performance, possible application in energy centers, and produce ^{233}U fuel. These systems are to be included in the next phase of this study.

The attractiveness of diverted fissile material cannot be fully appreciated without an assessment of the technology and resources needed to produce a weapon. Cost, manpower, time, and technical sophistication are factors to be evaluated, including "quick and dirty" routes from the point of diversion to the ultimate goal. Each functional flow diagram needs to be analyzed (perhaps at Level 2 detail) to identify diversion sidestreams and their required additional treatment.

The analysis should also include an appraisal of the difficulty, cost, manpower, and time to modify existing back cycle facilities to obtain access to fissile material.

Mass flow data, including isotopic compositions, will be developed to give a relationship between spent reactor fuel and weapon capability. In-depth analyses of the back cycles will complement companion studies on reactor performance and resource requirements in order to identify the fuel cycle (or cycles) that offer the greatest proliferation resistance.

1.4 REFERENCES

1. United States Energy Research and Development Administration, *Nonproliferation Alternative Systems Assessment Program - Preliminary Program Plan*, (May 1977).

2. FUEL CYCLE EVALUATIONS FOR URANIUM-PLUTONIUM FUELS IN THE LMFBR AND THE GCFR (Argonne National Laboratory)

M. J. Steindler/D. S. Webster

CORE AND AXIAL BLANKET OF U/Pu FAST BREEDERS

Group 4.1 LMFBR U-Pu Cycle

4.1.1 LMFBR/LWR standard recycle (Reference)

4.1.2 LMFBR/LWR coprocessed, contaminated, recycled oxide fuel

4.1.3 LMFBR/LWR coprocessed, contaminated, recycled carbide fuel

Reprocessing and refabrication of cores and axial blankets for the above options are treated in this section.

Group 5.1 GCFR U-Pu Cycle

5.1.1 GCFR/LWR standard recycle (Reference)

Not treated separately, since the system is essentially identical to option 4.1.1 with regard to both chemistry and proliferation susceptibility. (Since there is no sodium present, step 3.3 in reprocessing is eliminated.)

5.1.2 GCFR/LWR coprocessed, contaminated, recycled oxide fuel

Not treated separately, since it is the same as option 4.1.2

5.1.3 GCFR/LWR coprocessed, contaminated, recycled carbide fuel

Not treated separately - same as option 4.1.3

8.1 Energy center containing a U/Pu LMFBR

A reactor within an energy center is postulated to be free of proliferation possibilities by means of institutional arrangements for the center. Consequently, the LMFBR system is that of option 4.1.1.

8.2 Energy center containing a U/Pu GCFR

This option includes the GCFR of option 5.1.1

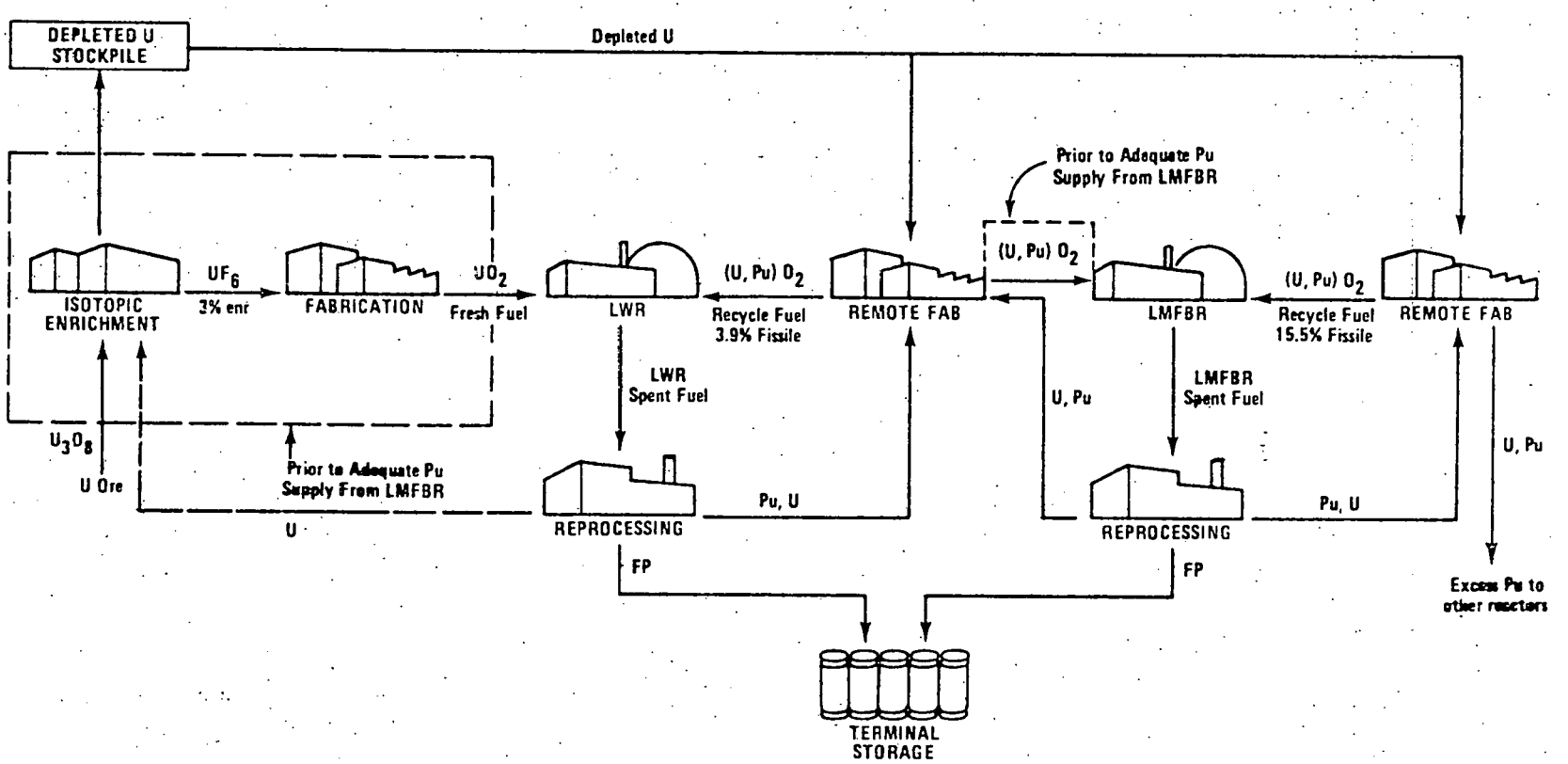
Option 4.1.1 LMFBR/LWR Full U/Pu Recycle, Oxide Fuel

(Reference Case)

Option 4.1.1 is shown in the attached Figure B-11 taken from the draft "Preliminary Plan for Nonproliferation Alternatives Assessment Program." The figure represents two different periods of time. During the first period LWR's are operated with low-enriched uranium fuel (LEU) which is reprocessed to provide both plutonium for fabricating LMFBR cores (and some LWR fuel), and uranium that goes back to the isotope enrichment plant. As construction of LMFBR's progresses, a situation is eventually reached in which no additional LMFBR's are required; during this second period the plutonium bred in excess of that needed to fuel the existing LMFBR's is recycled to LWR's. This fission-power sequence is that envisioned for fast breeders before the recent reconsideration of alternatives, and is consequently a reference case.

The functional flow diagrams (FFD) and process rating tables that follow Figure B-11 deal only with the LMFBR core and axial blanket; the first set of FFD's and tables apply to reprocessing, the second set to refabrication.

The analysis of LWR Fuel reprocessing and refabrication is the same as for option 1.1.1, so will not be repeated here. Reprocessing of the radial blanket is the same as for LWR fuel, but with inclusion of the sodium removal step, operation 3.3 of the attached core assessment.



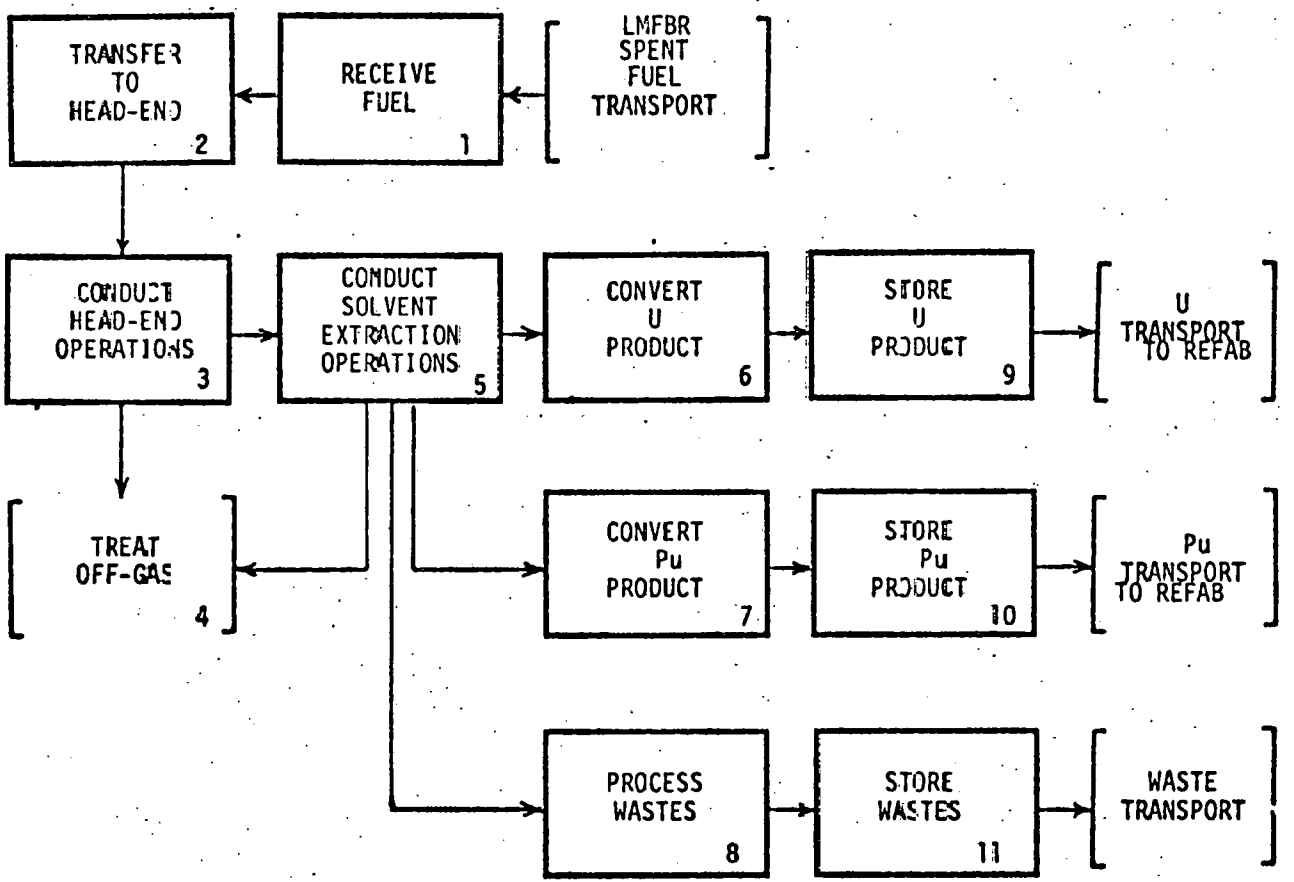
LMFBR/LWR FULL U-Pu RECYCLE

[This Diagram Applies to Option 4.1.1]

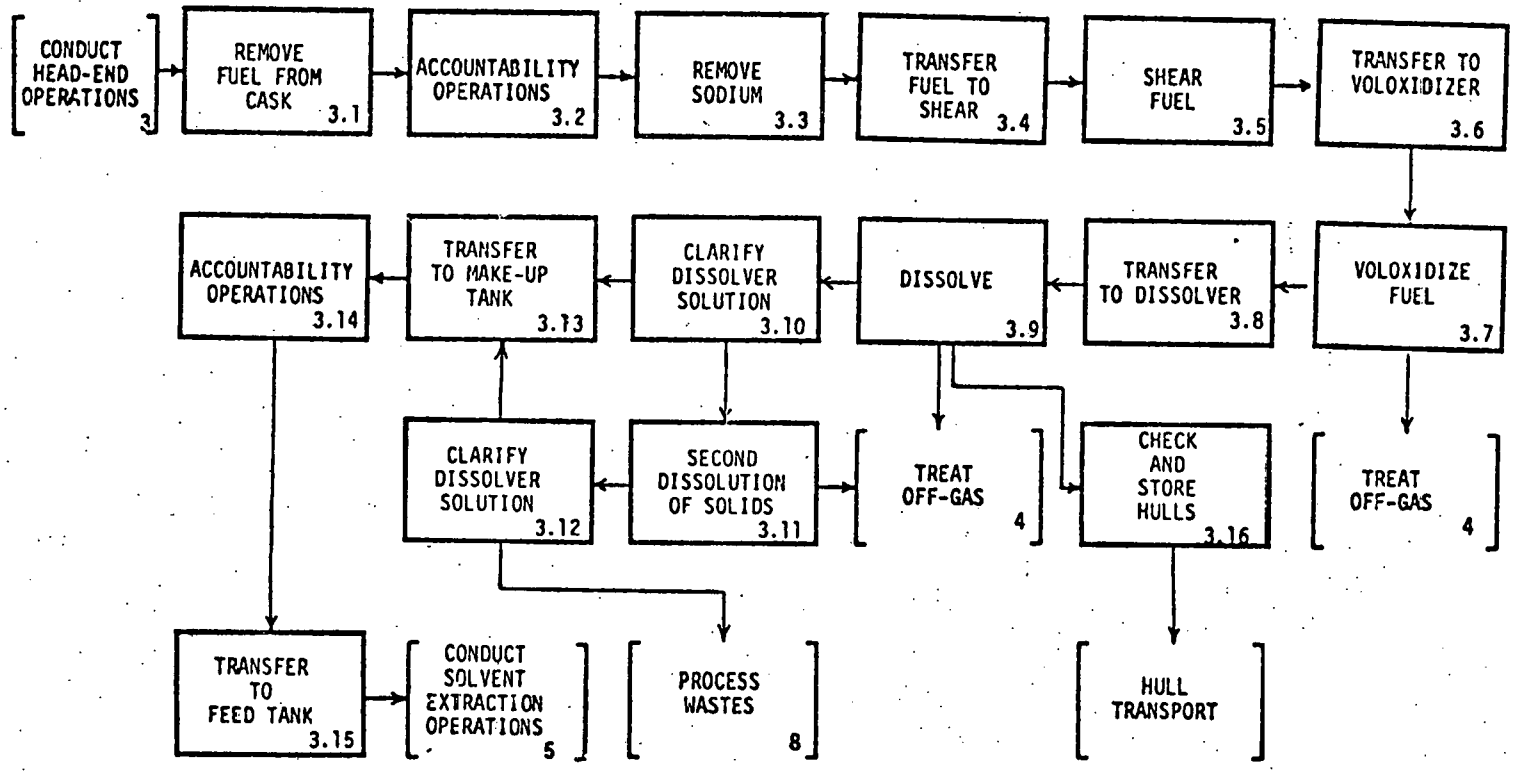
Figure B-11

Option 4.1.1

LMFBR/LWR, U-Pu Recycle--Oxide Fuel
(Reference Case)
Level 0

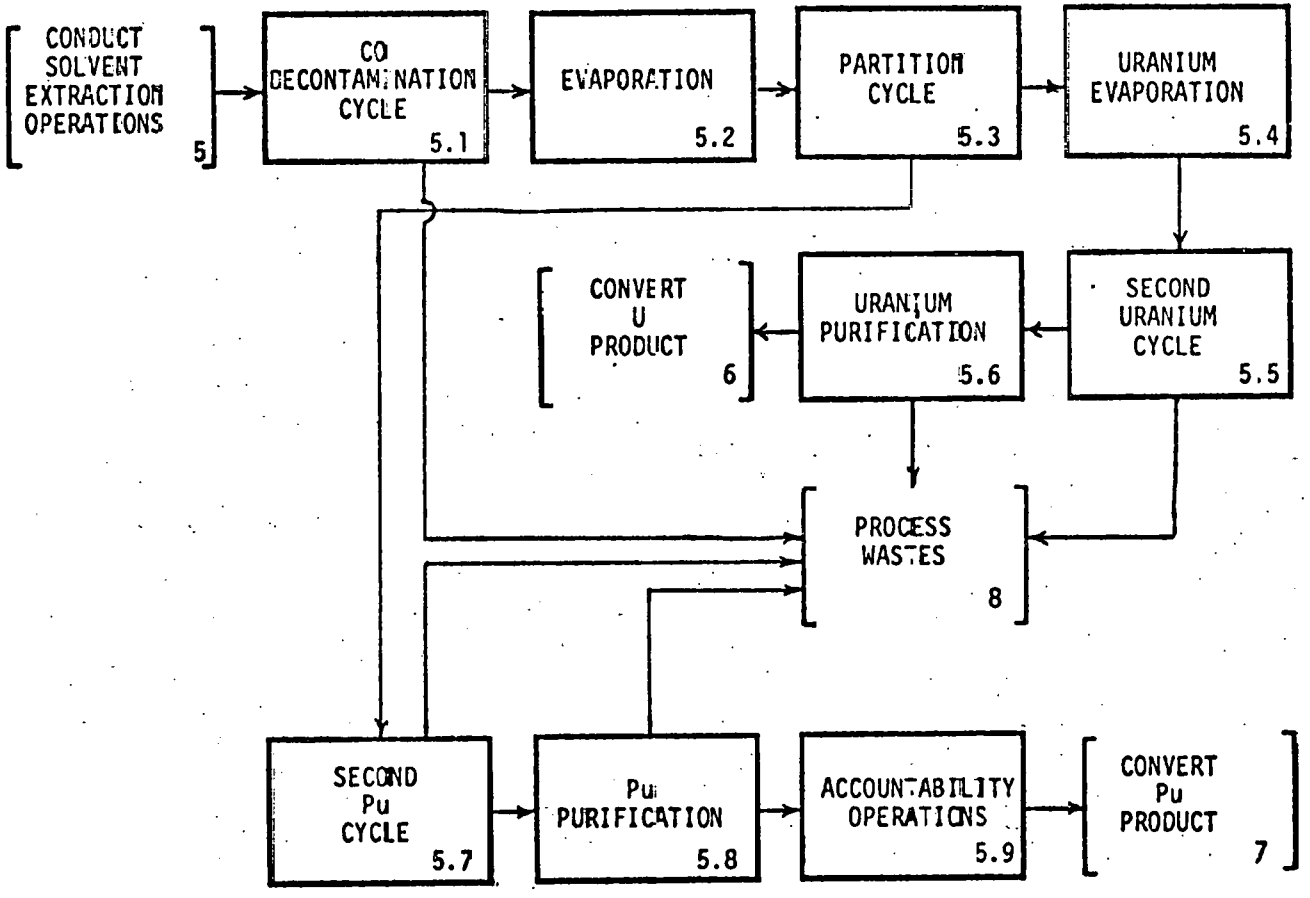


Option 4.1.1
LMFBR/LWR, U-Pu Recycle--Oxide Fuel
(Reference Case)
Level 1



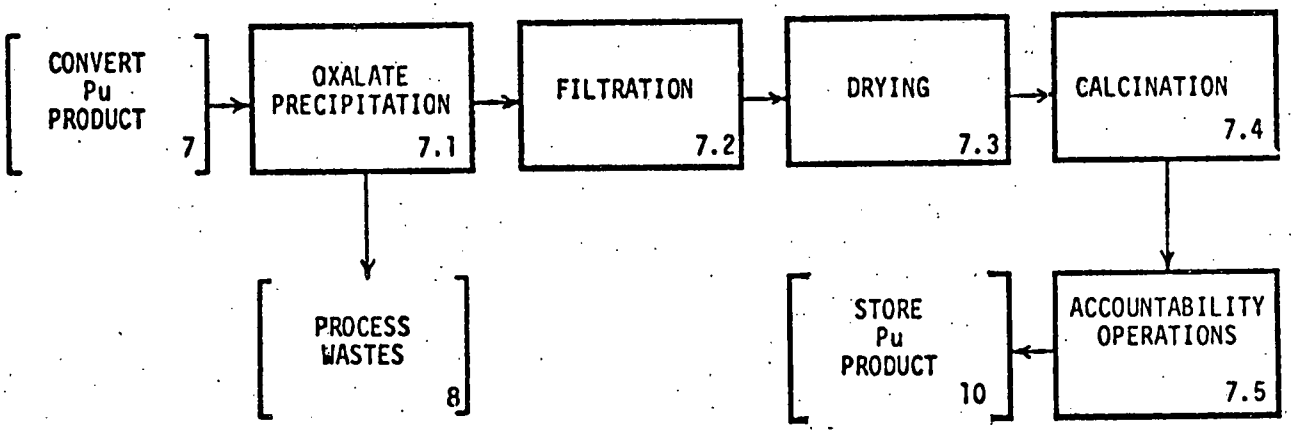
Option 4.1.1

LMFBR/LWR, U-Pu Recycle--Oxide Fuel
(Reference Case)
Level 1



Option 4.1.1

LMFBR/LWR, U-Pu Recycle--Oxide Fuel
(Reference Case)
Level 1



Option 4.1.1 LMFBR/LWR U-Pu Recycle, Oxide Fuel (Reference)

	Operation	Development Needed	Material Location	Material Descriptions	Convertibility	Radiation Hazard
1.	Receive & Store Fuel	Developed	Water Pool	Irra'd Fuel	c	High
2.	Transfer to Head End	Developed	Water Pool	Irra'd Fuel	c	High
3.	Conduct Head-End Operations					
3.1	Remove Fuel	Developed	Hot Cell	Irra'd Fuel	c	High
3.2	Accountability Operations	Developed	Hot Cell	Irra'd Fuel	c	High
3.3	Remove Sodium	Cold Lab	Hot Cell	Irra'd Fuel	c	High
3.4	Transfer Fuel to Shear	Developed	Hot Cell	Irra'd Fuel	c	High
3.5	Shear Fuel	Hot Engineering	Hot Cell	Fuel/Hulls	c	High
3.6	Transfer to Voloxidizer	Hot Engineering	Hot Cell	Fuel/Hulls	c	High
3.7	Voloxidize Fuel	Hot Engineering	Hot Cell	Powder/Hulls	c	High
3.8	Transfer to Dissolver	Developed	Hot Cell	Powder/Hulls	c	High
3.9	Dissolve	Developed	Hot Cell	Solution/Residue/Hulls	c	High
3.10	Clarify Dissolver Solutions	Developed	Hot Cell	Solution/Residue/Hulls	c	High
3.11	Second Dissolution of Solids	Cold Engineering	Hot Cell	Solution/Residue/Hulls	c	High
3.12	Clarify Dissolver Solution	Developed	Hot Cell	Solution/Residue/Hulls	c	High
3.13	Transfer to Make-up Tank	Developed	Hot Cell	U/Pu Solution	c	High
3.14	Accountability Operations	Developed	Hot Cell	U/Pu Solution	c	High
3.15	Transfer to Feed Tank	Developed	Hot Cell	U/Pu Solution	c	High
3.16	Check & Store Hulls	Developed	Hot Cell/ Water Pool	Hulls	NonFissionable	High
4.	Treat Off Gas	Hot Engineering	Hot Cell	Gas, Particulate	NonFissionable	High
5.	Conduct Solvent Extraction Operations					
5.1	Co-Decontamination Cycle	Developed	Hot Cell	U-Pu Solution	c	High
5.2	Evaporation	Developed	Hot Cell	U-Pu Solution	c	Medium
5.3	Partition Cycle	Developed	Hot Cell	U-Pu Solution	c/d	Medium
5.4	Uranium Evaporation	Developed	Hot Cell	U Solution	NonFissionable	Medium
5.5	Second Uranium Cycle	Developed	Hot Cell	U Solution	NonFissionable	Low
5.6	U Purification	Developed	Hands-on-Fac	U Solution	NonFissionable	Negl.
5.7	Second Pu Cycle	Developed	α Facility	Pu Solution	d	Medium
5.8	Pu Purification	Developed	α Facility	Pu Solution	d	Low
5.9	Accountability Operations	Developed	α Facility	Pu Solution	d	Low
6.	Convert U Product	Developed	Hands-on-Fac	U Solution/UO_2 Powder	NonFissionable	Negl.
7.	Convert Pu Product					
7.1	Oxalate Precipitation	Developed	α Facility	Pu Solution/Pu Precipitate	d	Low
7.2	Filtration	Developed	α Facility	Solution/Pu Precipitate	d	Low
7.3	Drying	Developed	α Facility	Pu Filter Cake	d	Low
7.4	Calcination	Developed	α Facility	PuO_2	d	Low
7.5	Accountability Operations	Developed	α Facility	PuO_2	d	Low
8.	Process Wastes	Prototype	Hot Cell	Solution/Solids	NonFissionable	Medium/High
9.	Store U Product	Developed	Hands-on-Fac	UO_2 Powder	NonFissionable	Negl.
10.	Store Pu Product	Developed	Hands-on-Fac	PuO_2 Powder	d	Low
11.	Store Wastes	Prototype, Hot Engineering	Buried tanks, Water pool	Solution/Solids	NonFissionable	Medium/High

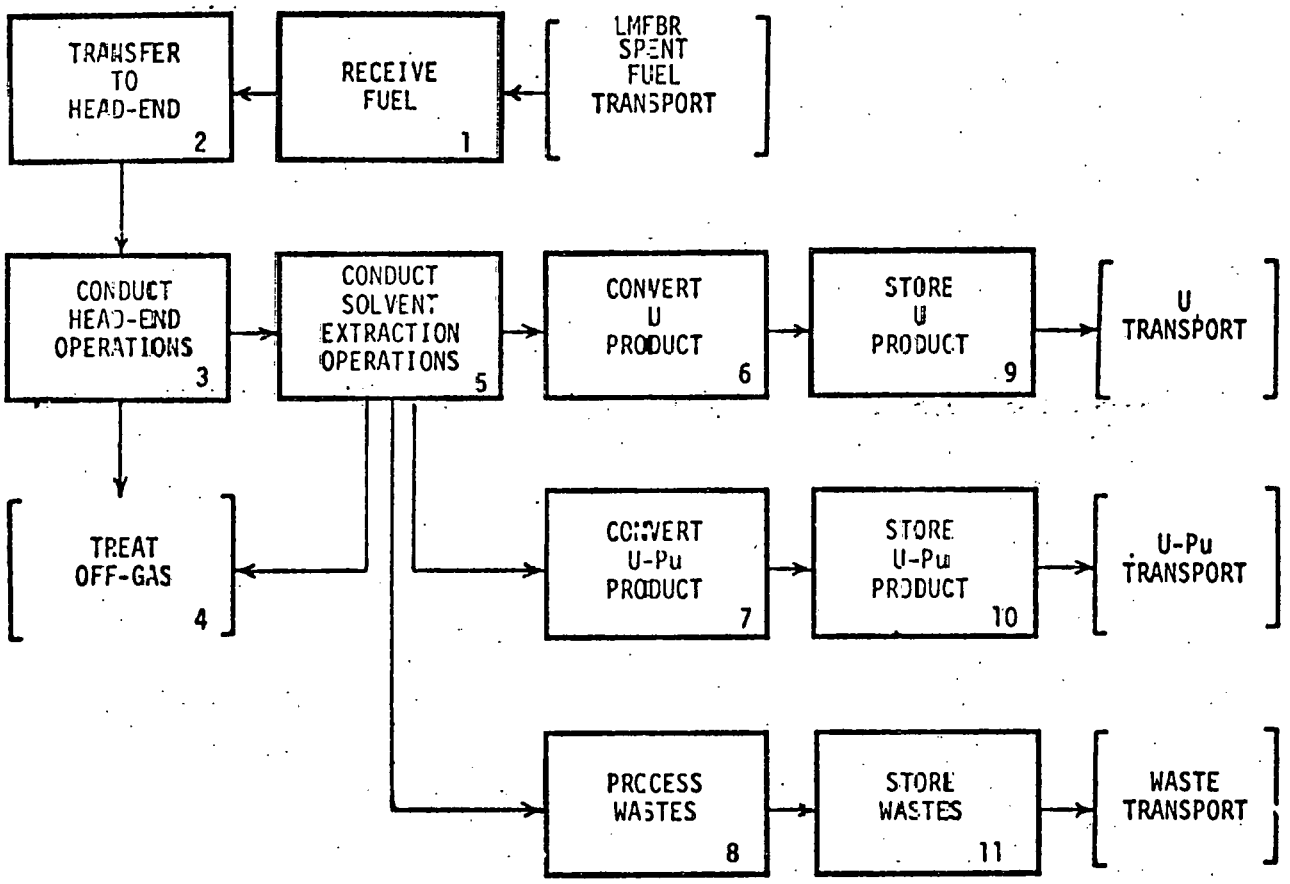
Option 4.1.2 LMFBR/LWR, U/Pu Coprocessed, contaminated, Recycled oxide Fuel

This option is similar to option 4.1.1 except that plutonium is always in the presence of uranium and enough fission products to make diversion difficult.

The following functional flow diagrams and process rating tables deal only with the LMFBR core and axial blanket: the first set of FFD's and tables apply to reprocessing, the second set to refabrication.

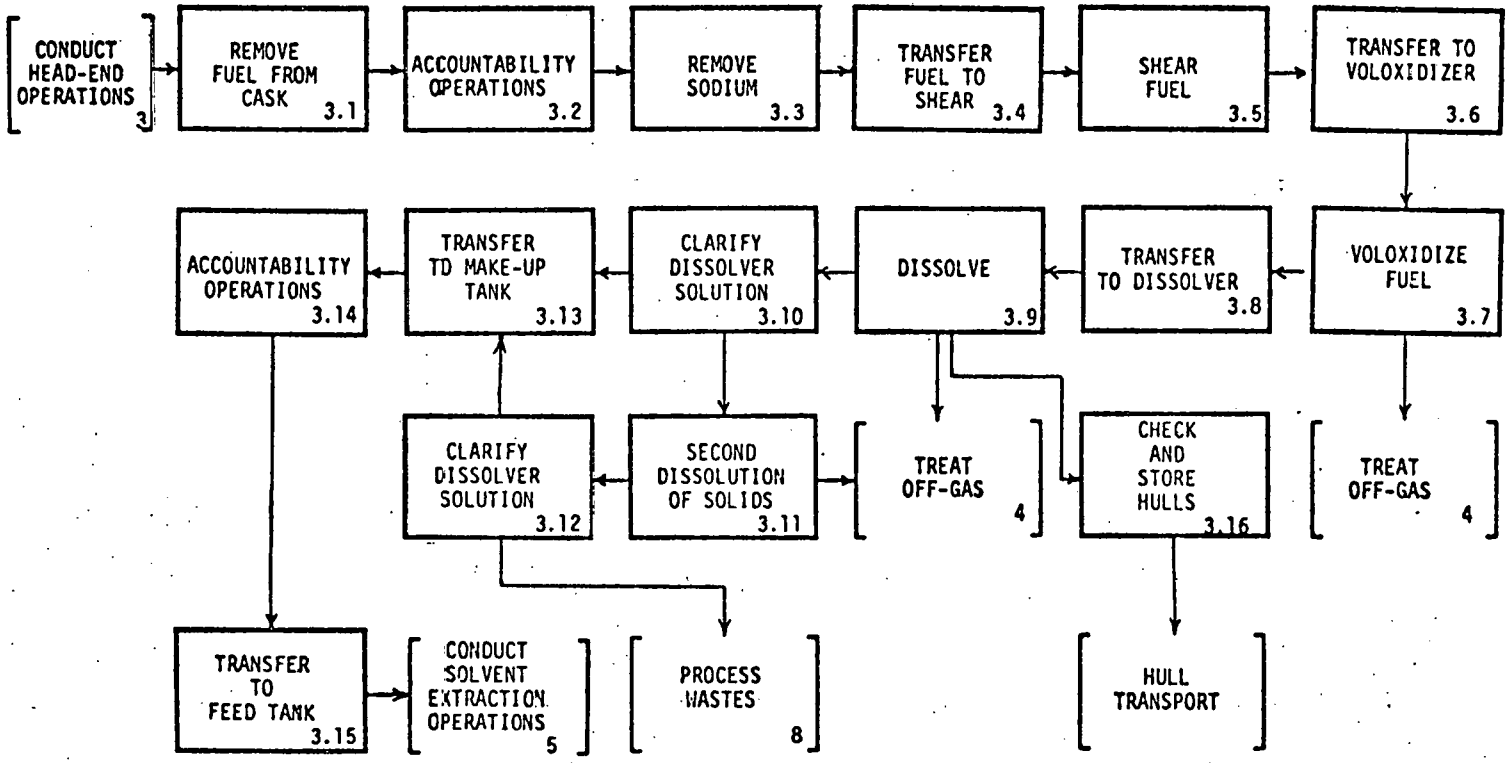
The assessment of LWR fuel reprocessing and refabrication is the same as for option 1.1.3, so will not be repeated here.

Option 4.1.2
LMFBR/LWR, U-Pu Coprocessed, Contaminated, and Recycled--Oxide Fuel
Level 0



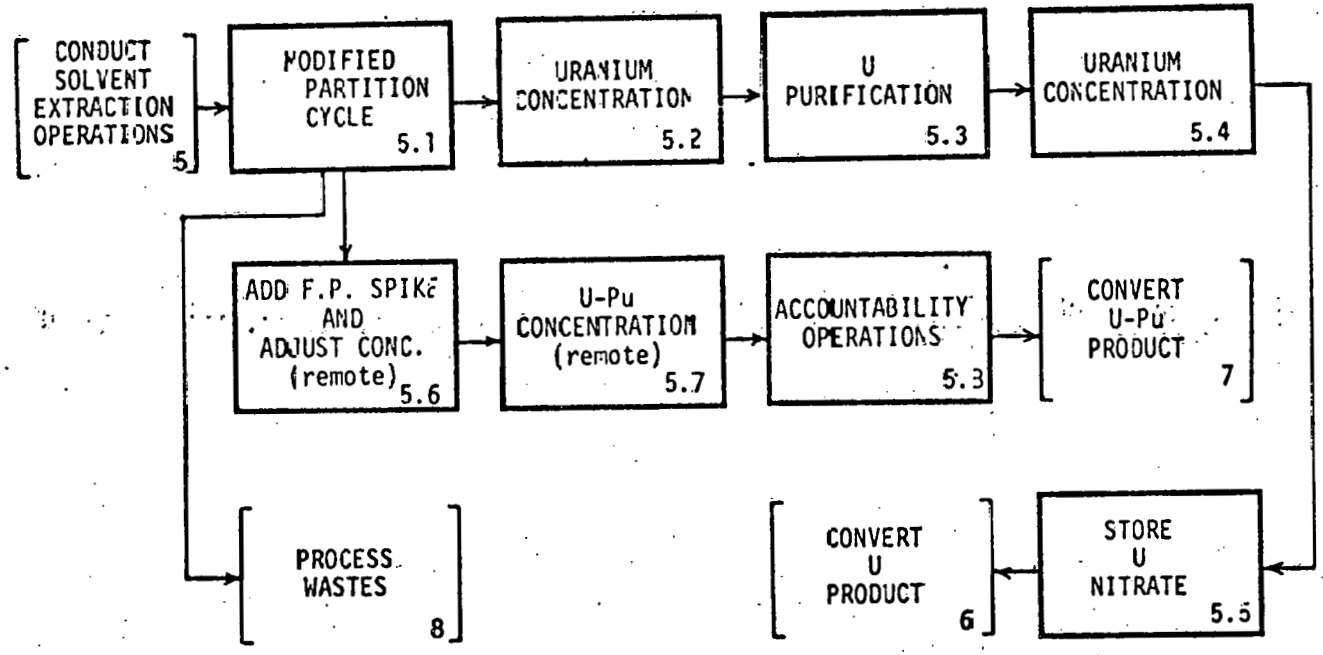
Option 4.1.2

LMFBR/LWR, U-Pu Coprocessed, Contaminated, and Recycled--Oxide Fuel
Level 1

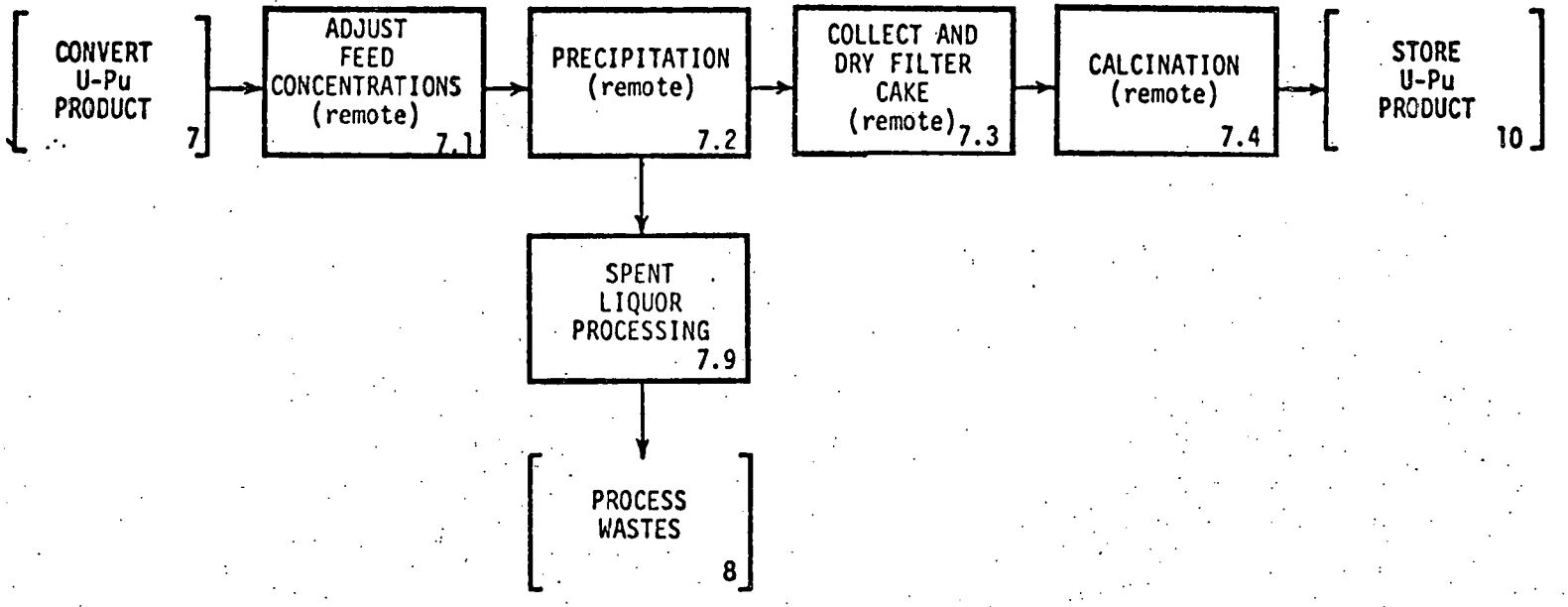


Option 4.1.2

LMFBR/LWR, U-Pu Coprocessed, Contaminated, and Recycled--Oxide Fuel
Level 1



Option 4.1.2
LMFBR/LWR, U-Pu Coprocessed, Contaminated,
and Recycled--Oxide Fuel
Level 1



Option 4.1.2 LMFBR/LWR U-Pu Coprocessed, Recycled Oxide Fuel

	Operation	Development Needed	Material Location	Material Descriptions	Convertibility	Radiation Hazard
1.	Receive & Store Fuel	Developed	Shielded Water Pool	Irradiated Fuel	c	High
2.	Transfer to Head End	Developed	Shielded Water Pool	Irradiated Fuel	c	High
3.	Conduct Head-End Operations					
3.1	Remove Fuel from Cask	Developed	Hot Cell	Irradiated Fuel	c	High
3.2	Accountability Operations	Developed	Hot Cell	Irradiated Fuel	c	High
3.3	Remove Sodium	Cold Lab	Hot Cell	Irradiated Fuel	c	High
3.4	Transfer Fuel to Shear	Developed	Hot Cell	Irradiated Fuel	c	High
3.5	Shear Fuel	Hot Engineering	Hot Cell	Hot MO Fuel Fragments, Hulls	c	High
3.6	Transfer to Voloxidizer	Hot Engineering	Hot Cell	Hot MO Fuel Fragments, Hulls	c	High
3.7	Voloxidize Fuel	Hot Engineering	Hot Cell	Hot MO Powder/Hulls	c	High
3.8	Transfer to Dissolver	Developed	Hot Cell	Hot MO Powder/Hulls	c	High
3.9	Dissolve	Developed	Hot Cell	Solution/Solids Residue/Hulls	c	High
3.10	Clarify Dissolver Solutions	Developed	Hot Cell	Solution/Solids Residue/Hulls	c	High
3.11	Second Dissolution of Solids	Hot Engineering	Hot Cell	Solution/Solids Residue/Hulls	c	High
3.12	Clarify Dissolver Solution	Developed	Hot Cell	Solution/Solids Residue/Hulls	c	High
3.13	Transfer to Make-up Tank	Developed	Hot Cell	Solution, U-Pu	c	High
3.14	Accountability Operations	Developed	Hot Cell	Solution, U-Pu	c	High
3.15	Transfer to Feed Tank	Developed	Hot Cell	Solution, U-Pu	c	High
3.16	Check & Store Hulls	Developed	Hot Cell/Shielded Water Pool	Hulls	NonFissionable	High
4.	Treat Off Gas	Hot Engineering	Hot Cell	Gas, Particulate	NonFissionable	High
5.	Conduct Solvent Extraction Operations					
5.1	Modified Partition Cycle	Cold Lab	Hot Cell	U-Pu Solution	c	High
5.2	Uranium Concentration	Developed	Hot Cell	U Solution	NonFissionable	Medium
5.3	Uranium Purification	Developed	Hot Cell	U Solution	NonFissionable	Medium
5.4	Uranium Concentration	Developed	Hands-on-Fac	U Solution	NonFissionable	Low
5.5	Store U-nitrate	Developed	Hands-on-Fac	U Solution	NonFissionable	Low
5.6	Add FP spike & Adjust Conc	Cold Lab	Shielded α Fac	U-Pu Solution	c	High
5.7	U-Pu Concentration	Cold Lab	Shielded α Fac	U-Pu Solution	c	High
5.8	Accountability Operations	Developed	Shielded α Fac	U-Pu Solution	c	High
6.	Convert U Product	Developed	Hands-on-Fac	U Solution/Solid	NonFissionable	Low
7.	Convert U-Pu Product					
7.1	Adjust Feed Concentration	Developed	Shielded α Fac	U-Pu Solution	c	High
7.2	Precipitation	Hot Lab	Shielded α Fac	U-Pu Solution/U-Pu Precipitate	c	High
7.3	Collect & Dry Filter Cake	Hot Lab	Shielded α Fac	U-Pu Oxide	c	High
7.4	Calcination	Hot Lab	Shielded α Fac	U-Pu Oxide	c	High
7.9	Spent Liquor Processing	Hot Lab	Shielded α Fac	Solution	NonFissionable	Medium
8.	Process Wastes	Prototype	Hot Cell	Solution, Solids	NonFissionable	Medium/ High
9.	Store U Product	Developed	Hands-on-Fac	UO_2 Powder	NonFissionable	Negl.
10.	Store U-Pu Product	Developed	Hot Cell	UO_2-PuO_2 Powder	c	High
11.	Store Wastes	Prototype, Hot Engineering	Buried tanks, Water Pool	Solution, Solid	NonFissionable	Medium/ High

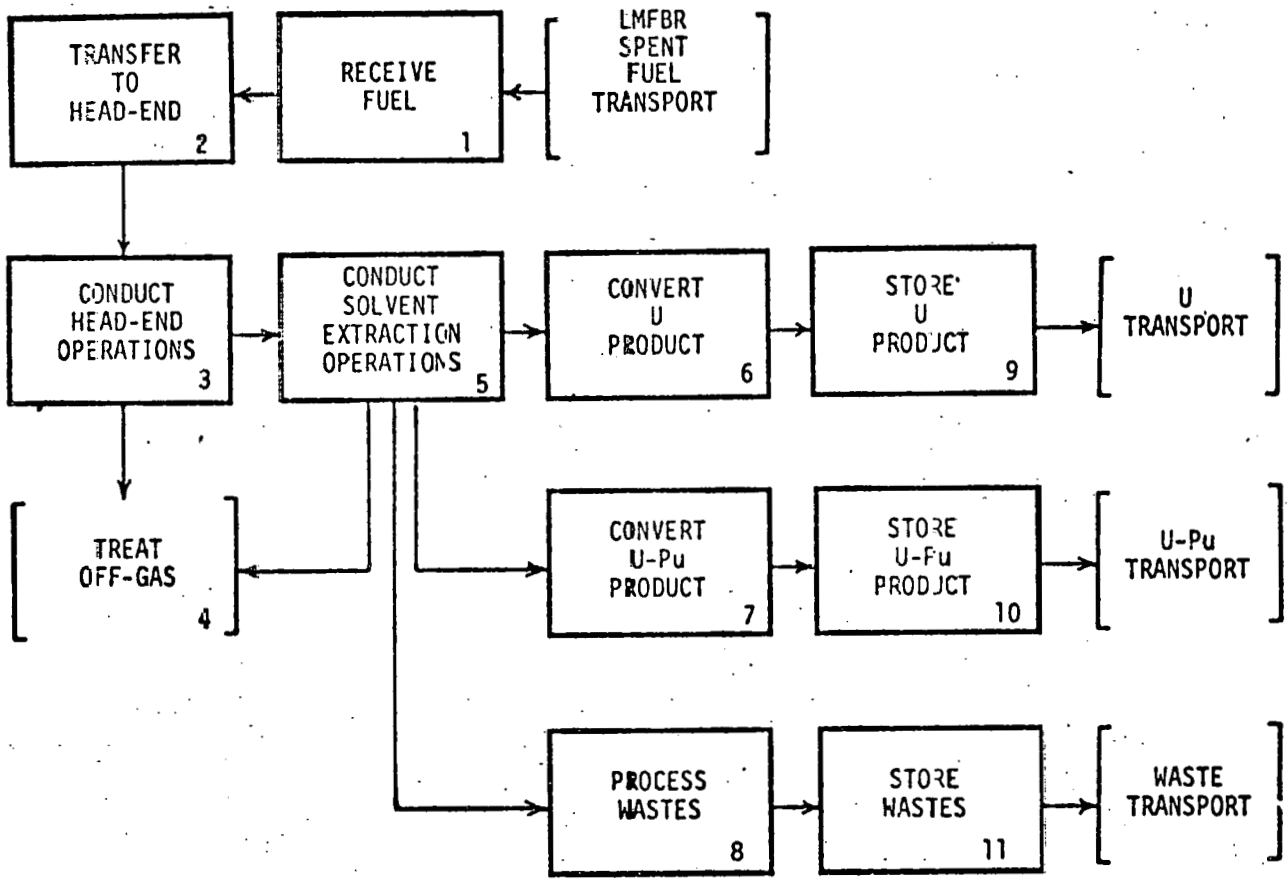
Option 4.1.3 LMFBR/LWR, U/Pu Coprocessed, Contaminated, Recycled Carbide Fuel

This option is similar to option 4.1.2--coprocessing and contamination with fission products--except that the LMFBR core and axial blanket are carbides rather than oxides.

Functional flow diagrams and process rating tables follow for reprocessing and refabrication of LMFBR core and axial blanket.

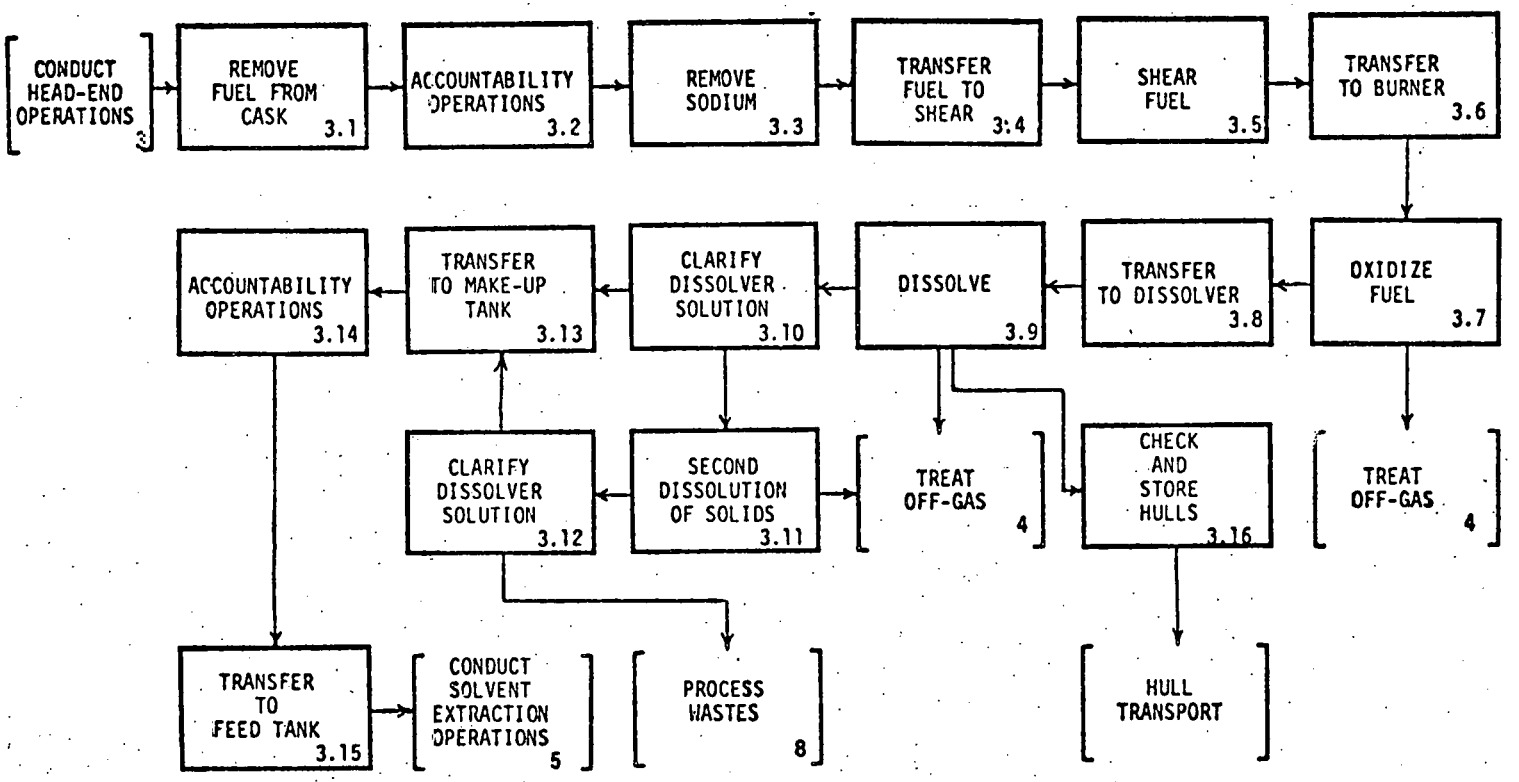
The LWR assessment is the same as for option 4.1.2; the only change is that during the first period (LMFBR buildup) material from LWR processing is converted to carbide before use in the LMFBR.

Option 4.1.3
LMFBR/LWR, U-Pu Coprocessed, Contaminated,
and Recycled--Carbide Fuel
Level 0

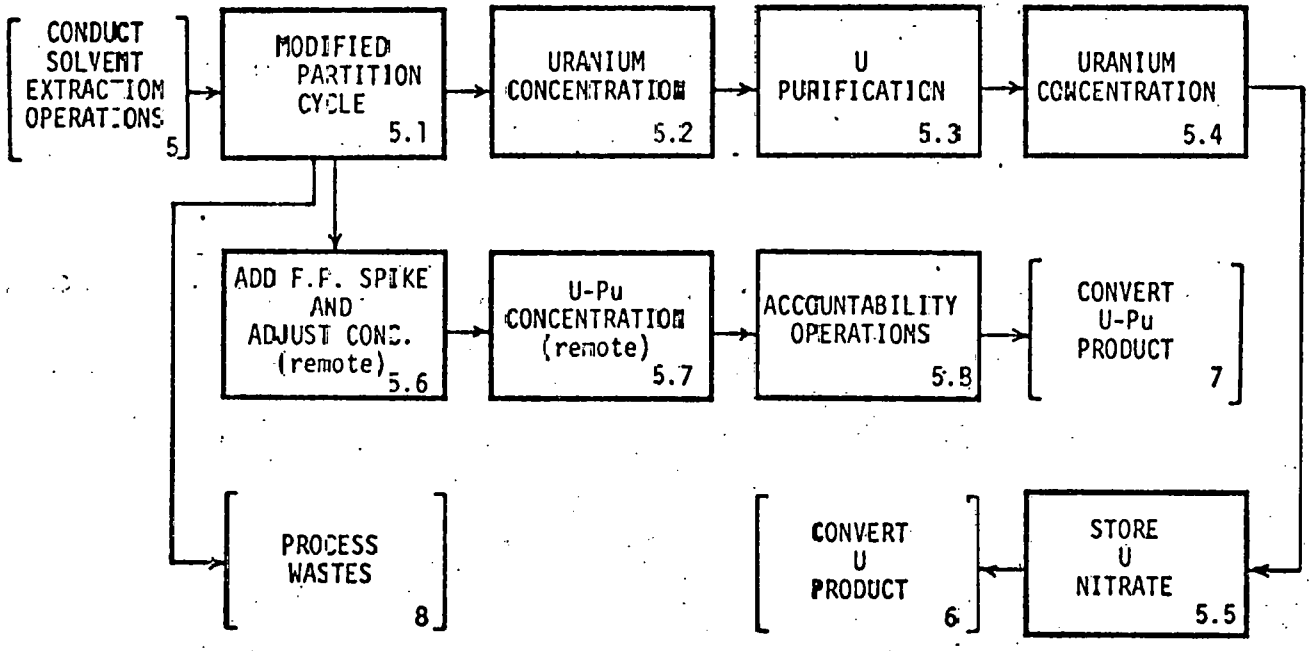


Option 4.1.3

LMFBR/LWR, U-Pu Coprocessed, Contaminated, and Recycled--Carbide Fuel
Level 1

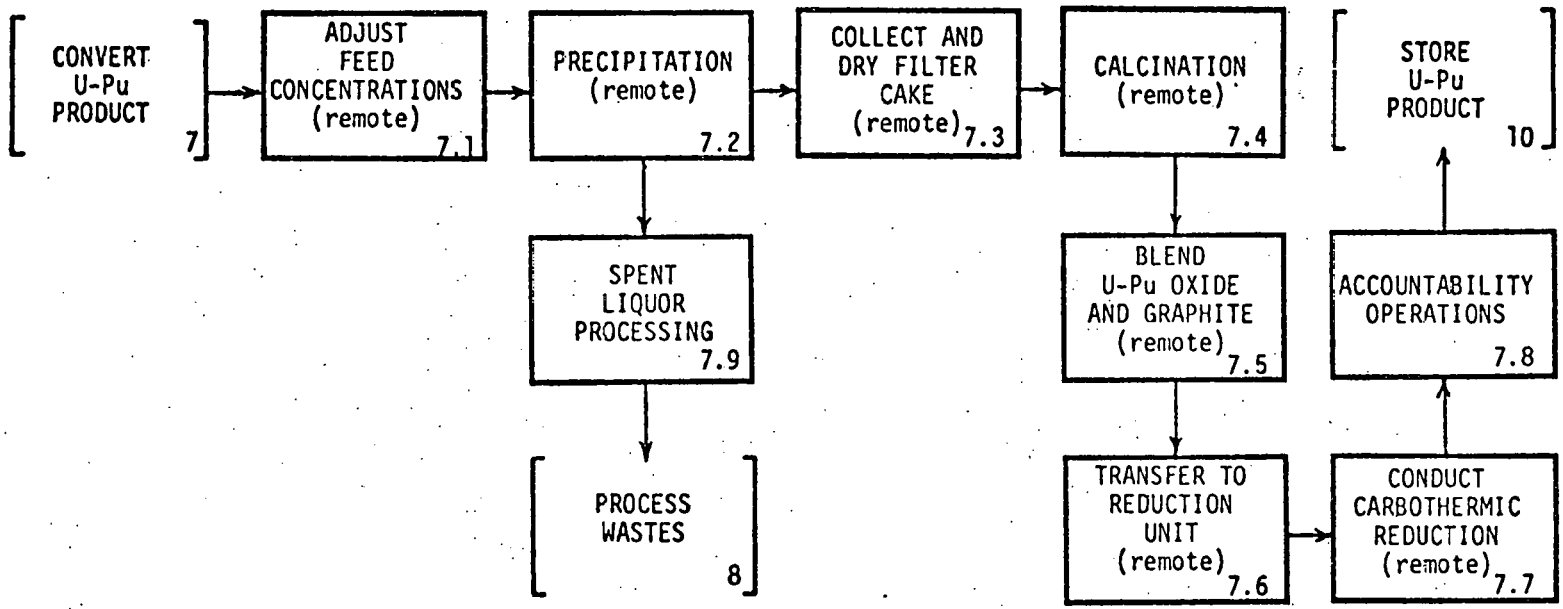


Option 4.1.3
LMFBR/LWR, U-Pu Coprocessed, Contaminated,
and Recycled--Carbide Fuel
Level 1



Option 4.1.3

LMFBR/LWR, U-Pu Coprocessed, Contaminated, and Recycled--Carbide Fuel
Level 1



Option 4.1.3 LMFBR/LWR U/Pu Coprocessed, Recycled Carbide Fuel

	Operation	Development Needed	Material Location	Material Descriptions	Convertibility	Radiation Hazard
1.	Receive & Store Fuel	Developed	Shielded Water Pool	Irradiated Fuel	c	High
2.	Transfer to Head End	Developed	Shielded Water Pool	Irradiated Fuel	c	High
3.	Conduct Head-End Operations					
3.1	Remove Fuel from Cask	Developed	Hot Cell	Irradiated Fuel	c	High
3.2	Accountability Operations	Developed	Hot Cell	Irradiated Fuel	c	High
3.3	Remove Sodium	Cold Lab	Hot Cell	Irradiated Fuel	c	High
3.4	Transfer Fuel to Shear	Developed	Hot Cell	Irradiated Fuel	c	High
3.5	Shear Fuel	Hot Engineering	Hot Cell	Hot MC Fuel Fragments/Hulls	c	High
3.6	Transfer to Burner	Hot Engineering	Hot Cell	Hot MC Fuel Fragments/Hulls	c	High
3.7	Oxidize Fuel	Hot Engineering	Hot Cell	Hot MO Powder/Hulls	c	High
3.8	Transfer to Dissolver	Developed	Hot Cell	Hot MO Powder/Hulls	c	High
3.9	Dissolve	Developed	Hot Cell	Solution/Solids Residue/Hulls	c	High
3.10	Clarify Dissolver Solutions	Developed	Hot Cell	Solution/Solids Residue/Hulls	c	High
3.11	Second Dissolution of Solids	Hot Engineering	Hot Cell	Solution/Solids Residue/Hulls	c	High
3.12	Clarify Dissolver Solution	Developed	Hot Cell	Solution/Solids Residue/Hulls	c	High
3.13	Transfer to Make-up Tank	Developed	Hot Cell	Solution, U-Pu	c	High
3.14	Accountability Operations	Developed	Hot Cell	Solution, U-Pu	c	High
3.15	Transfer to Feed Tank	Developed	Hot Cell	Solution, U-Pu	c	High
3.16	Check & Store Hulls	Developed	Hot Cell/Shielded Water Pool	Hulls	NonFissionable	High
4.	Treat Off Gas	Hot Engineering	Hot Cell	Gas, Particulate	NonFissionable	High
5.	Conduct Solvent Extraction Operations					
5.1	Modified Partition Cycle	Cold Lab	Hot Cell	U-Pu Solution	c	High
5.2	Uranium Concentration	Developed	Hot Cell	U Solution	NonFissionable	Medium
5.3	Uranium Purification	Developed	Hot Cell	U Solution	NonFissionable	Medium
5.4	Uranium Concentration	Developed	Hands-on-Fac	U Solution	NonFissionable	Low
5.5	Store U-nitrate	Developed	Hands-on-Fac	U Solution	NonFissionable	Low
5.6	Add FP spike & Adjust Conc	Cold Lab	Shielded α Fac	U-Pu Solution	c	High
5.7	U-Pu Concentration	Cold Lab	Shielded α Fac	U-Pu Solution	c	High
5.8	Accountability Operations	Developed	Shielded α Fac	U-Pu Solution	c	High
6.	Convert U Product	Developed	Hands-on-Fac	U Solution/Solid	NonFissionable	Low
7.	Convert U-Pu Product					
7.1	Adjust Feed Concentration	Developed	Shielded α Fac	U-Pu Solution	c	High
7.2	Precipitation	Hot Lab	Shielded α Fac	U-Pu Solution/U-Pu Precipitate	c	High
7.3	Collect & Dry Filter Cake	Hot Lab	Shielded α Fac	U-Pu Oxide	c	High
7.4	Calcination	Hot Lab	Shielded α Fac	U-Pu Oxide	c	High
7.5	Blend U-Pu Oxide & Graphite	Hot Lab	Shielded α Fac	U-Pu Oxide/Graphite	c	High
7.6	Transfer to Reduction	Hot Lab	Shielded α Fac	U-Pu Oxide/Graphite	c	High
7.7	Conduct Carbothermic Reduct	Hot Lab	Shielded α Fac	U-Pu Carbide	c	High
7.8	Accountability Operations	Hot Lab	Shielded α Fac	U-Pu Carbide	c	High
7.9	Spent Liquor Processing	Hot Lab	Shielded α Fac	Solution	NonFissionable	Medium

Option 4.1.3 LMFBR/LWR U/Pu Coprocessed, Recycled Carbide Fuel (Cont'd)

	Operation	Development Needed	Material Location	Material Descriptions	Convertibility	Radiation Hazard
8.	Process Wastes	Prototype	Hot Cell	Solution/Solids	NonFissionable	Medium/ High
9.	Store U Product	Developed	Hands-on-Fac	UO_2 Powder	NonFissionable	Negl.
10.	Store U-Pu Product	Developed	Hot Cell	UO_2-PuO_2 Powder	c	High
11.	Store Wastes	Prototype, Hot Engineering	Buried tanks, Water Pool	Solution/Solid	NonFissionable	Medium/ High

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3. FUEL CYCLE EVALUATIONS FOR URANIUM-PLUTONIUM FUELS IN THE LMFBR AND LMFBR/LWR (Hanford Engineering Development Laboratory)

H. C. F. Ripfel
M. J. Barr
S. R. Fields
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R. L. Plum
D. D. Scott

3.1 DISCUSSION

This chapter covers the fabrication of LMFBR fuel for the following cases:

- o LMFBR/LWR Standard Recycle - Case 29
- o LMFBR/LWR Coprocessed (Contaminated) Oxide - Case 30
- o LMFBR/LWR Coprocessed (Contaminated) Carbide - Case 31
- o LMFBR With U/Pu Core, Th Blanket, Inside Secure Energy Center; LMFBR With Denatured U Core, Th Blanket, Outside Energy Center - Case 66

3.1.1 LMFBR/LWR Standard Recycle - Case 29

This alternate using low (~12%) ^{240}Pu feed is established technology. It is included in the analysis as a "benchmark" case. Cost and process engineering data exist to be used as a baseline for evaluation of other cycles and extension of Case 29 to include the use of high (>20%) ^{240}Pu feed. Improvement in the existing proliferation resistance of this case can be obtained through modification of feed materials and process operations and improvements in material accountability and process line concepts.

3.1.2 LMFBR/LWR Coprocessed (Contaminated) Oxide - Case 30

This case is differentiated from Case 29 by the inclusion of gamma radioactive materials in the fuel. This measure increases proliferation resistance, particularly against diversion by a subnational (terrorist) group. Also, if diversion occurs, the radioactive "spiking" material hinders subsequent handling and processing of the fuel in the weapon factory. Removal of the spiking material would require use of a shielded chemical processing facility.

The required level of radioactivity induced by spiking is yet to be determined as is the method of spiking. For the purposes of this study it was arbitrarily assumed that:

- o A spikant already existed in incoming Pu or MOX feed.
- o The spikant was small quantities (<0.1%) and would not adversely affect processing or fuel behavior.
- o The spikant would be a high gamma emitter with a half life of about 1 year or more.
- o The radiation would be high enough to debilitate diverters with exposure to 1 KG for 1 HR.

The impact of spiking on fuel fabrication complexity, equipment requirements, personnel radiation exposure and cost is yet to be determined. The economics of the fuel cycle, although secondary in priority to nonproliferation, may ultimately determine whether the fuel cycle is viable.

3.1.3 LMFBR/LWR Coprocessed (Contaminated) Carbide - Case 31

This case is very similar to Case 30 except that it has a very few additional process steps; e.g., the addition of a carbothermic reduction and modified waste recycle process. It has been concluded that the proliferation aspects of these two cases will not differ significantly from each other and the two cases may be combined in this study; i.e., Case 31 is a "derived case."

3.1.4 LMFBR With U/Pu Core, Th Blanket, Inside Secure Energy Center; LMFBR With Denatured U Core, Th Blanket, Outside Energy Center - Case 66

The type of fuel used for the core of the "inside" LMFBR is similar to that of Case 29.

It is to be noted that the method of analysis incorporates no way of giving credit to the energy center concept for its antiproliferation value. "Accessibility" refers to the ease or difficulty of access to material usable for fabricating a nuclear weapon. The rating terms suggested by I. Spiewak, ORNL, to Saul Strauch, ERDA DNA, in his letter of August 3, 1977 (remote processing cell, hot cell, glovebox, shielded alpha facility and hands-on facility) do not allow consideration of other important factors such as the level and type of plant safeguard measures and security.

3.2 PROLIFERATION RESISTANCE FACTORS AND DEVELOPMENT REQUIREMENTS

The proliferation resistance factors and development needs for each of the four cases treated in this chapter are shown in Tables 3.1 through 3.4. The rating method used was the one described in the introduction to this report.

3.3 FUNCTIONAL PROCESS FLOW DIAGRAMS

Level I and Level II functional process flow diagrams for Cases 29, 30, 31 and 66 are shown in Figures 3.1 through 3.30. A level-0 diagram is provided for Case 66 to show how the fuel-fabrication process fits into the rather complex fuel cycle.

Table 3.1. Case 29, 3.2.3.1, LMFBR U/Pu Recycle (Reference Case) Fuel Fabrication

Process Step	Operation	Material Description	Development Needed	Material Location	Convertibility	Radiation Hazard
1.	Feed Material	UO_2, PuO_2 and MOX powder	Some(#)	Glovebox(Δ)	d*	Low(+)
2.	Fuel Fabrication	MOX powder and pellets	Some(#)	Glovebox(Δ)	d	Low(+)
3.	Pin Fabrication	MOX pellets and pins	Some(#)	Glovebox(Δ)	d	Low(+)
4.	Bundle Assembly	MOX pins and assemblies	Moderate(#)	Limited Access Assembly Area	d	Low(+)
5.	Scrap Recycle	MOX powder and pellets, solutions of Pu and U	Little**	Glovebox	d	Low**
6.	Waste Reprocess	Solid and liquid wastes containing low-level Pu and U contamination	Little**	Glovebox	d	Low**

(#) Processes are currently used in industry with low (~12%) ^{240}Pu material. Hot prototype needed for automated operation and use of ^{233}U or high (>20%) ^{240}Pu.

(*) UO_2 is nonfissionable.

(+) Negligible for normal or depleted UO_2. May be higher for $^{233}UO_2$.

(Δ) Containment with mechanized/automated operation.

(**) Fuel containing ^{233}U may need an additional separations step.

Table 3.2. Case 30, 3.2.3.2, LMFBR/LWR U/Pu
Coprocessed, Contaminated and Recycled Oxide Fuel

Process Step	Operation	Material Description	Development Needed	Material Location	Convertibility	Radiation Hazard
1.	Feed Material	UO_2, PuO_2 and MOX powder, contaminated	Hot Lab	Hot cell[+]	c[*]	medium to high
2.	Fuel Fabrication	MOX powder and pellets, contaminated	Hot Lab	Hot cell	c	medium to high
3.	Pin Fabrication	MOX pellets and pins, contaminated	Hot Lab	Hot cell	c	medium to high
4.	Bundle Assembly	MOX pins and assemblies, contaminated	Hot Lab	Hot cell	c	medium to high
5.	Scrap Recycle	MOX powder and pellets, solutions of Pu and U, all contaminated	Hot Engineering	Hot cell	c	medium to high
6.	Waste Reprocess	Solid and liquid wastes containing low-level Pu, U and contamination	Hot Engineering	Hot cell	c	medium to high

(+) Containment including local shielding with automated operation.

(*) UO_2 is nonfissionable.

Table 3.3. Case 31, 3.2.3.2, LMFBR/LWR U/Pu
Coprocessed, Contaminated and Recycled Carbide Fuel

Process Step	Operation	Material Description	Development Needed	Material Location	Convertibility	Radiation Hazard
1.	Feed Material	UO_2, PuO_2, UC, PuC, MOX and MC powder, contaminated	Hot Lab	Hot cell+	c*	medium
2.	Fuel Fabrication	MC powder and pellets, contaminated	Hot Lab	Hot cell	c	medium
3.	Pin Fabrication	MC pellets and pins, contaminated	Hot Lab	Hot cell	c	medium
4.	Bundle Assembly	MC pins and assemblies, contaminated	Hot Lab	Hot cell	c	medium
5.	Scrap Recycle	MOX and MC powder and pellets, solutions of Pu and U, all contaminated	Hot Engineering	Hot cell	c	medium
6.	Waste Reprocess	Solid and liquid wastes containing low-level Pu, U and gamma emitting contamination	Hot Engineering	Hot cell	c	medium

(+) Containment including local shielding with automated operation.

(*) UO_2 and UC are nonfissionable.

Table 3.4 Case 66, 3.4.2.6, Energy Center Containing LMFBR with U/Pu Core, Th Blanket Inside, Modified LEU/Th LMFBR Outside. External LMFBR Spent Fuel Returned to Center for Reprocessing

Process Step	Operation	Material Description	Development Needed	Material Location	Convertibility	Radiation Hazard
1.	Feed Material	UO_2, PuC_2 and MOX powder	Some#	Glovebox$^{\Delta}$	d*	Low+
2.	Fuel Fabrication	MOX powder and pellets	Some#	Glovebox$^{\Delta}$	d	Low+
3.	Pin Fabrication	MOX pellets and pins	Some#	Glovebox$^{\Delta}$	d	Low+
4.	Bundle Assembly	MOX pins and assemblies	Moderate#	Limited Access Assembly Area	d	Low+
5.	Scrap Recycle	MOX powder and pellets, solutions of Pu and U	Little**	Glovebox	d	Low**
6.	Waste Reprocess	Solid and liquid wastes containing low-level Pu and U contamination	Little**	Glovebox	d	Low**

(#) Processes are currently used in industry with low (~12%) ^{240}Pu material. Hot prototype needed for automated operation and use of ^{233}U or high (>20%) ^{240}Pu.

(*) UO_2 is nonfissionable.

(+) Negligible for normal or depleted UO_2. May be higher for $^{233}UO_2$.

(Δ) Containment with mechanized/automated operation.

(**) Fuel containing ^{233}U may need an additional separations step.

CASE 29
3.2.3.1 LMFBR U/Pu RECYCLE (REF CASE)

FUEL FABRICATION PLANT

LEVEL-1 FUNCTIONAL FLOW DIAGRAM

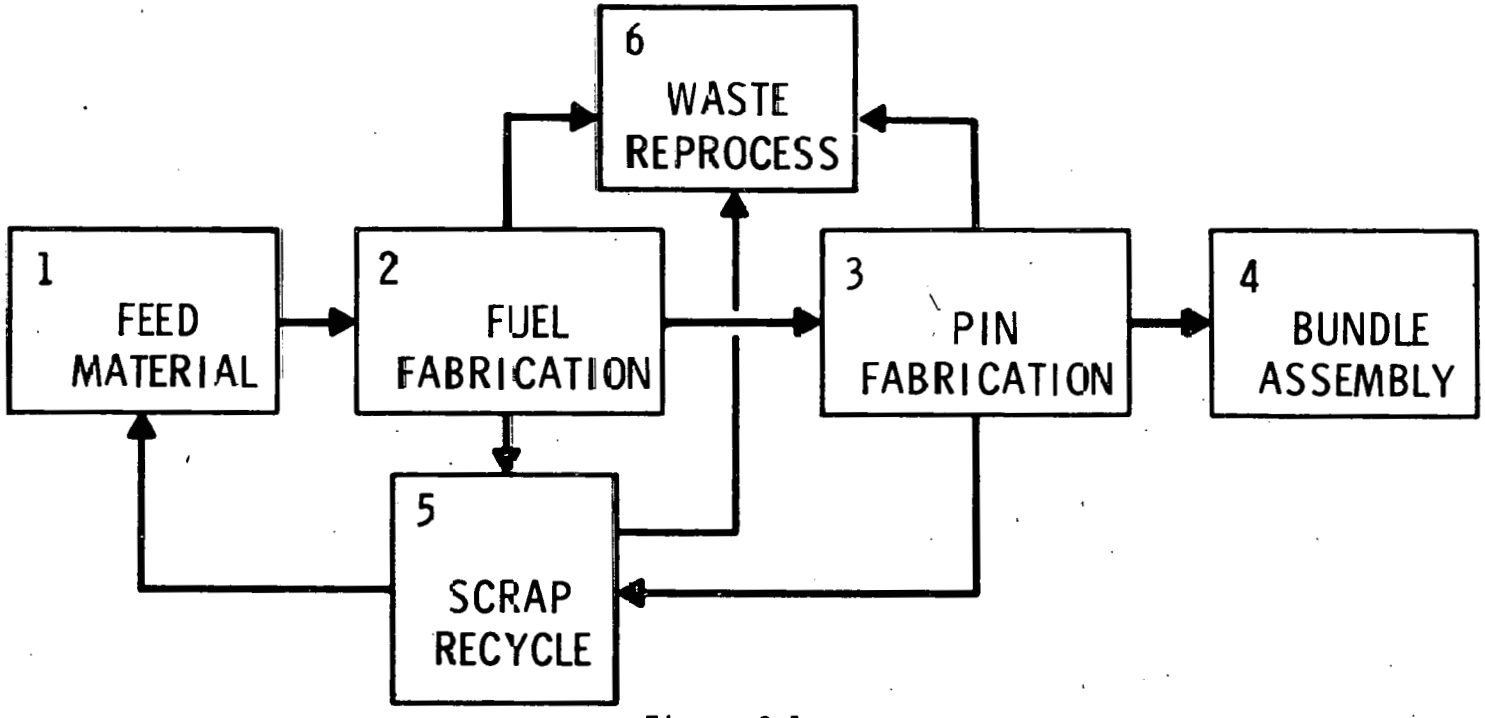

Figure 3.1

CASE 29

3.2.3.1 LMFBR U/Pu (REF. CASE)

FUEL FABRICATION PLANT

LEVEL-2 FUNCTIONAL FLOW DIAGRAM

1. FEED MATERIAL

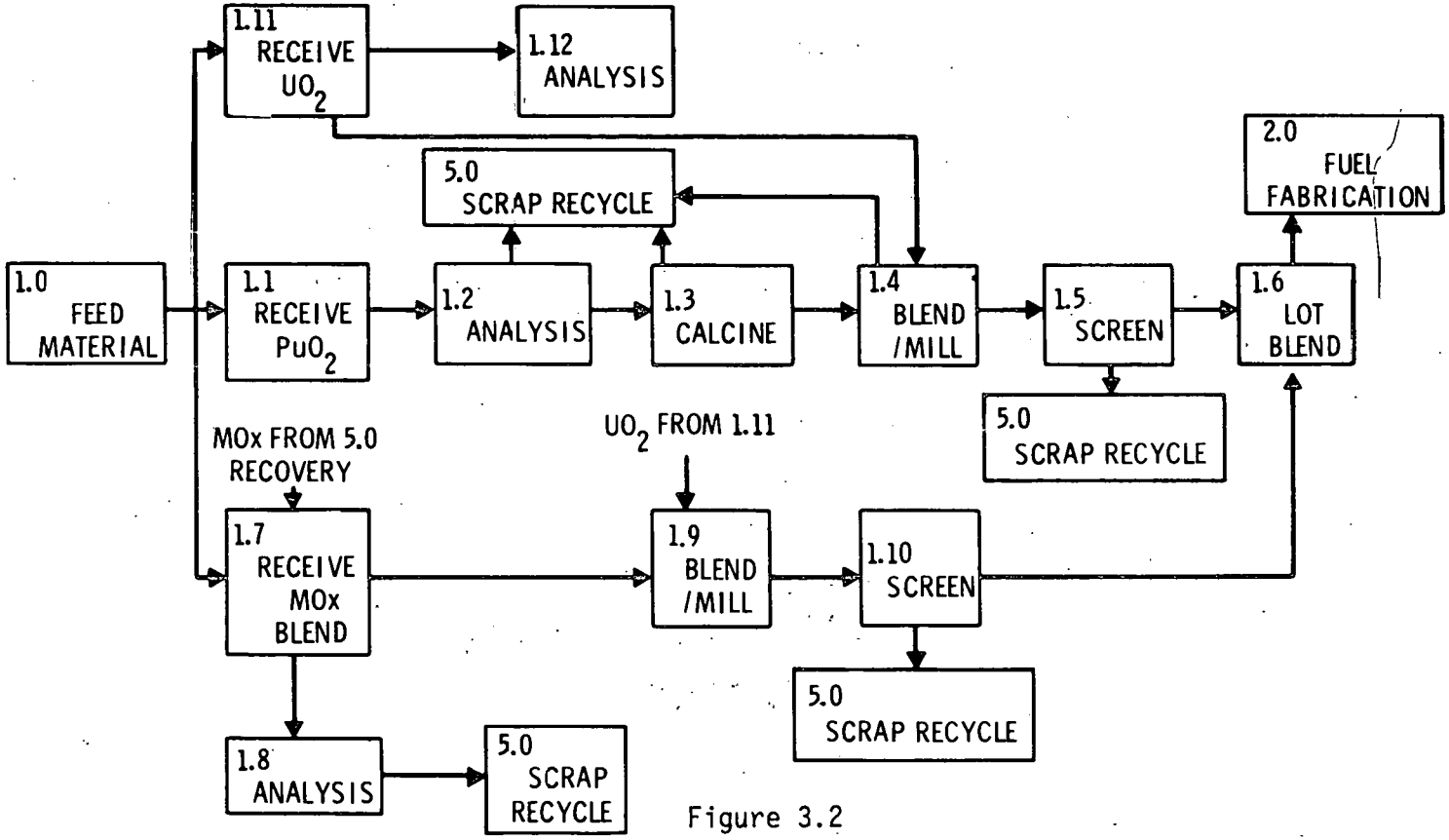

Figure 3.2

CASE 29

3.2.3.1 LMFBR U/Pu (REF CASE)

FUEL FABRICATION PLANT

LEVEL-2 FUNCTIONAL FLOW DIAGRAM

2. FUEL FABRICATION

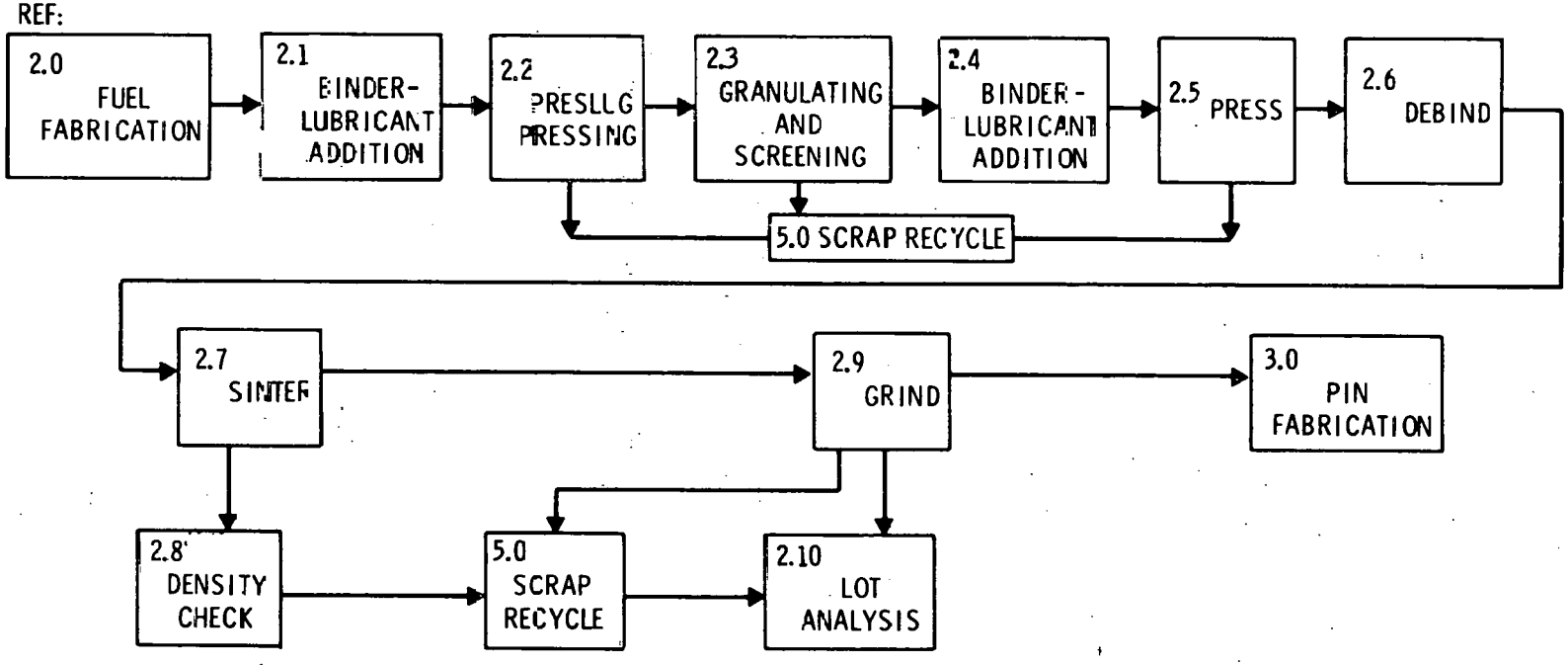

Figure 3.3

CASE 29

3.2.3.1 LMFBR U/Pu (REF. CASE)

FUEL FABRICATION PLANT

LEVEL 2 FUNCTIONAL FLOW DIAGRAM

3. PIN FABRICATION

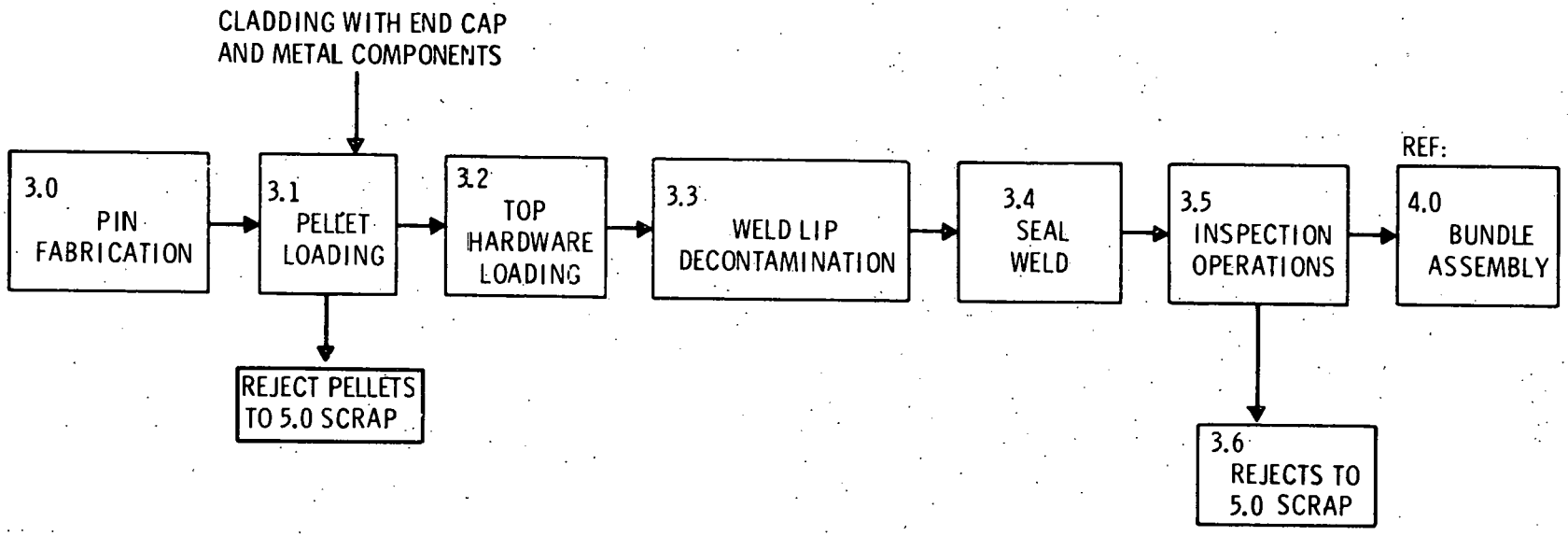

Figure 3.4

CASE 29

3.2.3.1 LMFBR U/Pu (REF. CASE)

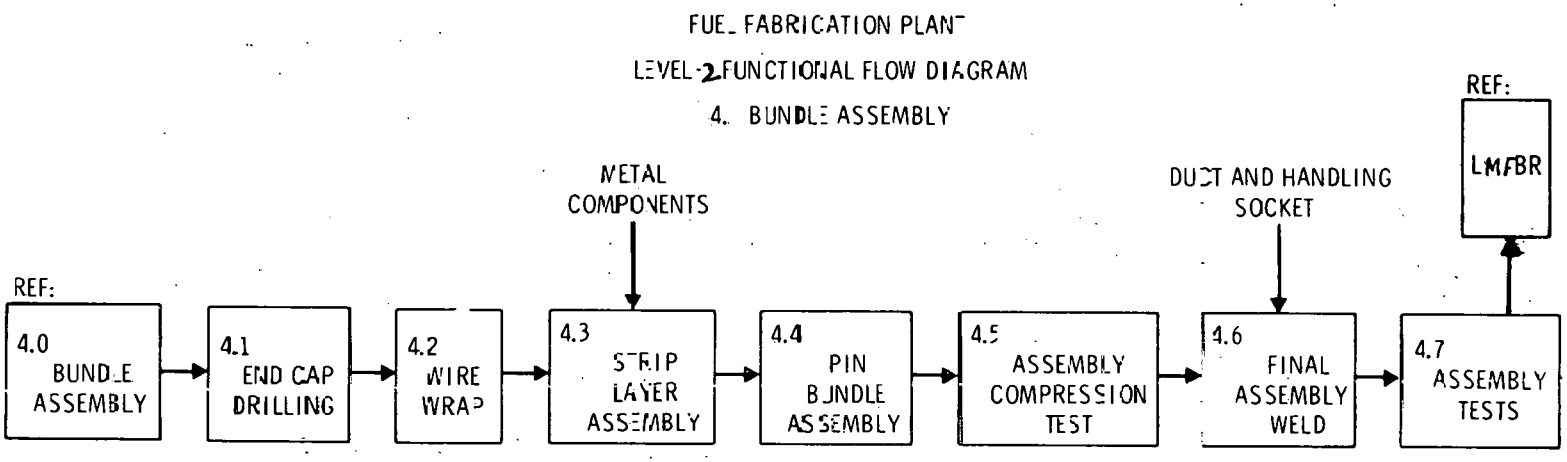

Figure 3.5

CASE 29
3.2.3.1 LMFBR U/Pu RECYCLE (REF CASE)

FUEL FABRICATION PLANT

LEVEL 2 FUNCTIONAL FLOW DIAGRAM

5. SCRAP RECYCLE

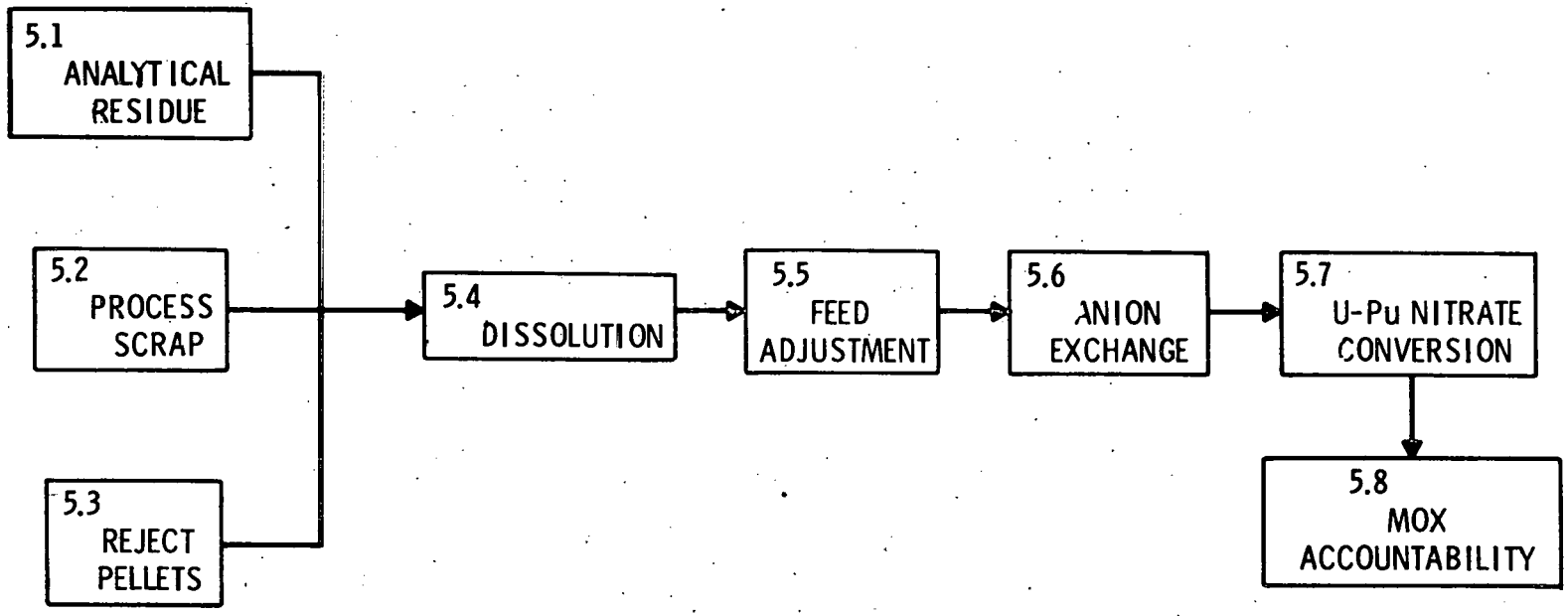

Figure 3.6

CASE 29
3.2.3.1 LMFBR U/Pu RECYCLE (REF CASE) AND RECYCLED OXIDE FUEL

FUEL REFABRICATION

LEVEL-2 FUNCTIONAL FLOW DIAGRAM

6. WASTE REPROCESS

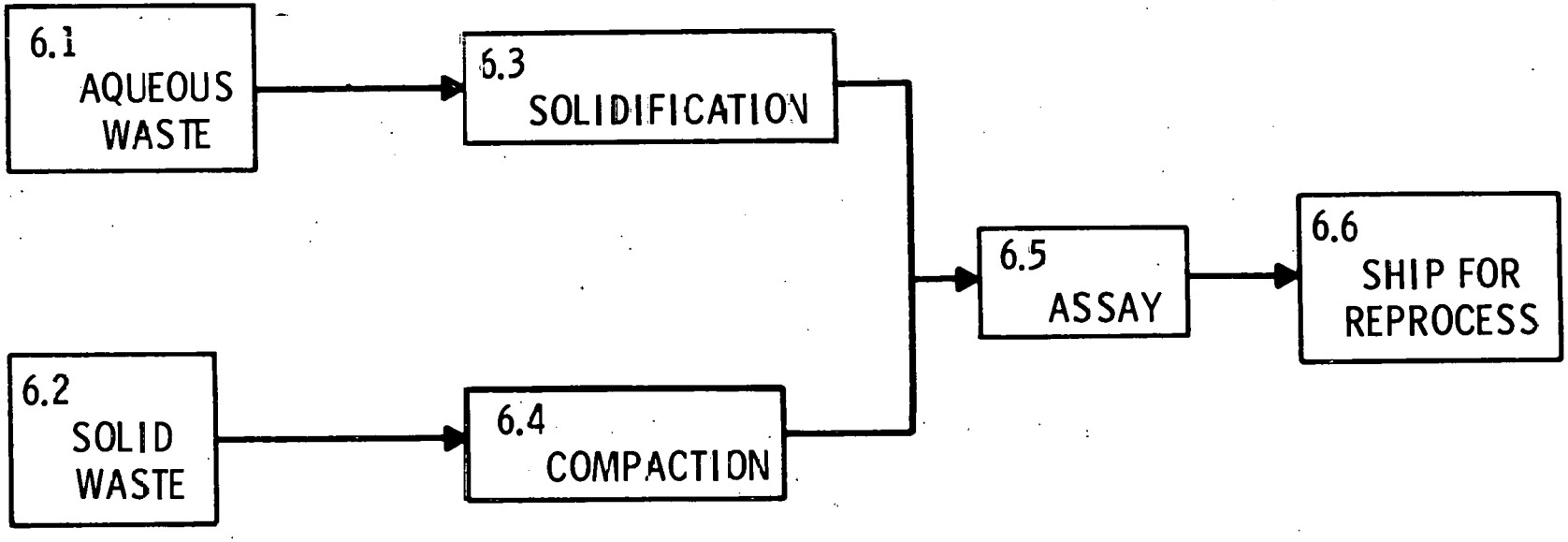

Figure 3.7

CASE 30

3.2.3.2 LMFBR/LWR U/Pu COPROCESSED, CONTAMINATED, AND RECYCLED OXIDE FUEL

FUEL FABRICATION PLANT

LEVEL-1 FUNCTIONAL FLOW DIAGRAM

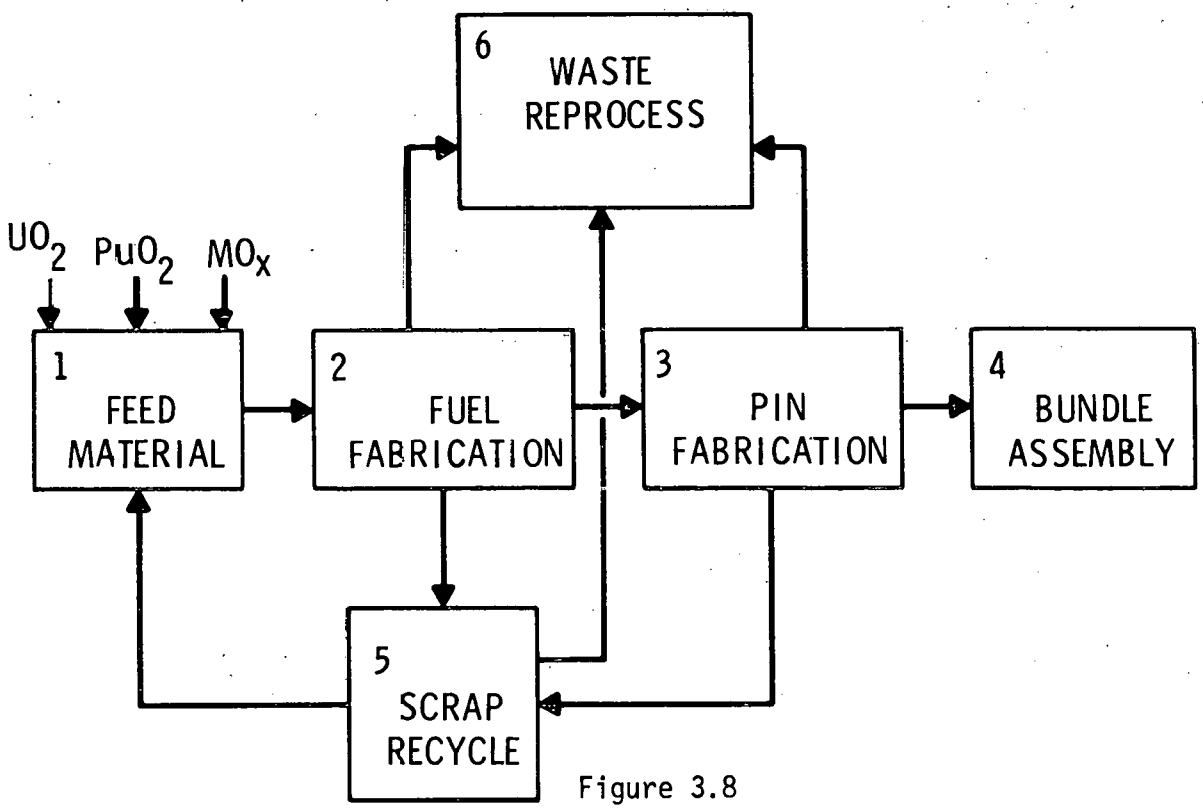

Figure 3.8

CASE 30

3.2.3.2 LMFBR/LWR U/Pu COPROCESSED, CONTAMINATED, AND RECYCLED OXIDE FUEL

FUEL FABRICATION PLANT

LEVEL-2 FUNCTIONAL FLOW DIAGRAM

1. FEED MATERIAL

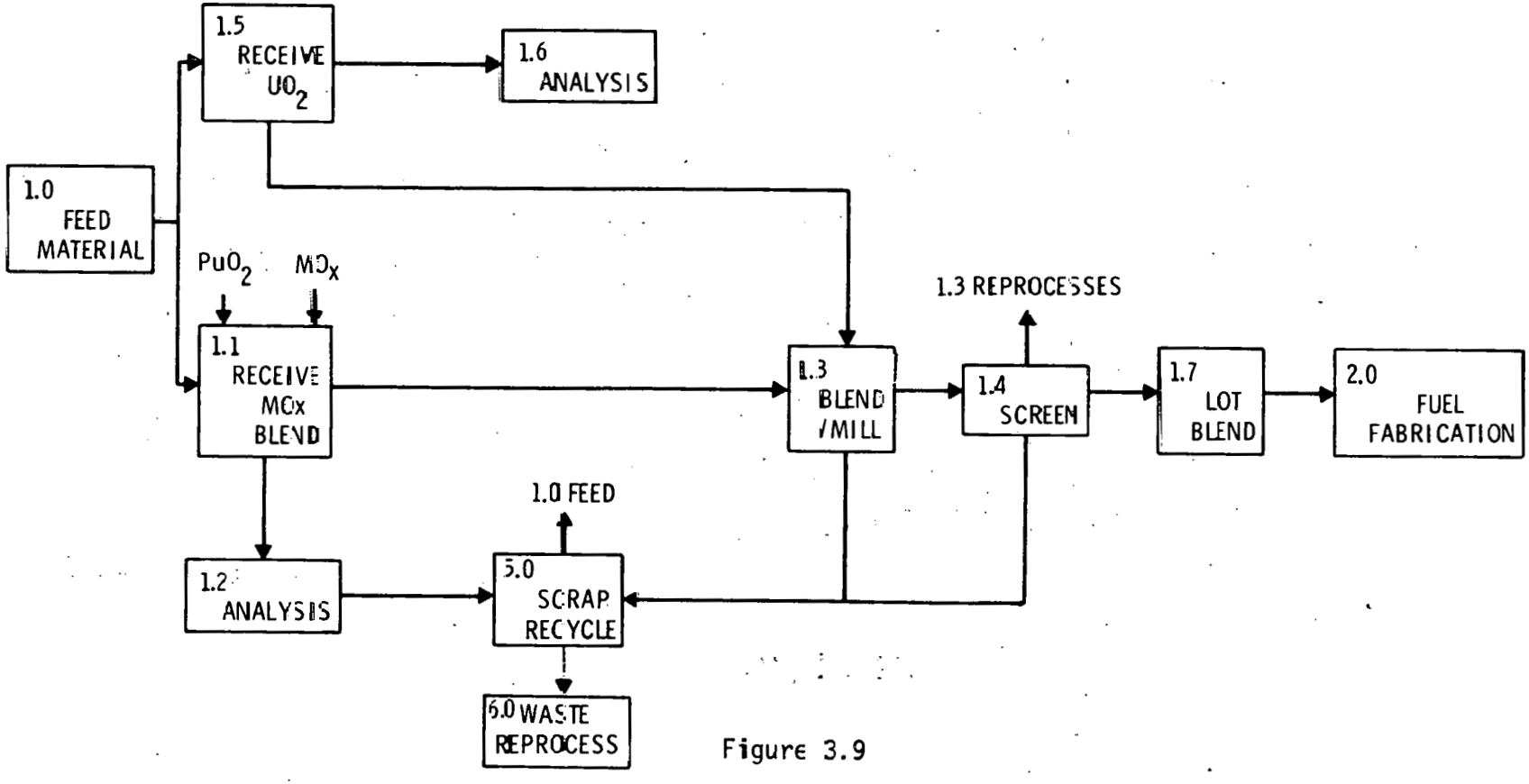

Figure 3.9

CASE 30
3.2.3.2 LMFBR/LWR U/Pu COPROCESSED, CONTAMINATED, AND RECYCLED OXIDE FUEL

FUEL FABRICATION PLANT

LEVEL-2 FUNCTIONAL FLOW DIAGRAM

2. FUEL FABRICATION

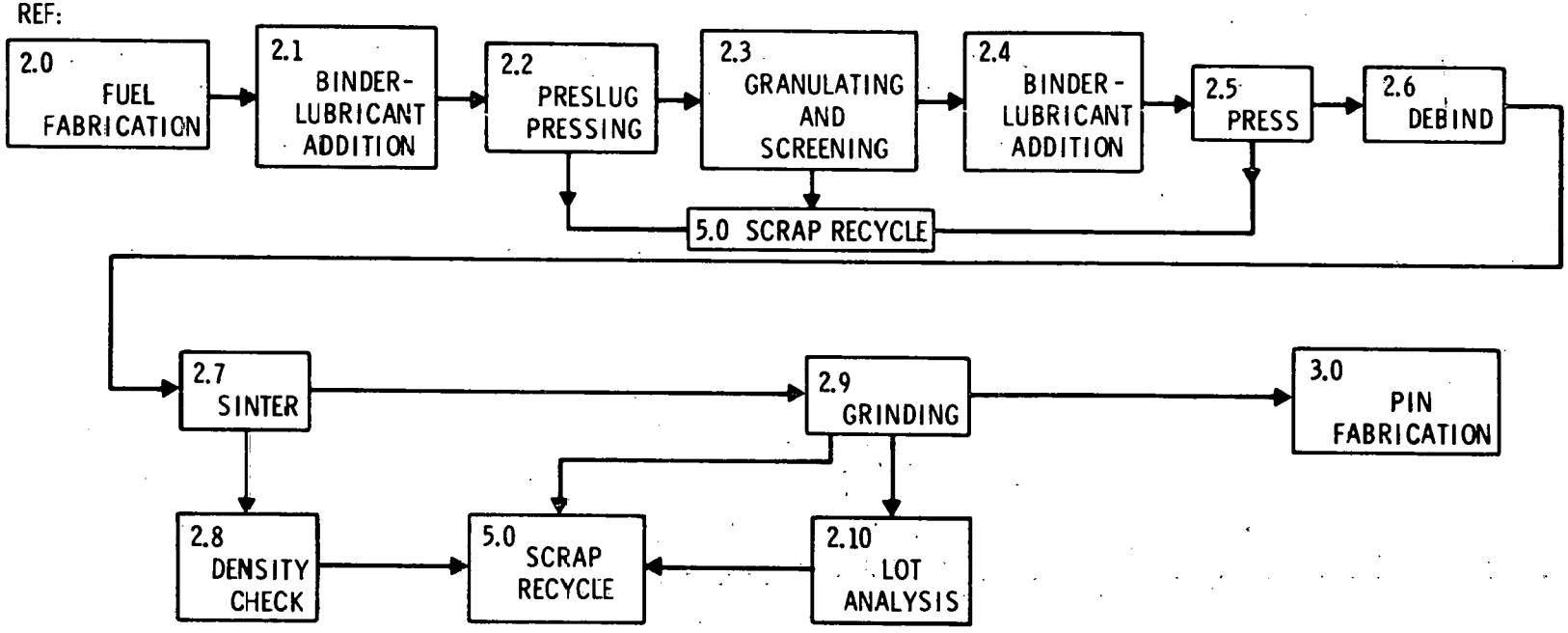

Figure 3.10

CASE 30
3.2.3.2 LMFBR/LWR U/Pu COPROCESSED, CONTAMINATED, AND RECYCLED OXIDE FUEL

FUEL FABRICATION PLANT

LEVEL-2 FUNCTIONAL FLOW DIAGRAM

3. PIN FABRICATION

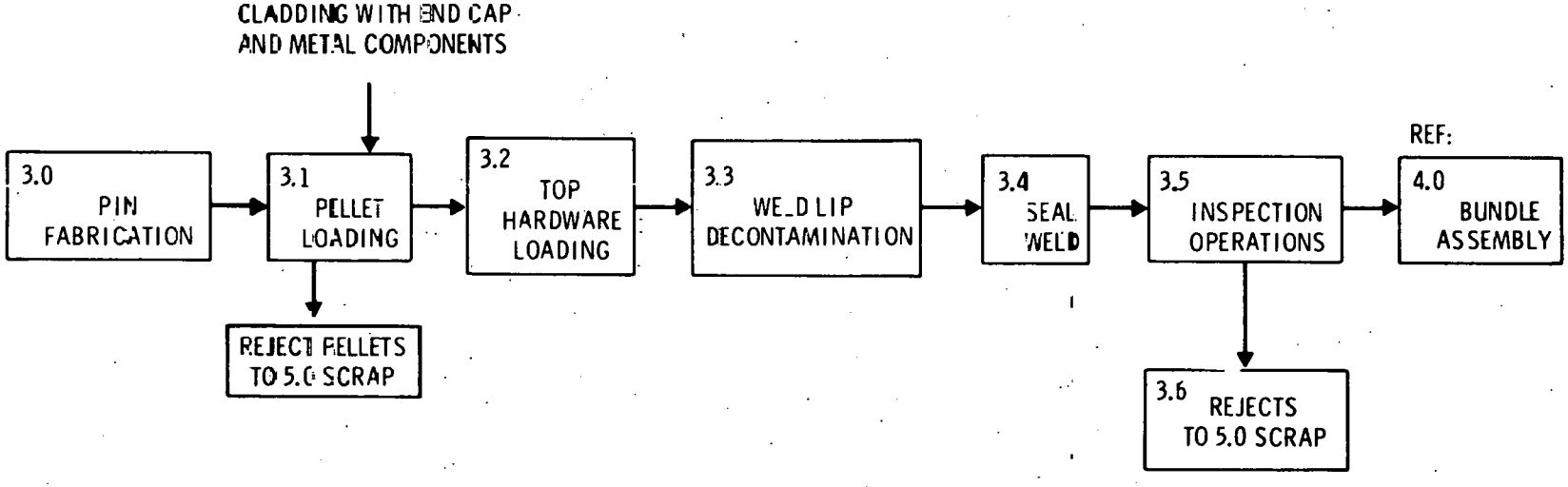

Figure 3.11

CASE 30
3.2.3.2 LMFBR/LWR U/Pu COPROCESSED, CONTAMINATED, AND RECYCLED OXIDE FUEL

FUEL REFABRICATION

LEVEL-2 FUNCTIONAL FLOW DIAGRAM

4. BUNDLE ASSEMBLY

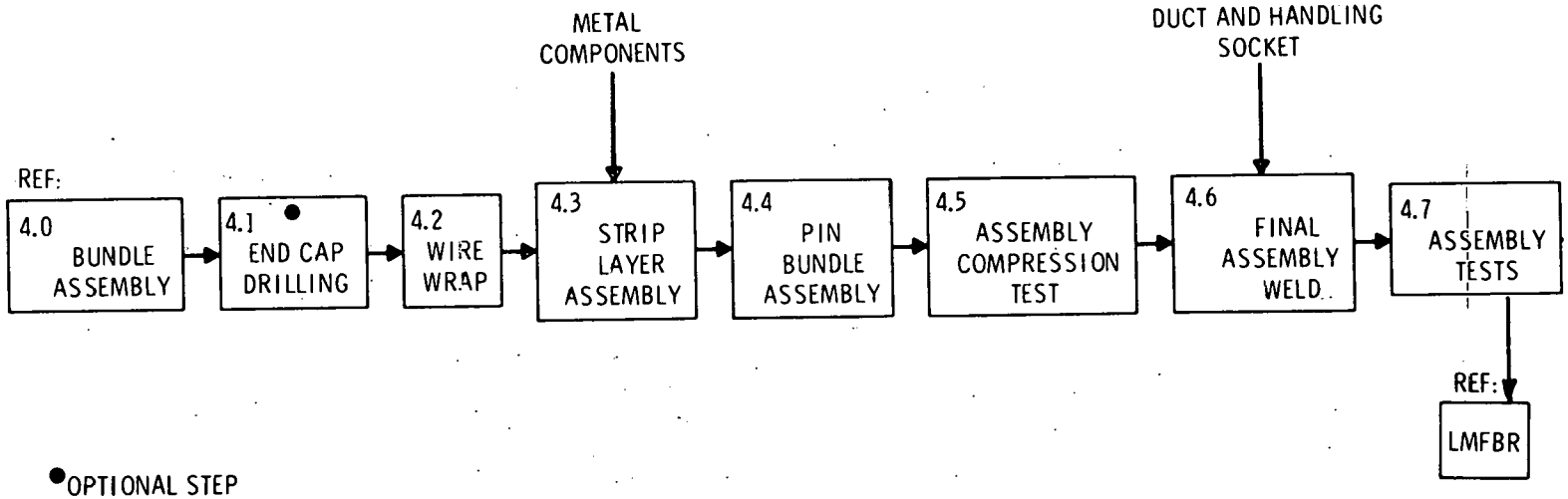

Figure 3.12

CASE 30
3.2.3.2 LMFBR/LWR U/Pu COPROCESSED, CONTAMINATED, AND RECYCLED OXIDE FUEL

FUEL FABRICATION PLANT

LEVEL-2 FUNCTIONAL FLOW DIAGRAM

5. SCRAP RECYCLE

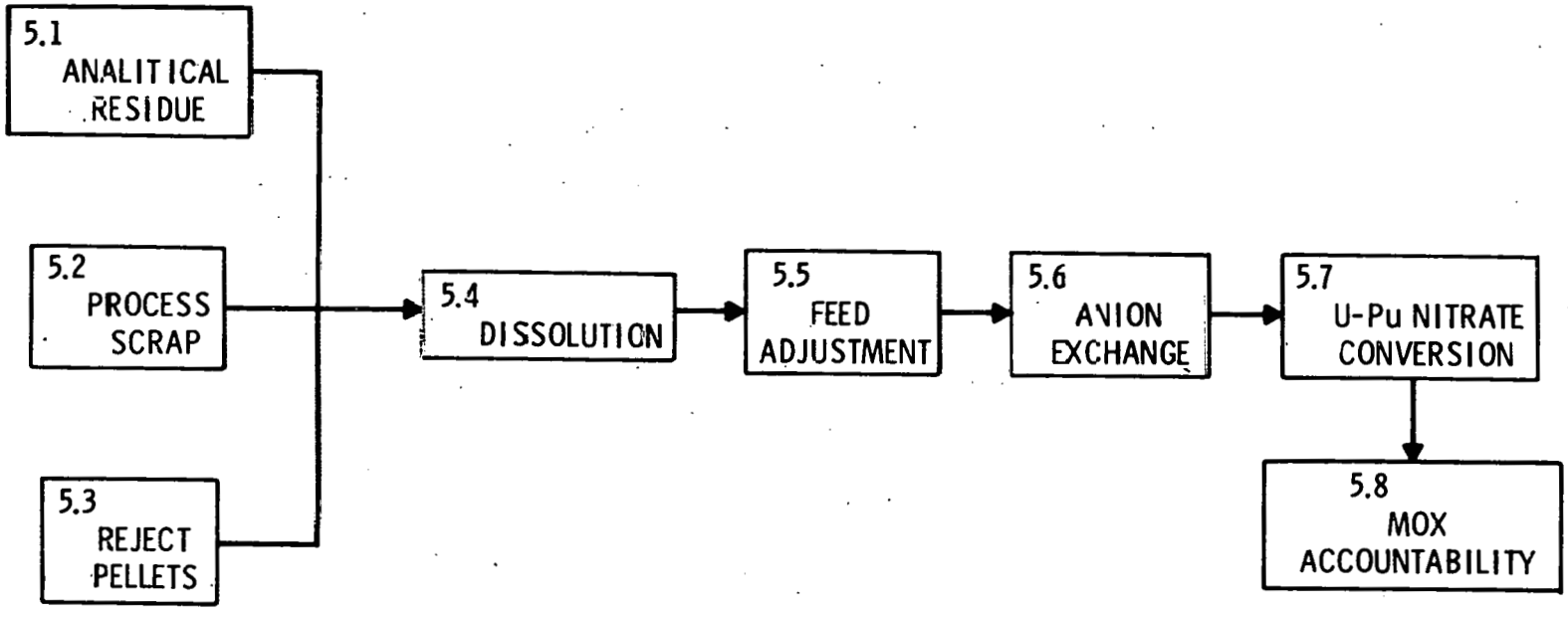

Figure 3.13

CASE 30
3.2.3.2 LMFBR/LWR U/Pu COPROCESSED, CONTAMINATED, AND RECYCLED OXIDE FUEL

FUEL REFABRICATION

LEVEL-2 FUNCTIONAL FLOW DIAGRAM

6. WASTE REPROCESS

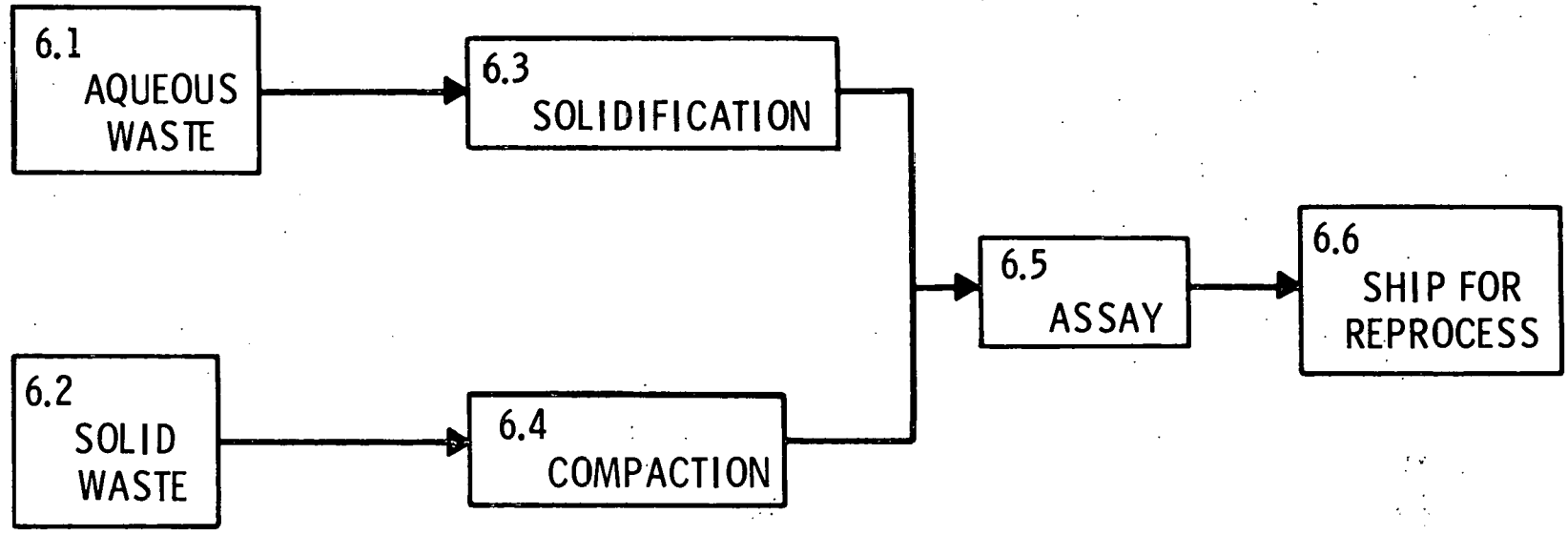

Figure 3.14

CASE 31
3.2.3.2 LMFBR /LWR U/Pu COPROCESSED, CONTAMINATED, AND RECYCLED CARBIDE FUEL

FUEL FABRICATION PLANT

LEVEL-1 FUNCTIONAL FLOW DIAGRAM

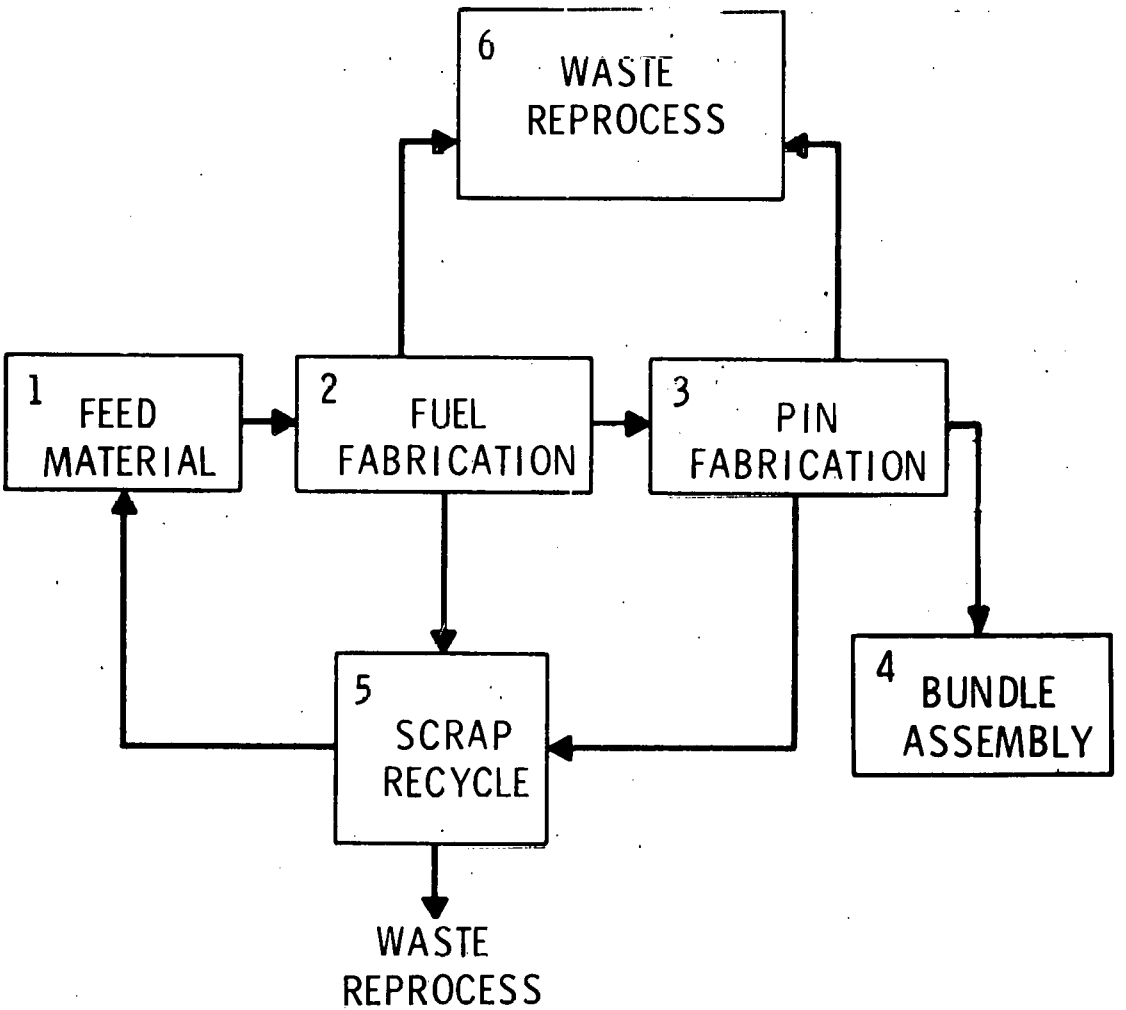

Figure 3.15

CASE 31

3.2.3.3 LMFBR/LWR U/Pu COPROCESSED, CONTAMINATED, AND RECYCLED CARBIDE FUEL

FUEL FABRICATION PLANT

LEVEL-2 FUNCTIONAL FLOW DIAGRAM

1. FEED MATERIAL

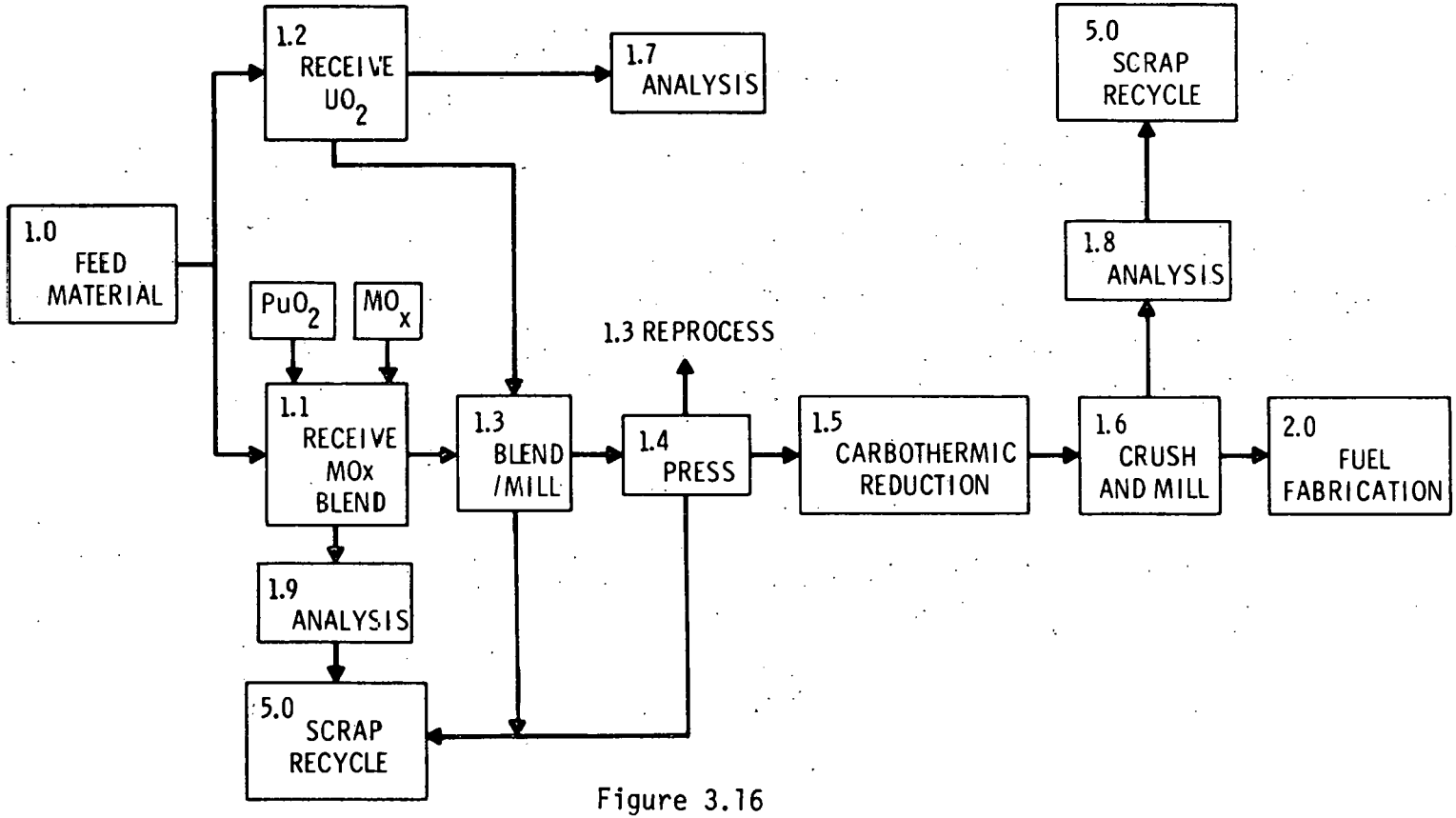

Figure 3.16

CASE 31
3.2.3.3 LMFBR/LWR U/Pu COPROCESSED, CONTAMINATED, AND RECYCLED CARBIDE FUEL

FUEL FABRICATION PLANT

LEVEL-2 FUNCTIONAL FLOW DIAGRAM

2. FUEL FABRICATION

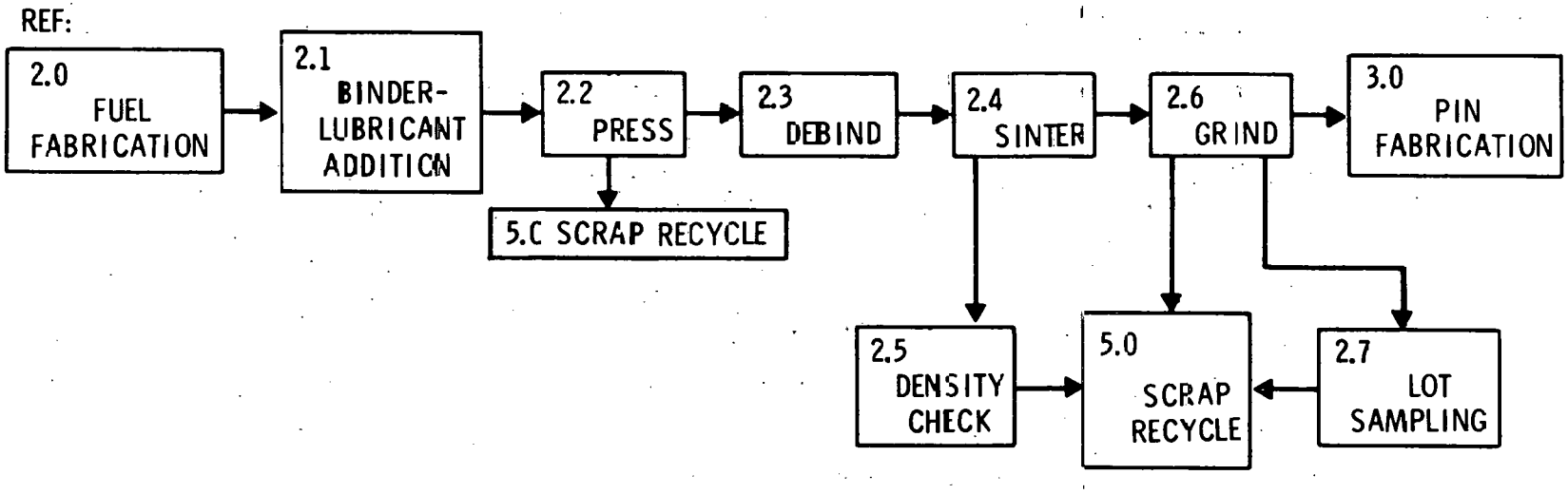

Figure 3.17

CASE 31
3.2.3.3 LMFBR/LWR U/Pu COPROCESSED, CONTAMINATED, AND RECYCLED CARBIDE FUEL

FUEL FABRICATION PLANT

LEVEL-2 FUNCTIONAL FLOW DIAGRAM

3. PIN FABRICATION

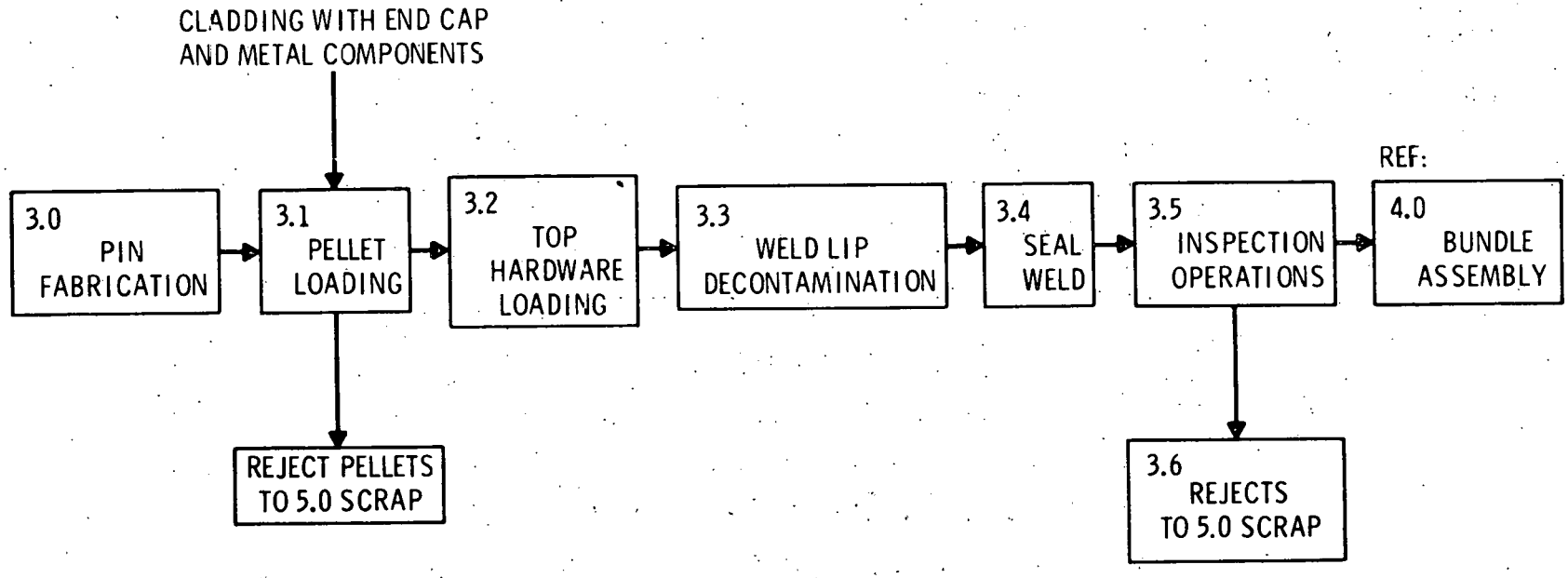

Figure 3.18

CASE 31
3.2.3.3 LMFBR/LWR U/Pu COPROCESSED, CONTAMINATED, AND RECYCLED CARBIDE FUEL

FUEL FABRICATION PLANT

LEVEL-2 FUNCTIONAL FLOW DIAGRAM

4. BUNDLE ASSEMBLY

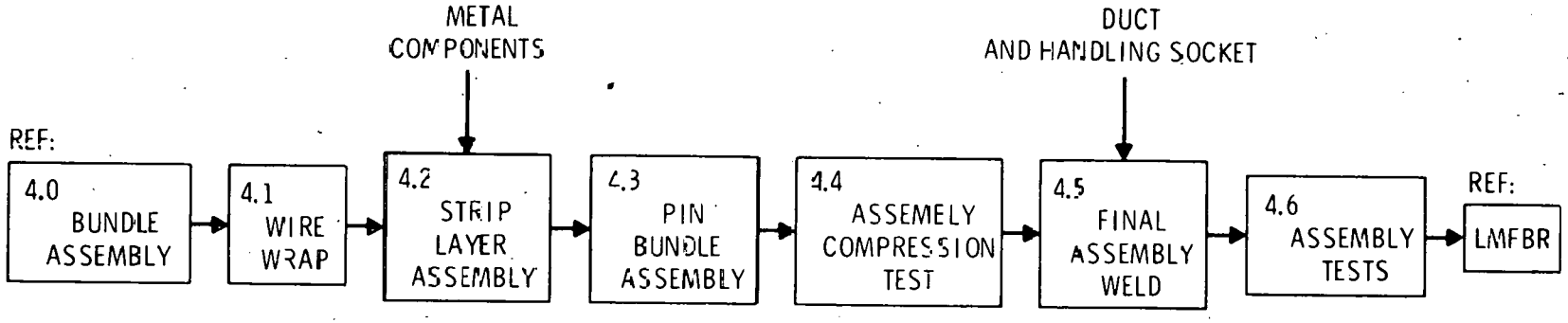

Figure 3.19

CASE 31
3.2.3.3 LMFBR/LWR U/Pu COPROCESSED, CONTAMINATED, AND RECYCLED CARBIDE FUEL

FUEL FABRICATION PLANT

LEVEL-2 FUNCTIONAL FLOW DIAGRAM

5. SCRAP RECYCLE

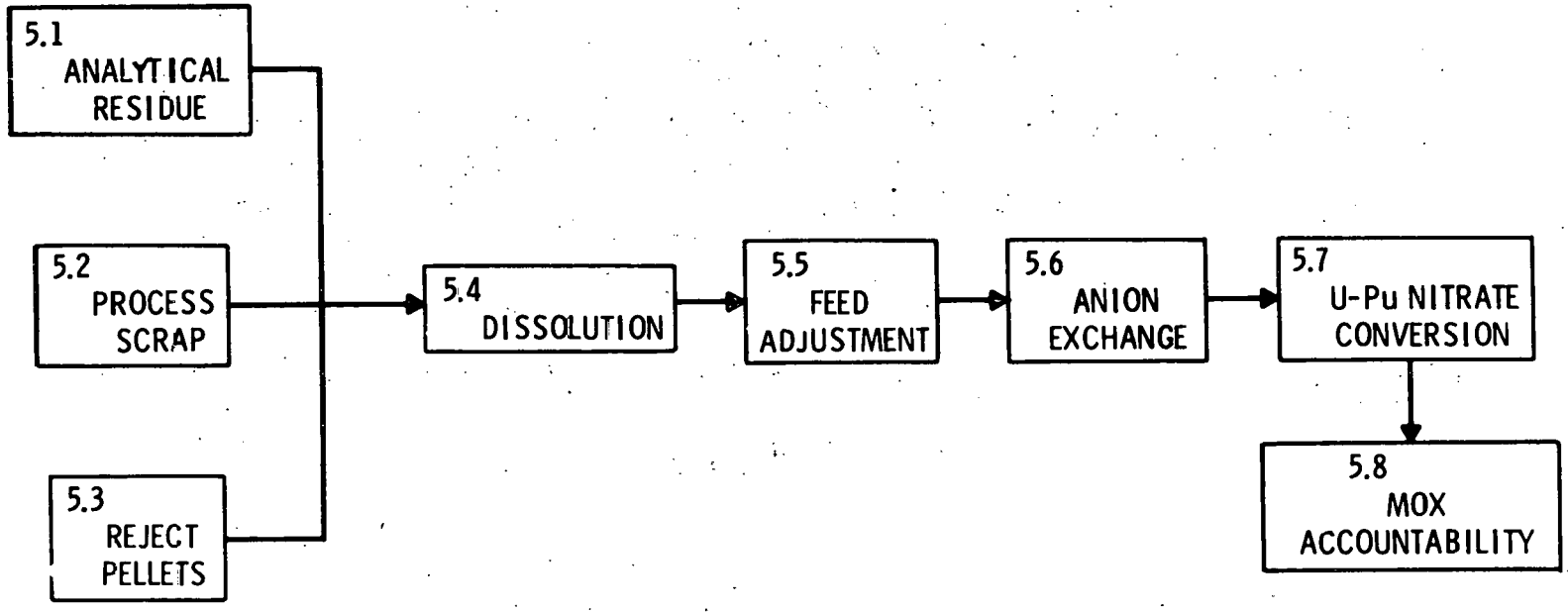

Figure 3.20

CASE 31

3.2.3.3 LMFBR/LWR U/Pu COPROCESSED, CONTAMINATED, AND RECYCLED CARBIDE FUEL

FUEL REFABRICATION

LEVEL-2 FUNCTIONAL FLOW DIAGRAM

6. WASTE REPROCESS

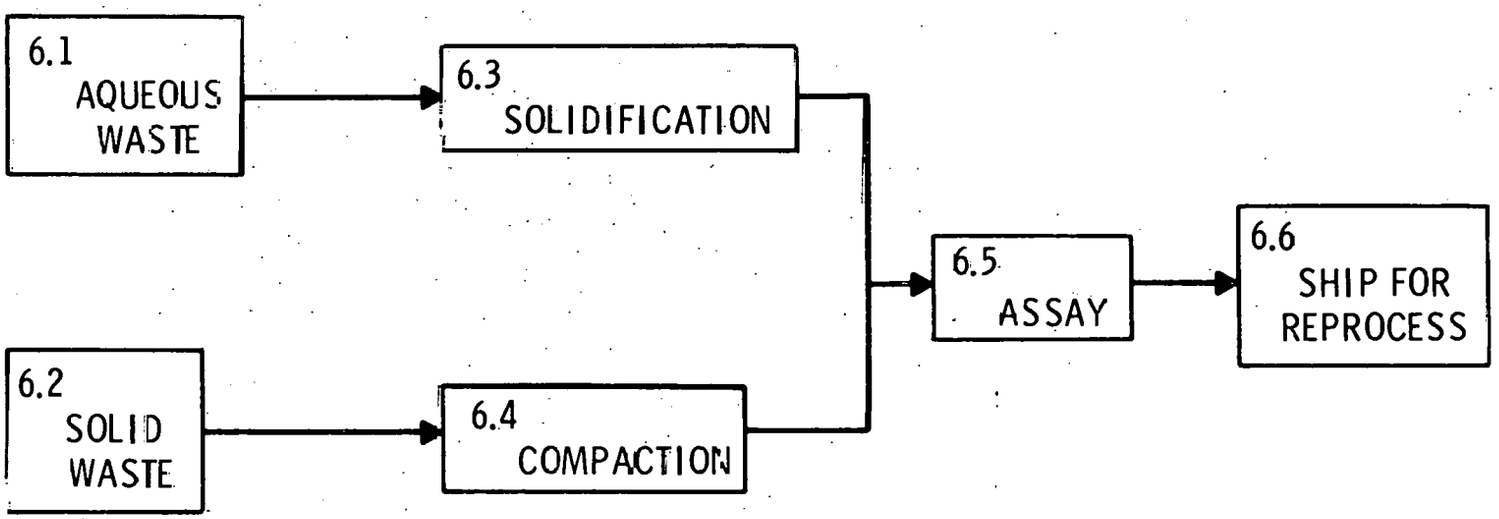

Figure 3.21

CASE 66 3.4.2.6 ENERGY CENTER CONTAINING LMFBR WITH U-Pu CORE, Th BLANKET INSIDE, MODIFIED LEU-Th LMFBR OUTSIDE. EXTERNAL LMFBR SPENT FUEL RETURNED TO CENTER FOR REPROCESSING

LEVEL-0 FUNCTIONAL FLOW DIAGRAM

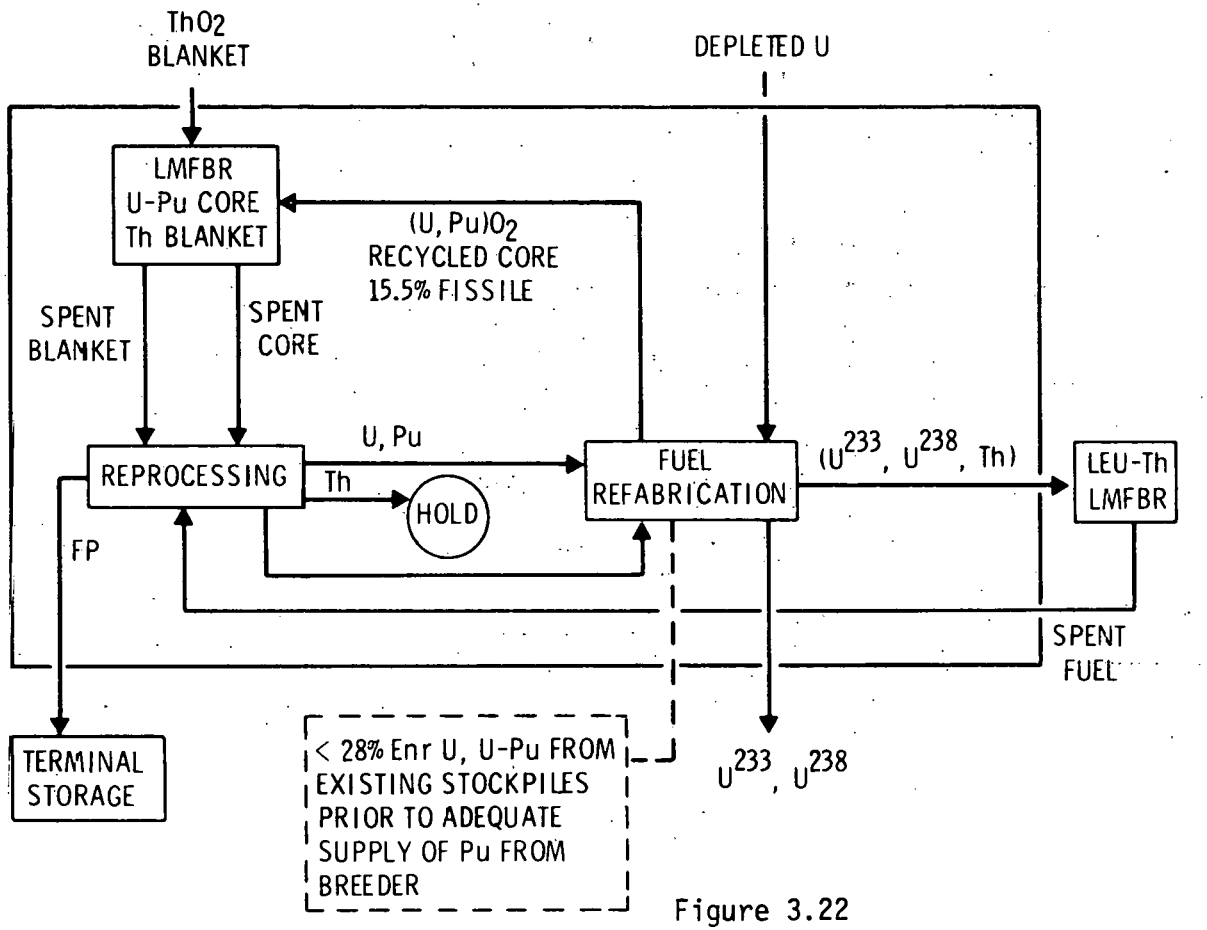

Figure 3.22

CASE 66

3.4.2.6 ENERGY CENTER CONTAINING LMFBR WITH U-Pu CORE, Th BLANKET INSIDE, MODIFIED LEU-Th LMFBR OUTSIDE. EXTERNAL LMFBR SPENT FUEL RETURNED TO CENTER FOR REPROCESSING

FUEL REFABRICATION

LEVEL-1 FUNCTIONAL FLOW DIAGRAM

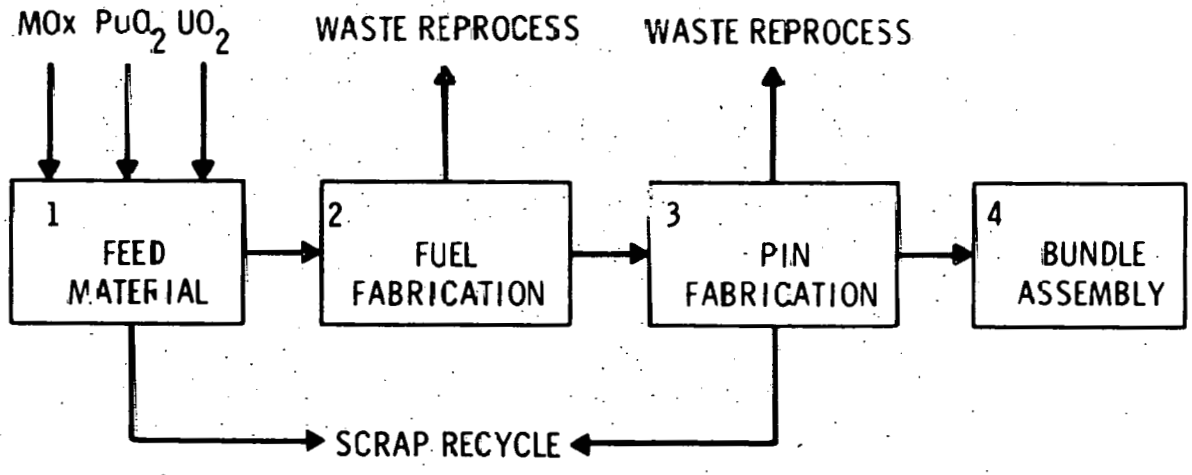

Figure 3.23

CASE 66 3.4.2.6 ENERGY CENTER CONTAINING LMFBR WITH U-Pu CORE, Th BLANKET INSIDE, MODIFIED LEU-Th LMFBR OUTSIDE. EXTERNAL LMFBR SPENT FUEL RETURNED TO CENTER FOR REPROCESSING

FUEL REFABRICATION

LEVEL-2 FUNCTIONAL FLOW DIAGRAM

1. FEED MATERIAL

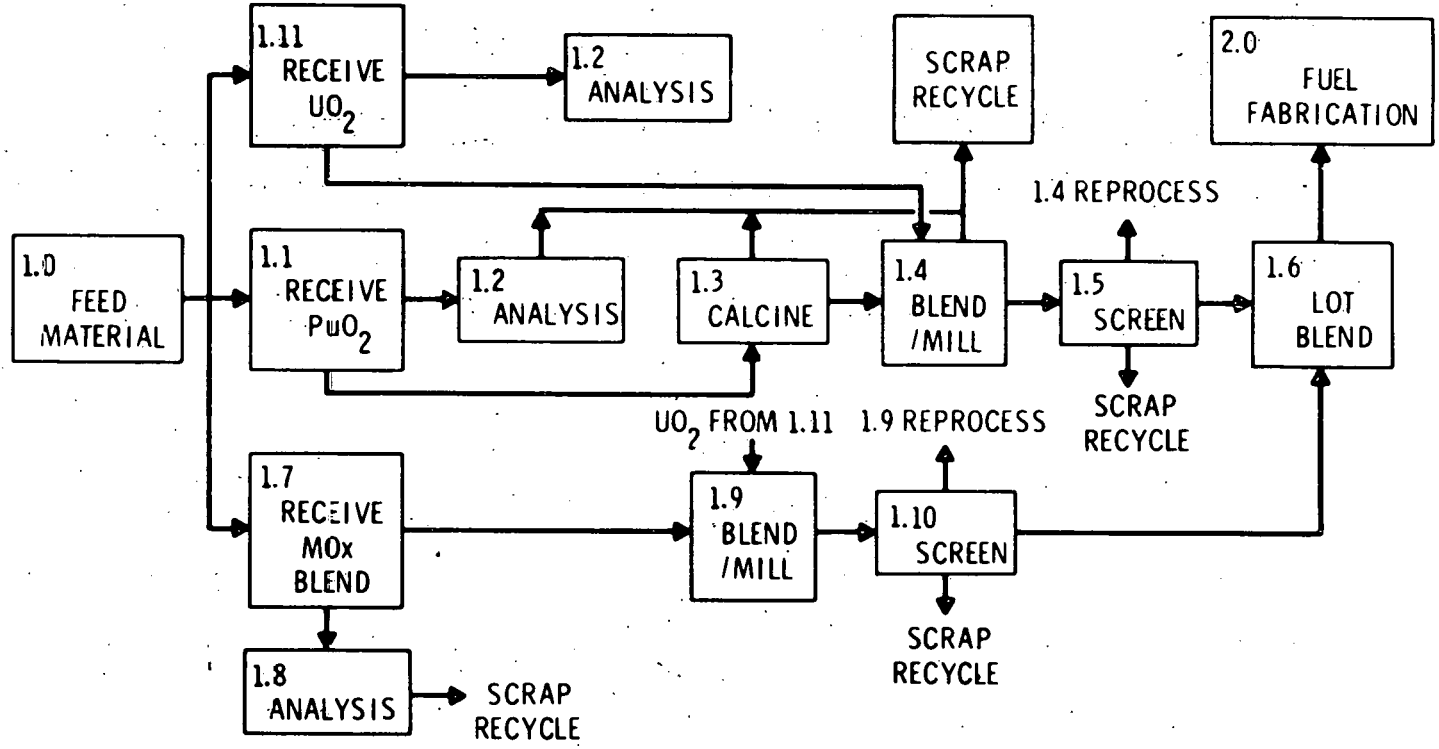

Figure 3.24

CASE 66

3.4.2.6 ENERGY CENTER CONTAINING LMFBR WITH U-Pu CORE, Th BLANKET INSIDE, MODIFIED LEU-Th LMFBR OUTSIDE. EXTERNAL LMFBR SPENT FUEL RETURNED TO CENTER FOR REPROCESSING

FUEL REFABRICATION

LEVEL-2 FUNCTIONAL FLOW DIAGRAM

2. FUEL FABRICATION

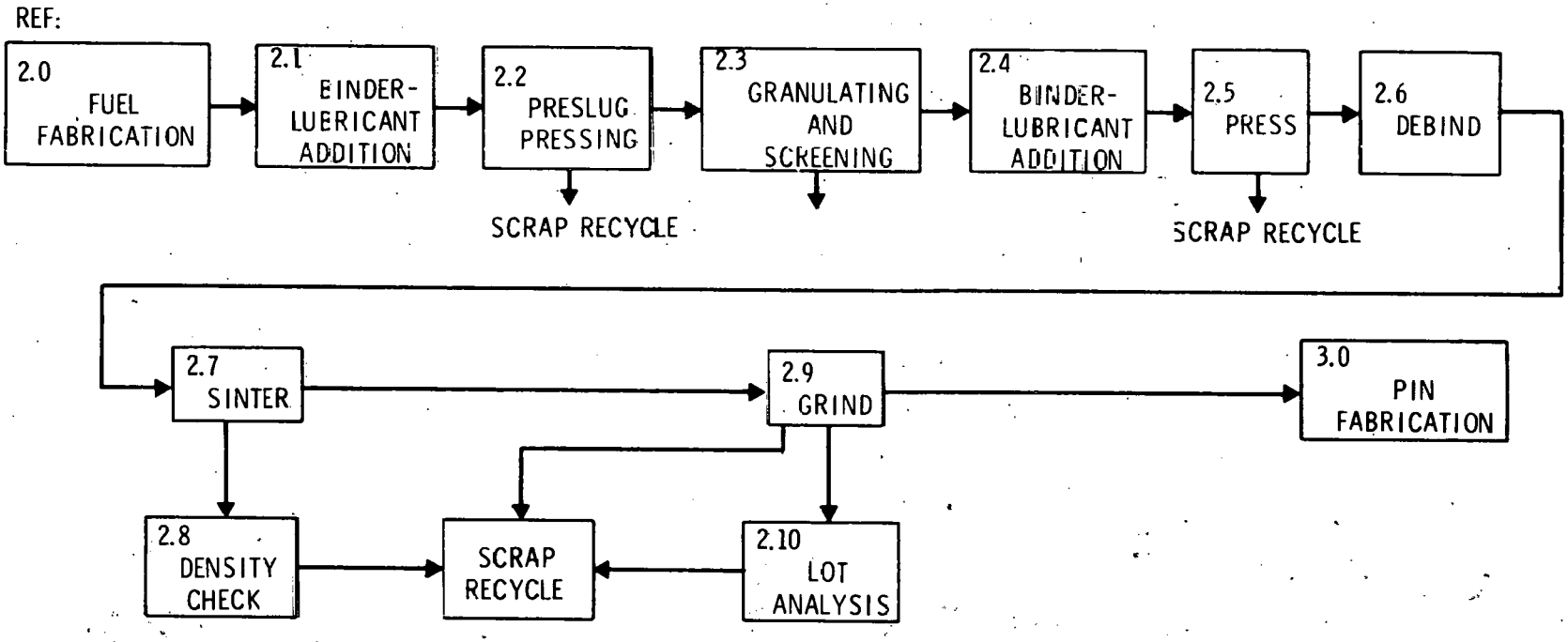

Figure 3.25

CASE 66

3.4.2.6 ENERGY CENTER CONTAINING LMFBR WITH U-Pu CORE, Th BLANKET INSIDE, MODIFIED LEU-Th LMFBR OUTSIDE. EXTERNAL LMFBR SPENT FUEL RETURNED TO CENTER FOR REPROCESSING

FUEL REFABRICATION

LEVEL-2 FUNCTIONAL FLOW DIAGRAM

3. PIN FABRICATION

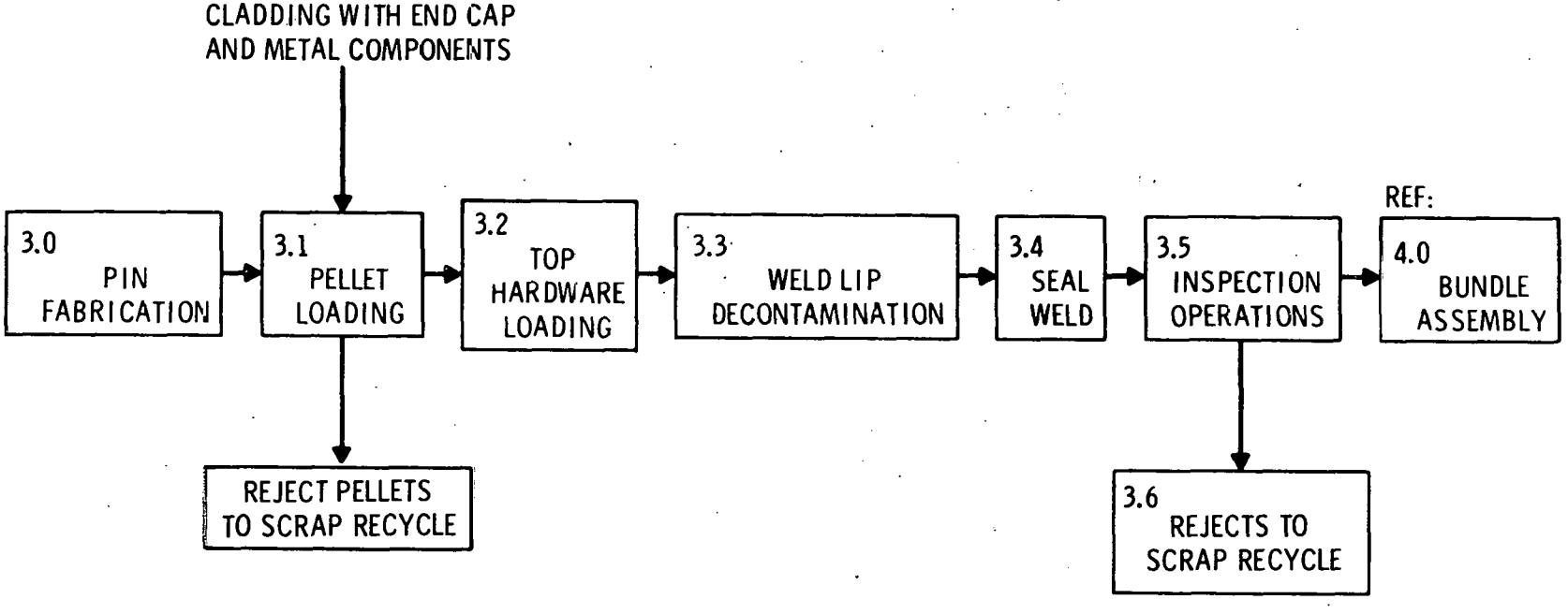

Figure 3.26

CASE 66

3.4.2.6 ENERGY CENTER CONTAINING LMFBR WITH U-Pu CORE, Th BLANKET INSIDE, MODIFIED LEU-Th LMFBR OUTSIDE. EXTERNAL LMFBR SPENT FUEL RETURNED TO CENTER FOR REPROCESSING

FUEL REFABRICATION

LEVEL 2 FUNCTIONAL FLOW DIAGRAM

4. BUNDLE ASSEMBLY

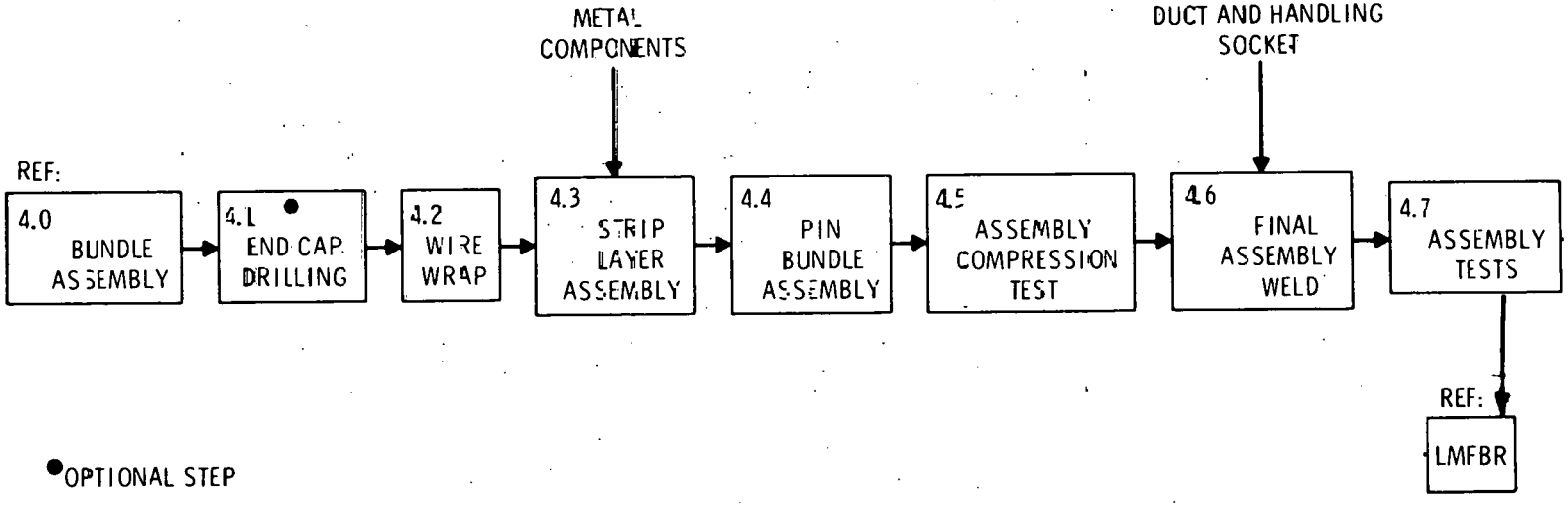

Figure 3.27

CASE 66 3.4.2.6 ENERGY CENTER CONTAINING LMFBR WITH U-Pu CORE, Th BLANKET INSIDE, MODIFIED LEU-Th BRP OUTSIDE. EXTERNAL LMFBR SPENT FUEL RETURNED TO CENTER FOR REPROCESSING

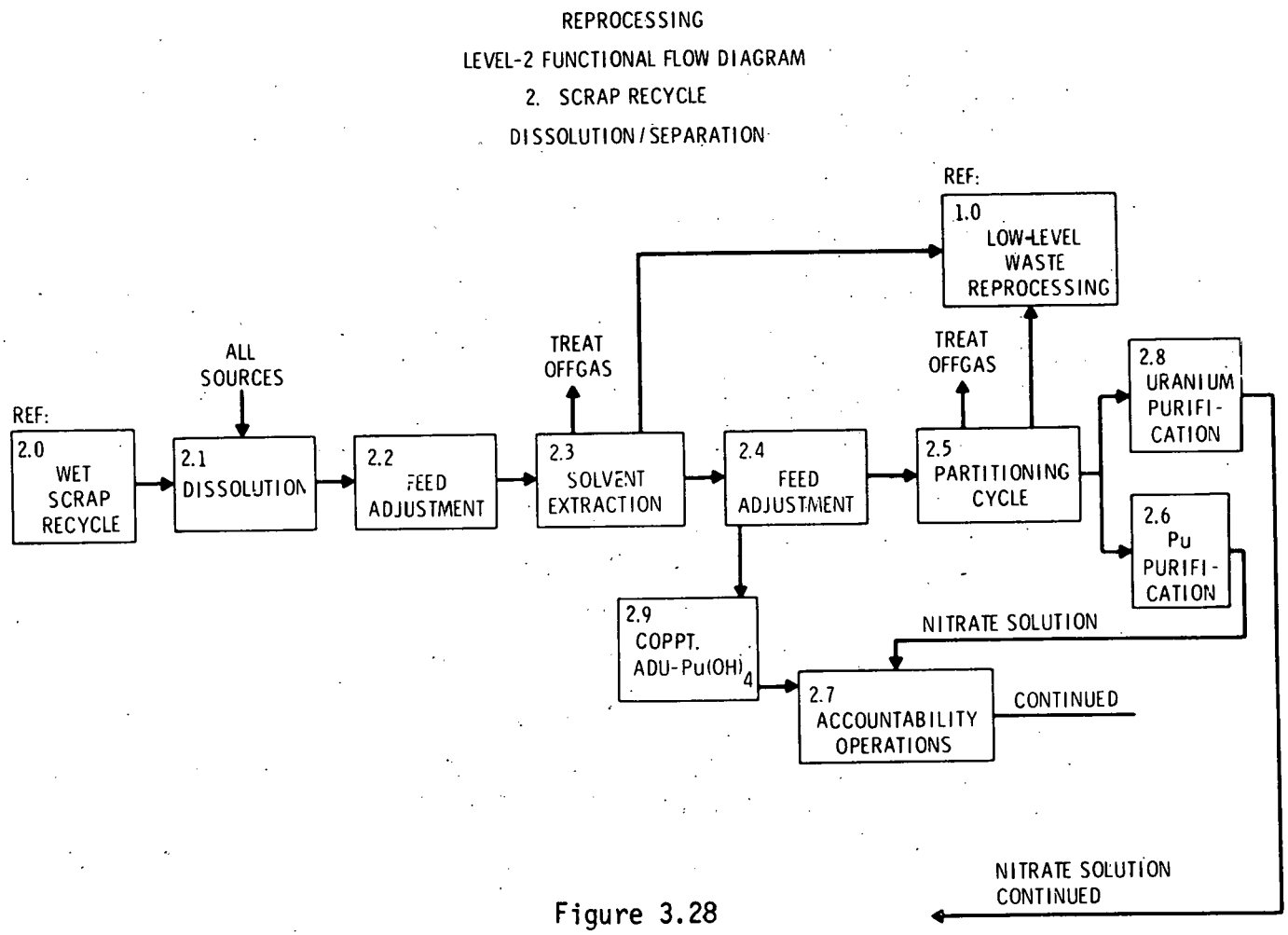

Figure 3.28

CASE 66 3.4.2.6 ENERGY CENTER CONTAINING LMFBR WITH U-Pu CORE, Th BLANKET INSIDE, MODIFIED LEU-Th LMFBR OUTSIDE. EXTERNAL LMFBR SPENT FUEL RETURNED TO CENTER FOR REPROCESSING

REPROCESSING

LEVEL-2 FUNCTIONAL FLOW DIAGRAM

2. SCRAP RECYCLE

OXIDE CONVERSION

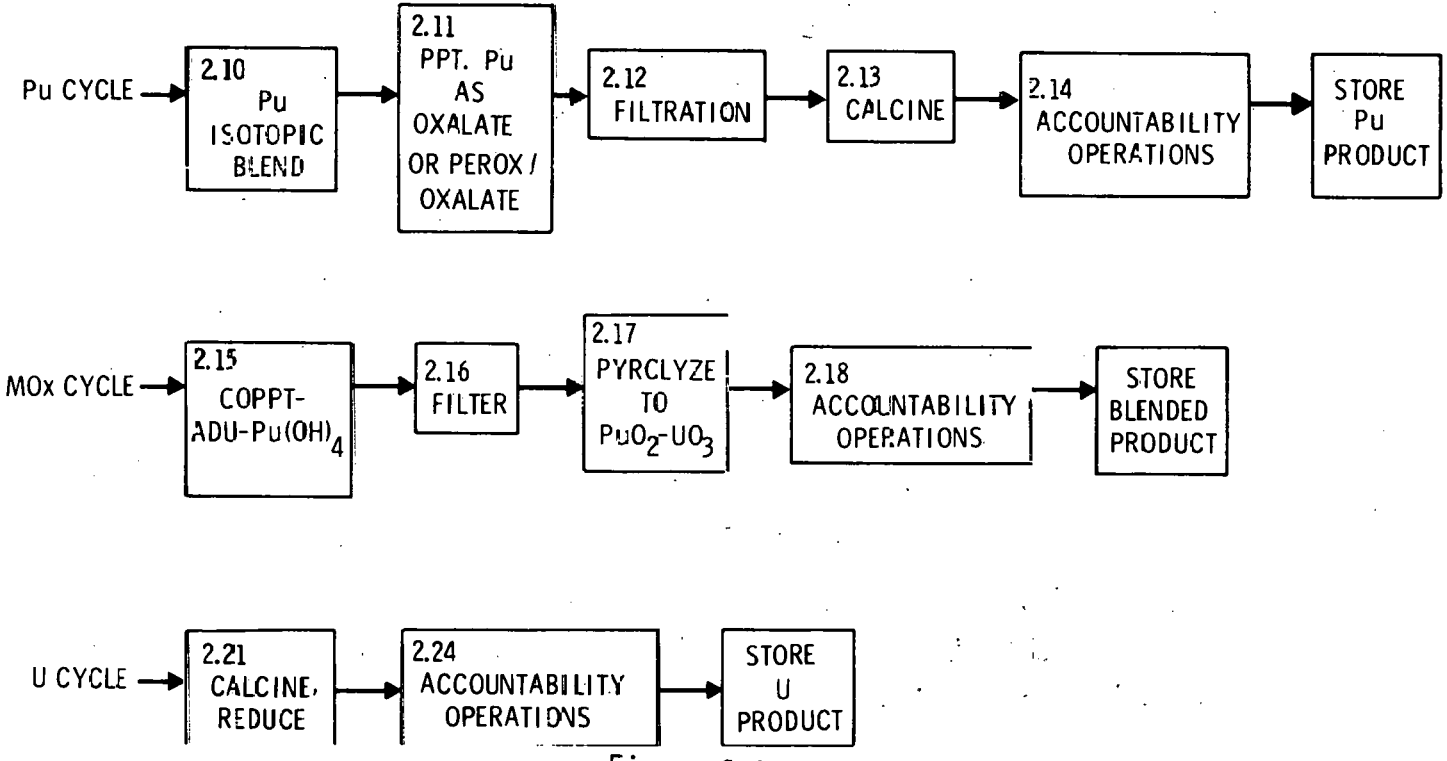

Figure 3.29

CASE 66

3.4.2.6 ENERGY CENTER CONTAINING LMFBR WITH U-Pu CORE, Th BLANKET INSIDE, MODIFIED LEU-Th LMFBR OUTSIDE. EXTERNAL LMFBR SPENT FUEL RETURNED TO CENTER FOR REPROCESSING

REPROCESSING

LEVEL 2 FUNCTIONAL FLOW DIAGRAM

3. LOW-LEVEL WASTE

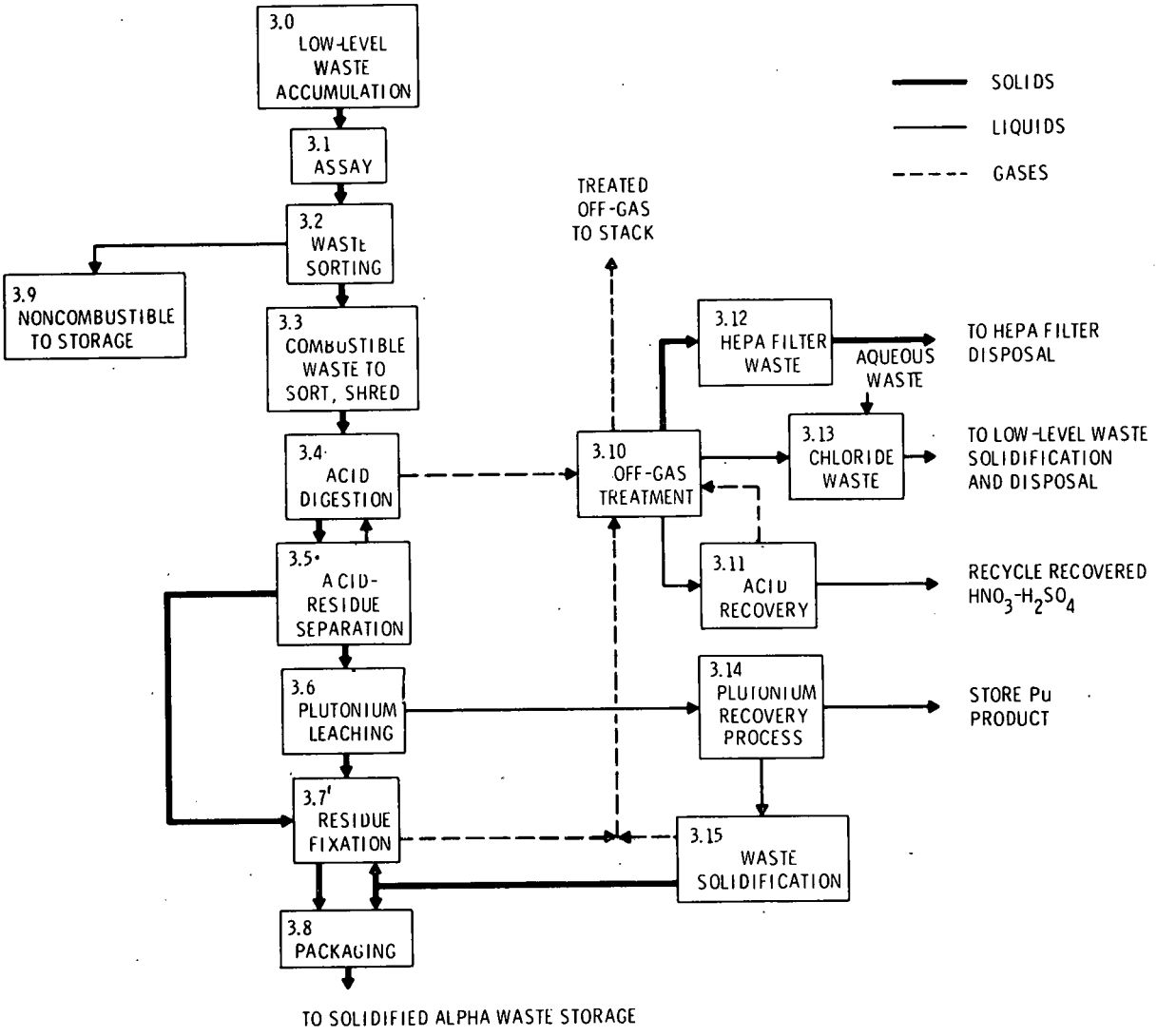

Figure 3.30

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4. FUEL CYCLE EVALUATIONS FOR THORIUM-FUELED REACTORS AND THE HTGR (Oak Ridge National Laboratory)

R. H. Rainey, W. L. Carter, D. R. Johnson, and J. A. Horak

4.1 INTRODUCTION

The analyses of the back cycles for the HTGRs using either the conventional or plutonium-uranium fuel and of the various other nuclear reactor systems with nonconventional (thorium-containing) fuels were made by staff members of the Oak Ridge National Laboratory.

At the time this report was originally prepared, the proliferation risk assessment techniques under development by Science Applications, Inc. (SAI)[1] and the descriptive matter on the various fuel types proposed in the EPRI report[2] were not available as a basis for the evaluation. This report therefore represents the authors' understanding of the various reactor systems based on the material distributed at the Atlanta meeting.[3] An example of the differences between the fuel handling systems evaluated in this report and the fuel handling in ref. 1 is as follows: in this report, except where specified, recycled fuel was assumed to return to the same reactor; that is, if the fuels to the reactor were a driver fuel of $^{235}UO_2$-$^{238}UO_2$ and a fertile fuel of ThO_2 and the ^{233}U was to be recycled to the UO_2, then the new driver fuel would contain a mixture of ^{233}U, ^{235}U and ^{238}U. In the EPRI report the new charge would contain two separate driver fuels, ^{235}U in ^{238}U and ^{233}U in ^{238}U, which may be loaded in different reactors. This difference influences the proliferation evaluation, since credit is taken for the presence of the ^{232}U associated with the ^{233}U to make this fuel less attractive, either for a weapon directly or as a feed to an isotope enrichment plant. For this reason if highly enriched ^{235}U contained ^{233}U, it was assumed to be much less attractive for proliferation than if the ^{233}U were not present.

A report[4] by H. C. Carney et al., *HTGR Fuel Cycle — Definitions for Nonproliferation Assessment*, General Atomic Company (Draft), was prepared at approximately the same time as this report. There was not enough time to make the two reports consistent for this publication, but the GA report will be very valuable for the next level of study.

4.2 EVALUATION PROCEDURE

Figures 4.1 through 4.5 present the choice tree diagrams for the back cycle processing schemes of the various reactor systems being evaluated. A brief description is given of each type of reactor fuel recycle scheme and the composition of the fuel that would be the feed to the recovery plant. These descriptions assume that "homogeneous fuels" indicates that the reactor contains only one type fuel, which contains all the designated fissile and fertile materials. "Heterogeneous fuels" indicates that two or more types of fuel are in the reactor and that these are separable before entering the solvent extraction system. When needed, there is an indication that the fuel contains highly enriched uranium (HEU), low-enriched uranium (LEU), or either. The horizontal blocks indicate the reprocessing scheme that would be required for the head-end treatment, reprocessing, and refabrication of the fuel. An attempt was made to include all the choices that could be used to recycle the fuel from each reactor back to the same reactor. The form of the product(s) from each reprocessing and refabrication scheme is included.

The Level 1 Functional Flow Diagrams (FFDs) for the dissolution of metal-clad reactor fuel are given in Fig. 4.6. For Zircaloy-clad UO_2-PuO_2 fuel or stainless-steel-clad UO_2-PuO_2 or UO_2-ThO_2 a size reduction followed by a dissolution of the exposed fuel (chop-leach) would be used. For Zircaloy-clad UO_2-ThO_2 some as yet undecided method for removing the cladding would be required before the fuel is dissolved. The head-end processes for water-cooled reactors and for liquid-metal-cooled reactors are evaluated in Table 4.1. The FFD for HTGR fuel is given in Fig. 4.7 and evaluated in Table 4.2. For these FFDs, and all that follow, the processing steps being evaluated are enclosed in solid lines. The blocks enclosed by dashes are used to indicate related but not evaluated operations.

The FFDs for the various choices for reprocessing the reactor fuels are given in Figs. 4.8 through 4.21 and evaluated in Tables 4.3 through 4.25. These FFDs and tables include the modifications of the Purex and Thorex solvent extraction systems to accommodate the combinations of ^{235}U, ^{233}U, ^{238}U, Pu, and Th that have been proposed for use in the reactors and the choices of coextraction and partitioning of these fuels

in the reprocessing systems. In evaluating the development needed for the various operations, we assumed that mixtures of ^{235}U and ^{238}U would be converted to UF_6 for return to an isotope enrichment facility. When ^{233}U was present, we assumed that it would be enriched by blending with additional ^{233}U or enriched ^{235}U. Since the plutonium from recycled power reactor fuels would contain relatively large quantities of ^{238}Pu, the conversion and refabrication would require remotely operated equipment. Commercial-scale equipment for these operations has not been developed.

Columns giving Material Location and Material Description are included for used in future more refined evaluations of the attractiveness of the material for proliferation. The scales used in rating the effects of the convertibility of the material and its radiation on the proliferation attractiveness are given in Chap. 1.

Figures 4.8 through 4.14 show various modifications of the Purex solvent extraction process for the reprocessing of fuels containing $^{235-238}U$ or $^{233-235-238}U$ and plutonium. These modifications include the flowsheets for partitioning or coextraction of the uranium and plutonium or systems in which the plutonium remains with the fission products in the aqueous waste. Figures 4.15 through 4.21 give similar modifications for the Thorex process for the reprocessing of either ^{233}U and thorium or mixtures of ^{233}U, thorium, and plutonium. There is only limited laboratory-scale data for the three-component system (U-Pu-Th), but the chemistry of the system indicates that processes of the types presented could be successfully operated.

The fuel refabrication flowsheets can all be described in terms of two functional flow diagrams. Figure 4.22 is the functional flow diagram for metal-clad sintered pellet fuel. This flow diagram applies to the LWR, HWR, and FBR cases. Figure 4.23 is the functional flow diagram for the fabrication of graphite fuel elements containing coated fuel microspheres. This flow diagram is peculiar to the HTGR. Tables 4.26 through 4.38 provide the preliminary proliferation analysis for the LWR, HWR, and FBR fuel cycles; each of these tables is associated with functional flow diagram 1, Fig. 4.22, Tables 4.39 through 4.46 are associated with functional flow diagram 2, Fig. 4.23, and provide the preliminary proliferation analyses for the HTGR fuel cycles. In addition to the information tabulated in the

reprocessing analyses, Tables 4.26 through 4.46 also have tabulations of uranium enrichment as LEU, fissile content less than 4%; MEU, fissile content greater than 4% and less than 15% (in the case of ^{235}U, 20%); and HEU, fissile content greater than 15%. The fissile concentration of fuel mixtures is also tabulated as low, less than 4%; medium, between 4 and 15%; and high, greater than 15%.

For the HTGR cases, colocation of the reprocessing and refabrication plants has been assumed. The product of the HTGR fuel reprocessing plant will be aqueous solutions of uranium, plutonium, and thorium nitrates, which will be pumped directly to the fuel refabrication plant.

In all cases, the refabricated fuel has been assumed to be in the form of metal oxides. The functional flow diagrams for metal carbide or nitride fuels would be no different at the Level 1 stage. Thus, the distinctions between the oxide fuel and more advanced types would not influence this preliminary analysis.

The analyses of fuel refabrication methods consider the uranium enrichments that are currently used or planned for refabricated fuel: LEU for LWR, HWR, and SSCR, and MEU for FBR and HTGR. The choice tree diagram, Fig. 4.1, considers some advanced HEU fuel as well. The refabrication analyses have not been tabulated for the advanced HEU systems. Nonetheless, the HEU systems can be evaluated by modifying the reference refabrication methods indicated on the choice tree diagram to indicate a compatibility value of d for process steps 1, 2, and 3. The refabrication methods for which the HEU fuel might be considered are 1, 2, 3, 4, 7, 8, 13, and 14.

In the evaluation of the effects on proliferation potential due to the difficulty associated with the conversion of the fissionable material to a form suitable for a weapon, it was assumed that low-enriched uranium (LEU) would require isotopic enrichment but that HEU or plutonium would require only chemical conversion. It was also assumed that the irradiation level of plutonium separated from fission products would not be a major deterrent to its conversion to weapons material. On the other hand, the ^{232}U content of a weapon quantity of ^{233}U would result in a much higher irradiation field and was considered a deterrent.

4.3 CONCLUSION FROM EVALUATION

The evaluation of the reprocessing processes shows that a major area of proliferation potential is in the "rework and recycle" and the "waste treatment" areas. Even when the main process line is designed to minimize the potential for proliferation there is a relatively good chance that off-specification materials, because of either unintentional or intentional operational errors in the plant, would contain fissionable materials that would be attractive for proliferation. Essentially all process vessels are designed so that solutions can be sent to rework or waste. Due to its isotopic dilution LEU remains unattractive under all conditions. Similarly ^{233}U is always associated with ^{232}U and so is relatively unattractive. On the other hand, plutonium has no "natural" protection and is relatively easy to separate from impurities and so is a more attractive source of material for a weapon. This study shows that this remains true even when the processing plant is designed to leave the plutonium with the fission products.

This study assumed that the ^{233}U was associated with several hundred parts per million of ^{232}U. Data that became available[5] after much of this report was prepared indicate that the ^{233}U may contain as much as 1% ^{232}U. This would indicate that a 10-kg ^{233}U weapon may contain about 100 g of ^{232}U and have a radiation field of about 10^4 R at 0.3 m (1 ft). At this level the "Radiation Hazard" would rate as "high" rather than "medium," as it was rated in this report. In order for the presence of the ^{232}U to contribute to the irradiation level of the ^{233}U, the decay products of the ^{232}U must be present. When the ^{232}U is separated from all of its decay products, the regrowth of activity is limited primarily by the 1.9-year half-life of the ^{228}Th. The activity reaches 1% of its maximum value in about 15 days. If the uranium is not separated from the thorium in the reprocessing, the regrowth of the decay products is limited by the 3.6-day half-life of ^{224}Ra. The time available for converting diverted ^{233}U to a weapon material before the decay products of the ^{232}U reach thousands of roentgens at one foot is only a few days when the uranium has been separated from thorium and is too short to be of practical value when the uranium and thorium are not separated.

4.4 REFERENCES

1. G. Borgonnavi et al., *A Preliminary Methodology for Evaluating the Proliferation Resistance of Alternative Nuclear Power Systems*, SAI-78-596-WA, Science Applications, Inc. (June 15, 1977).
2. N. L. Shapiro and R. A. Matzie, *Assessment of Thorium Fuel Cycles in Pressurized Water Reactors*, EPRI-NP-359, Combustion Engineering, Inc. (February 1977).
3. Letter from J. A. Lenhard (ORO) to H. Postma (ORNL), Transmittal of Candidate Alternatives Regarding Proliferation Constraints on Various Fuel Cycles, with enclosure, "Nonproliferation Alternative Systems Assessment Program — Preliminary Program Plan," May 18, 1977.
4. H. C. Carney et al., *HTGR Fuel Cycle — Definitions for Nonproliferation Assessment*, General Atomic Company (Draft).
5. T. H. Pigford and C. S. Yang, "Thorium Fuel Cycles," Draft prepared for U.S. Environmental Protection Agency under EPA Contract 68-01-1962 with Science Applications, Inc. (June 1977).

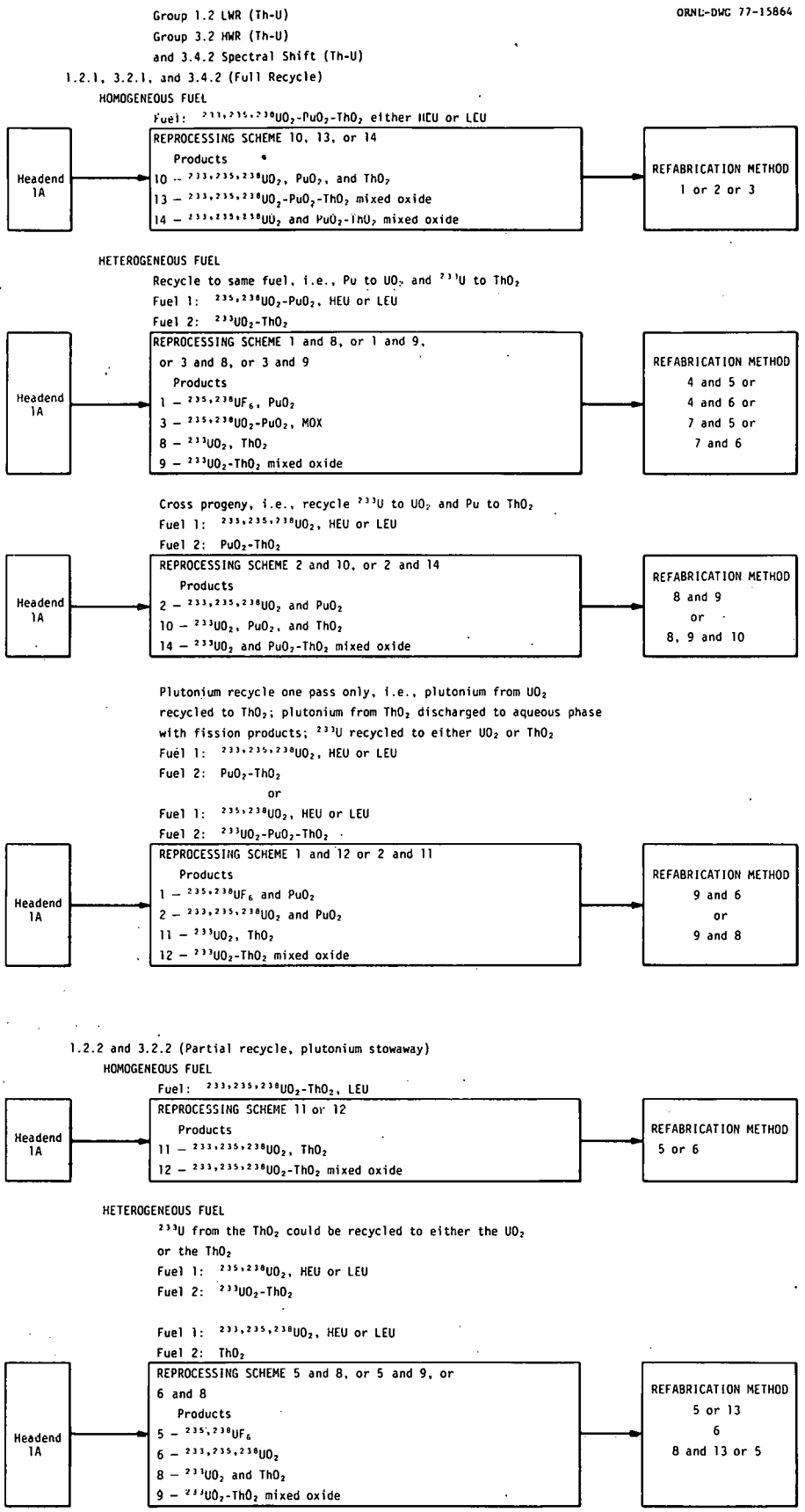


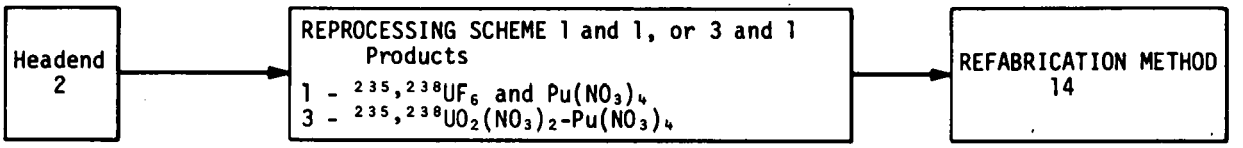
Fig. 4.1. Choice Tree Diagram for Recycle of Th-U Fuels in LWRs, HWRs, and SSCRs.

ORNL-DWG 77-15865

GROUP 2.1 GAS-COOLED REACTORS (U-Pu CYCLE)

2.1.1 Standard Recycle in HTGR
2.1.2 Spiked Recycle in HTGR
2.1.7 Standard Recycle in AGR

Fuel 1: $^{235,238}UO_2$-PuO_2, HEU
Fuel 2: $^{235,238}UO_2$ (natural)

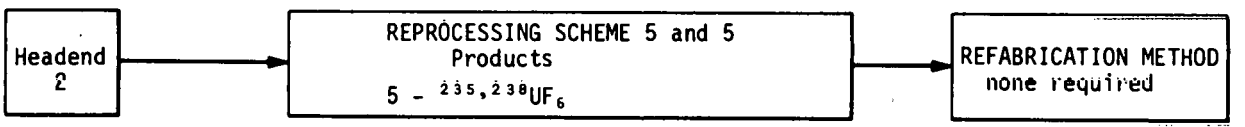


2.1.3 Coprocessed Recycle in HTGR

Coprocessing both fuel types would not be a satisfactory procedure. The plutonium concentration in the natural uranium would increase continuously so the design of the fuel would have to change continuously. See above for acceptable choices.

2.1.4 Uranium Recycle with Plutonium Throwaway (stowaway)

Fuel 1: $^{235,238}UO_2$, HEU
Fuel 2: $^{235,238}UO_2$ (natural)



2.1.5 Once-through cycle----oxide fuel
2.1.6 Once-through cycle----metallic fuel

Fig. 4.2. Choice Tree Diagram for Recycle of U-Pu Fuels in HTGRs.

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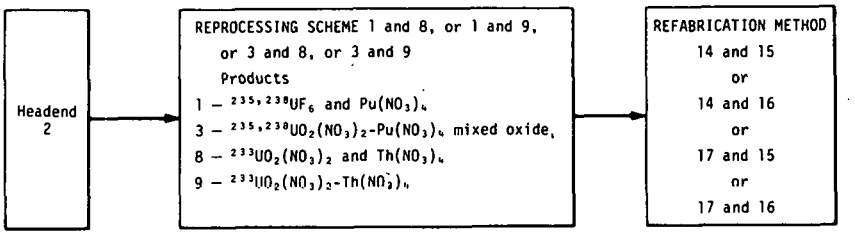
GROUP 2.2 HTGR (Th-U Cycle)

2.2.1 Full Recycle

Recycle-Fissile-Materials-to-same-Microspheres

Fuel 1: $^{235,238}UO_2$-PuO_2

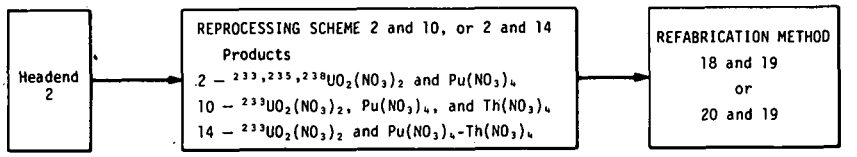
Fuel 2: $^{233}UO_2$-ThO_2



Cross Progeny; Recycle Fissile Materials to Opposite Microspheres

Fuel 1: $^{233,235,238}UO_2$

Fuel 2: PuO_2-ThO_2

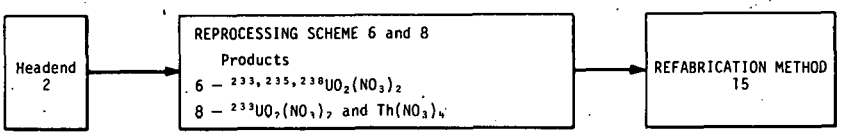


2.2.2 Partial Recycle (Plutonium to Aqueous with Fission Products)

Recycle ^{233}U to the UO_2 Fuel

Fuel 1: $^{233,235,238}UO_2$

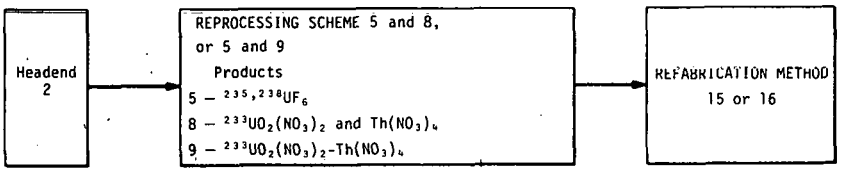
Fuel 2: ThO_2



Recycle ^{233}U to the ThO_2 Fuel

Fuel 1: $^{235,238}UO_2$ HEU or LEU

Fuel 2: $^{233}UO_2$-ThO_2



2.2.3 HEU-Th(Carbide) HTGR Full Recycle (Reforonco)

2.2.4 HEU-Th in Pebble-Bed Gas Reactor with Full Recycle (Reference)

Fuel 1: $^{233,235,238}UO_2$

Fuel 2: ThO_2

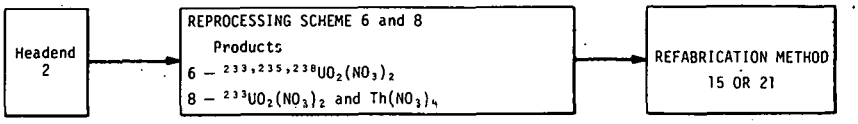


Fig. 4.3. Choice Tree Diagram for Recycle of Th-U Fuels in HTGRs.

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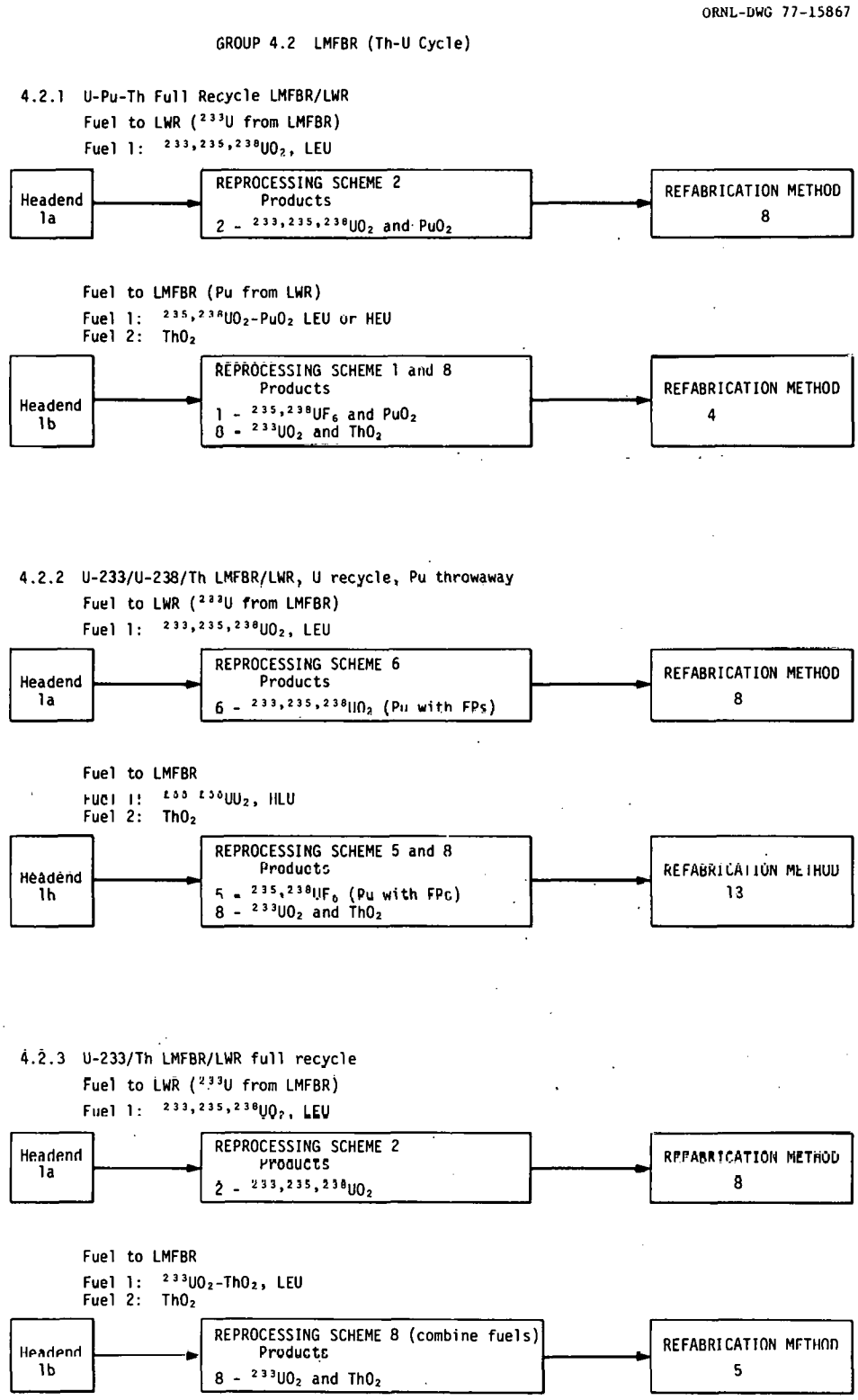

Fig. 4.4. Choice Tree Diagram for Recycle of Th-U Fuels in LMFBRs.

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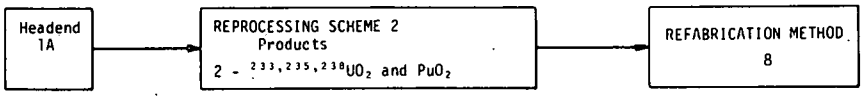
GROUP 5.2 GCFR (Th-U Cycle)

5.2.1 U-Pu-Th Full Recycle GCFR/LWR

Fuel to LWR (^{233}U from GCFR)

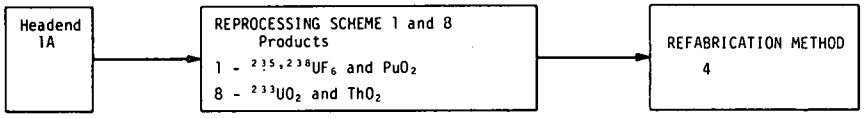
Fuel 1: $^{233,235,238}UO_2$, LEU



Fuel to GCFR (Pu from LWR)

Fuel 1: $^{235,238}UO_2$-PuO_2, LEU or HEU

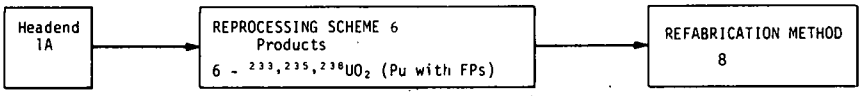
Fuel 2: ThO_2



5.2.2 U-233/U-238/Th GCFR/LWR, U recycle, Pu stowaway

Fuel to LWR (^{233}U from GCFR)

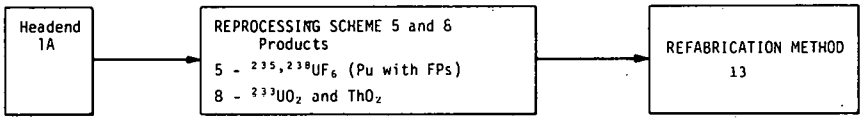
Fuel 1: $^{233,235,238}UO_2$, LEU



Fuel to GCFR

Fuel 1: ^{235,238} UO_2, HEU

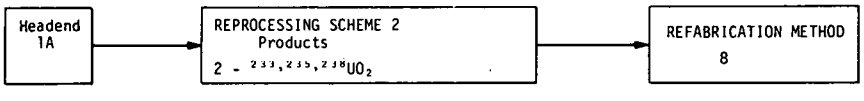
Fuel 2: ThO_2



5.2.3 U-233/Th GCFR/LWR, full recycle

Fuel to LWR (^{233}U from GCFR)

Fuel 1: $^{233,235,238}UO_2$, LEU



Fuel to GCFR

Fuel 1: $^{233}UO_2$-ThO_2, LEU

Fuel 2: ThO_2

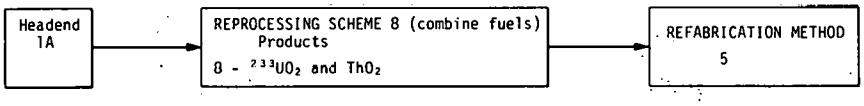


Fig. 4.5. Choice Tree Diagram for Recycle of Th-U Fuels in GCFRs.

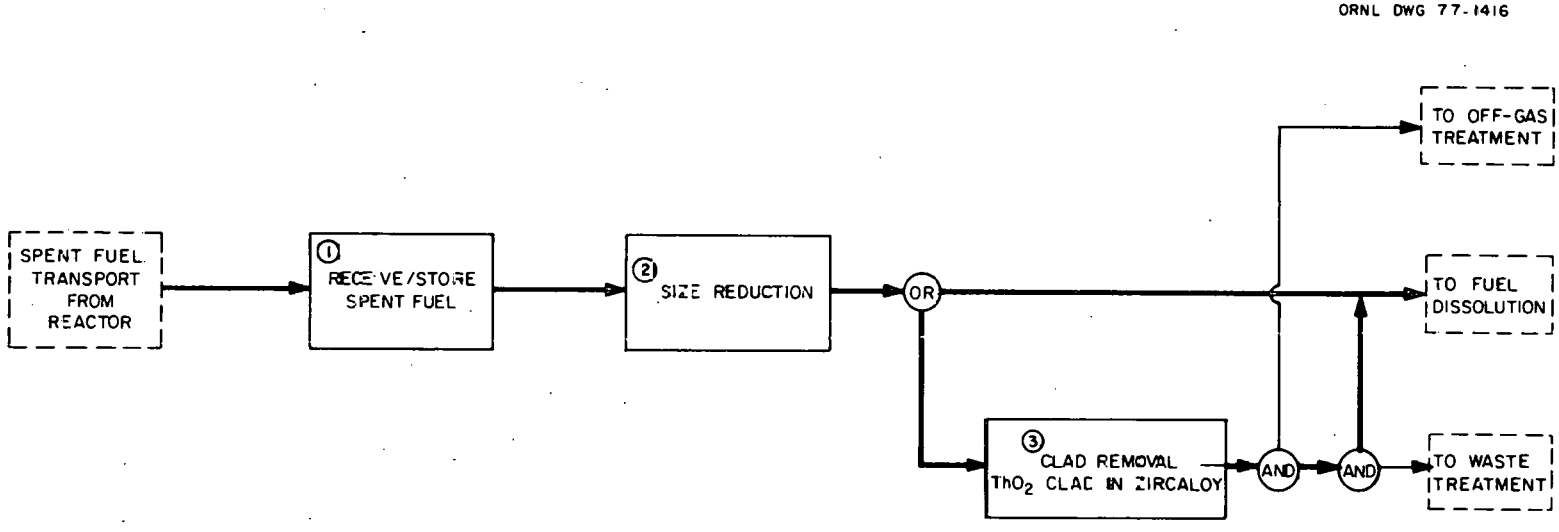


Fig. 4.6. Head-End Scheme 1 — Level 1 Functional Flow Diagram: Head-End Treatment of LWR, HWR, and FBR Fuels.

Table 4.1. Head-End Processing Scheme 1

Process Step and Operation		Development Needed	Material Location	Material Description	Convertibility	Radiation Hazard
		Water-Cooled Reactor Fuels and GCFR				
1	Receiving and storage	Developed	Under water	Irradiated fuel elements	c or less*	High
2	Size reduction	Developed	Hot cell	Irradiated fuel elements	c or less*	High
3	Cladding removal	Cold lab or hot engineering	Hot cell	Irradiated fuel elements	c or less*	High
		Liquid-Metal-Cooled Reactor Fuel				
1	Receiving and storage	Cold engineering	Storage undecided	Irradiated fuel elements	c or less*	High
2	Size reduction	Cold engineering	Hot cell	Irradiated fuel elements	c or less*	High
3	Cladding removal	Cold laboratory or hot engineering	Hot cell	Irradiated fuel elements	c or less*	High

*Could also require isotope separation, which would lower the rating of the convertibility.

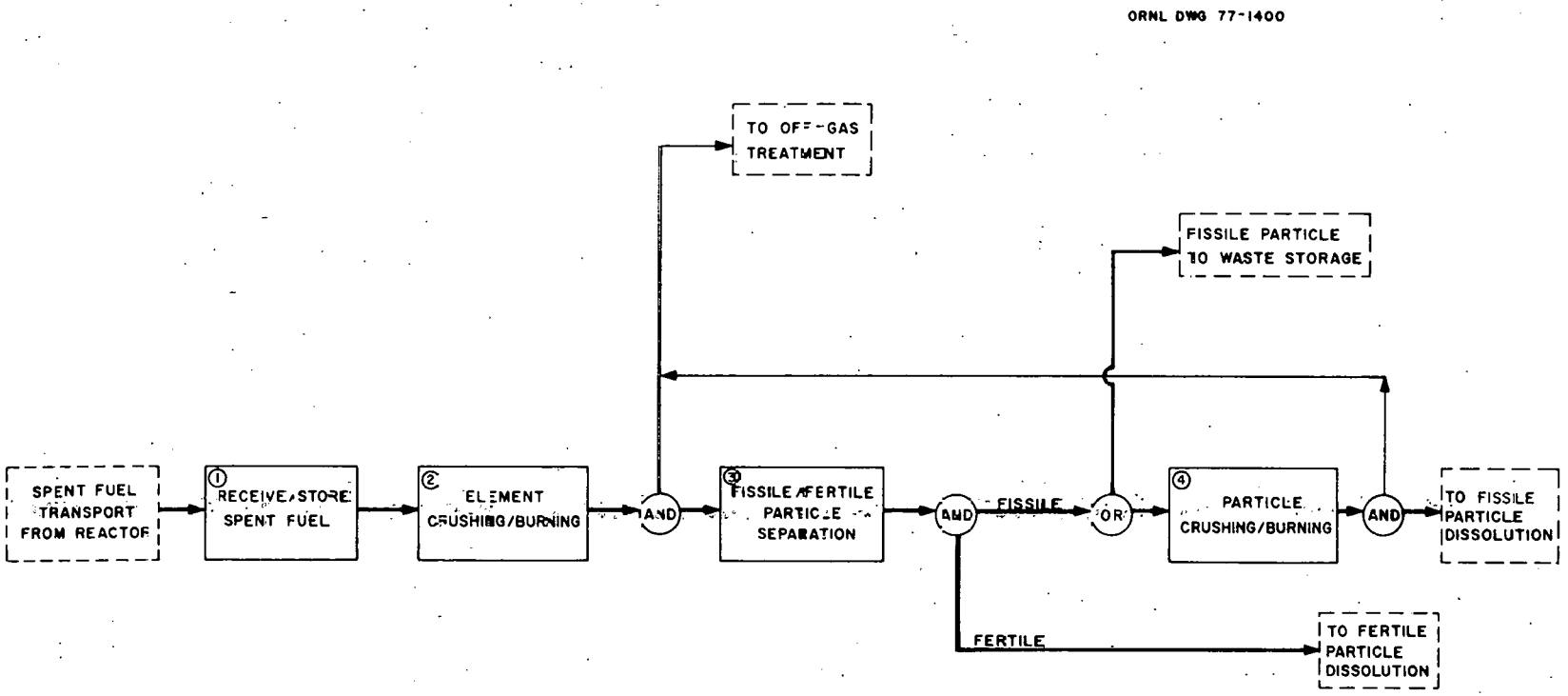


Fig. 4.7. Head-End Scheme 2 — Level 1 Functional Flow Diagram: Head-End Treatment of HTGR Fuel.

Table 4.2. Head-End Reprocessing Scheme 2 for HTGR

Process Step and Operation	Development Needed	Material Location	Material Description	Convertibility	Radiation Hazard
1 Receiving and storage	Cold engineering	Vault storage	Irradiated fuel blocks	c or less*	High
2 Fuel element crush and burn	Cold engineering	Hot cell	Irradiated microspheres	c or less*	High
3 Fissile-fertile separation	Cold engineering	Hot cell	Irradiated microspheres	c or less*	High
4 Particle crush and burn	Cold engineering	Hot cell	Irradiated fuel kernels	c or less*	High

*Could also require isotope separation, which would lower the rating of the convertibility.

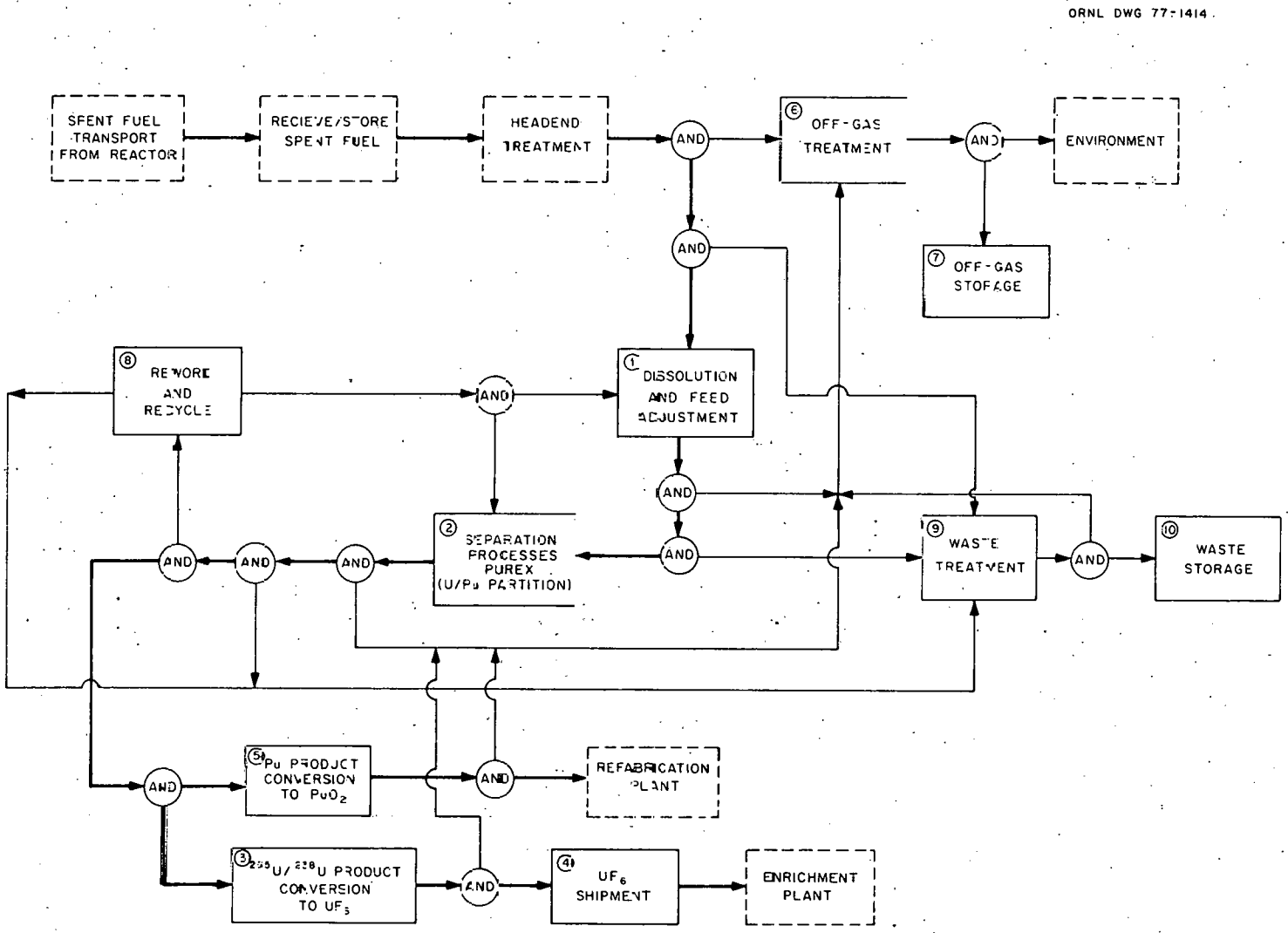


Fig. 4.8. Reprocessing Scheme 1 — Level 1 Functional Flow Diagram: Reprocessing $(^{235}U, ^{238}U)O_2$-PuO_2 Fuel by the Purex Process. Partition of uranium and plutonium.

Table 4.3. Analysis for Reprocessing Scheme 1 for <20% ^{235}U

Feed: $^{235,238}UO_2(NO_3)_2$-$Pu(NO_3)_4$ (<20% ^{235}U)

Products: (1) $^{235,238}UF_6$ (<20% ^{235}U)
(2) PuO_2

Process Step and Operation		Development Needed	Material Location	Material Description	Convertibility	Radiation Hazard
1	Dissolution and feed adjustment	Cold engineering	Hot cell	Dissolved fuel material	c (Pu)	High
2	Separations process	Prototype	Hot cell	$UO_2(NO_3)_2$ or $Pu(NO_3)_4$	d (Pu)	Low
3	Uranium conversion	Developed	Semiremote facility	UO_2 and UF_6	b	Low
4	Ship UF_6 product	Developed	Gas cylinders	UF_6	b	Low
5	Plutonium conversion	Cold engineering	Shielded alpha facility	PuO_2	d	Low
6	Off-gas treatment	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	
7	Gas storage	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	
8	Rework and recycle	Hot engineering	Remote facility	Radioactive liquid	c to d	Medium to low
9	Waste treatment	Cold engineering	Remote facility	Radioactive liquid	c to d	High to medium
10	Waste storage	Hot engineering	Remote facility	Radioactive liquid	c	High

Table 4.4. Analysis for Reprocessing Scheme 1 for >20% ^{235}U

Feed: $^{235,238}UO_2(NO_3)_2$-$Pu(NO_3)_4$ (>20% ^{235}U)

Products: (1) ^{235} $^{238}UF_6$ (>20% ^{235}U)
(2) PuO_2

Process Step and Operation		Development Needed	Material Location	Material Description	Convertibility	Radiation Hazard
1	Dissolution and feed adjustment	Cold engineering	Hot cell	Dissolved fuel material	c	High
2	Separations process	Prototype	Hot cell	$UO_2(NO_3)_2$ or $Pu(NO_3)_4$ solution	c	Low
3	Uranium conversion	Developed	Semiremote facility	UO_2 and UF_6	d	Low
4	Ship UF_6 product	Developed	Gas cylinders	UF_6	d	Low
5	Plutonium conversion	Cold engineering	Shielded alpha facility	PuO_2	d	Low
6	Off-gas treatment	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	
7	Gas storage	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	
8	Rework and recycle	Hot engineering	Remote facility	Radioactive liquid	c to d	Medium
9	Waste treatment	Cold engineering	Remote facility	Radioactive liquid	c to d	High
10	Waste storage	Hot engineering	Remote facility	Radioactive liquid	c	High

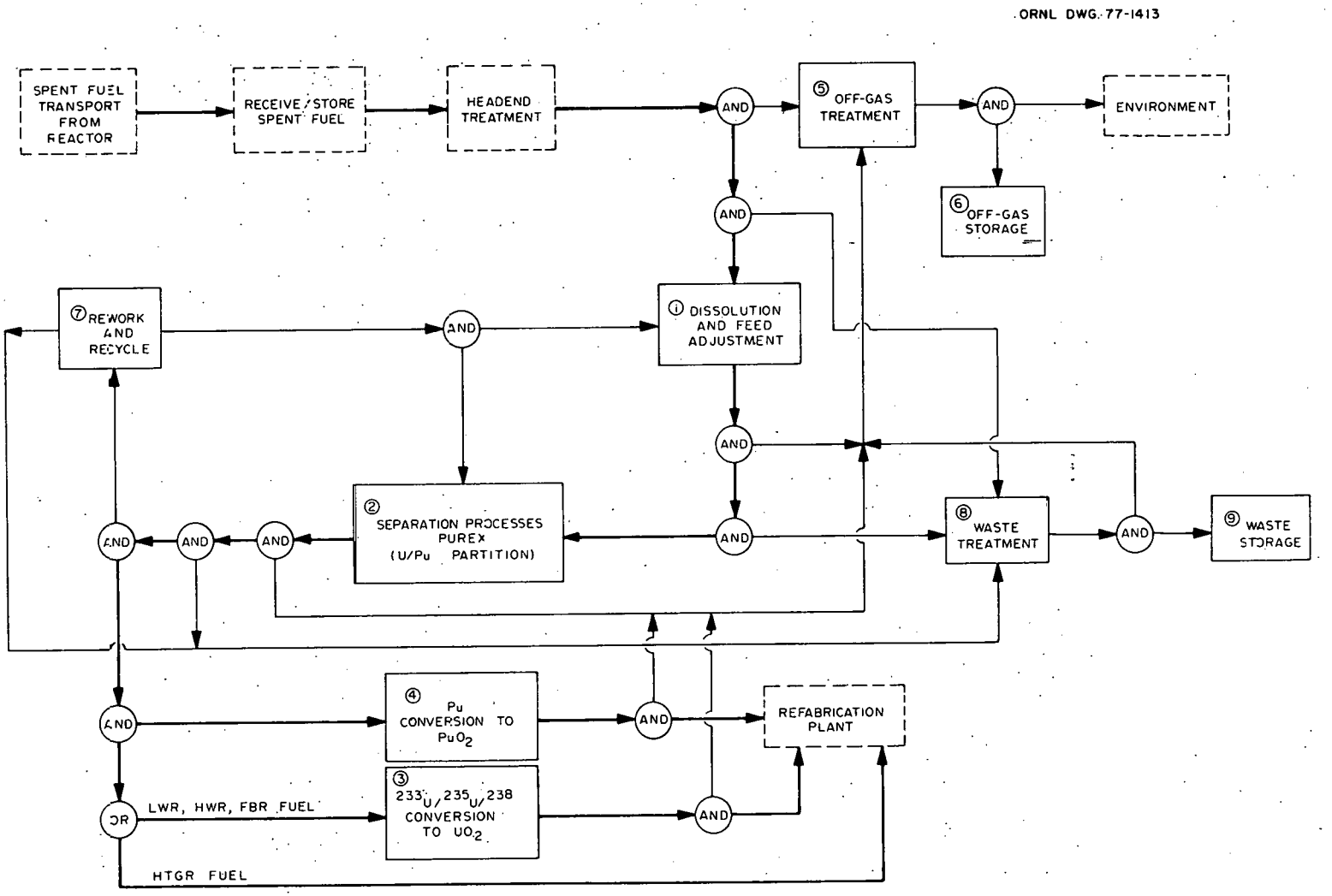


Fig. 4.9. Reprocessing Scheme 2 — Level 1 Functional Flow Diagram: Reprocessing (^{233}U, $^{238}U)O_2$ Fuel by the Purex Process. Partition of uranium and plutonium.

Table 4.5. Analysis for Reprocessing Scheme 2 for <15%-Enriched ^{233}U

Feed: $^{233,238}UO_2(NC_3)_2$-$Pu(NO_3)_4$ (<15% ^{233}U)

Products: (1) $^{233,238}UO_2$ (<15% ^{233}U)
(2) PuO_2

	Process Step and Operation	Development Needed	Material Location	Material Description	Convertibility	Radiation Hazard
1	Dissolution and feed adjustment	Cold engineering	Hot cell	Dissolved fuel material	c (Pu)	High
2	Separations process	Prototype	Hot cell	$UO_2(NO_3)_2$ or $Pu(NO_3)_4$ solution	d (Pu)	Low
3	Uranium conversion	Hot engineering	Shielded alpha facility	UO_2	d	Medium
4	Plutonium conversion	Cold engineering	Shielded alpha facility	PuO_2	d	Low
5	Off-gas treatment	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	
6	Gas storage	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	
7	Rework and recycle	Hot engineering	Remote facility	Radioactive liquid	c to d	Medium to low
8	Waste treatment	Cold engineering	Remote facility	Radioactive liquid	c to d	High to medium
9	Waste storage	Hot engineering	Remote facility	Radioactive liquid	c	High

Table 4.6. Analysis for Reprocessing Scheme 2 for >15%-Enriched ^{233}U

Feed: $^{233,238}UO_2(NO_3)_2$-$Pu(NO_3)_4$ (>15% ^{233}U)

Products: (1) $^{233,238}UO_2$
(2) PuO_2

	Process Step and Operation	Development Needed	Material Location	Material Description	Convertibility	Radiation Hazard
1	Dissolution and feed adjustment	Cold engineering	Hot cell	Dissolved fuel material	c	High
2	Separations process	Prototype	Hot cell	$UO_2(NO_3)_2$ or $Pu(NO_3)_4$ solution	d (Pu)	Low
3	Uranium conversion	Hot engineering	Shielded alpha facility	UO_2	d*	Medium
4	Plutonium conversion	Cold engineering	Shielded alpha facility	PuO_2	d	Low
5	Off-gas treatment	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	Medium
6	Gas storage	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	Medium
7	Rework and recycle	Hot engineering	Remote facility	Radioactive liquid	c to d	Medium to low
8	Waste treatment	Cold engineering	Remote facility	Radioactive liquid	c to d	High to medium
9	Waste storage	Hot engineering	Remote facility	Radioactive liquid	c	High

*A 10 kg batch of ^{233}U in equilibrium with the daughters of the ^{232}U it contains would result in about 100 to 1000 R radiation field at one foot distance.

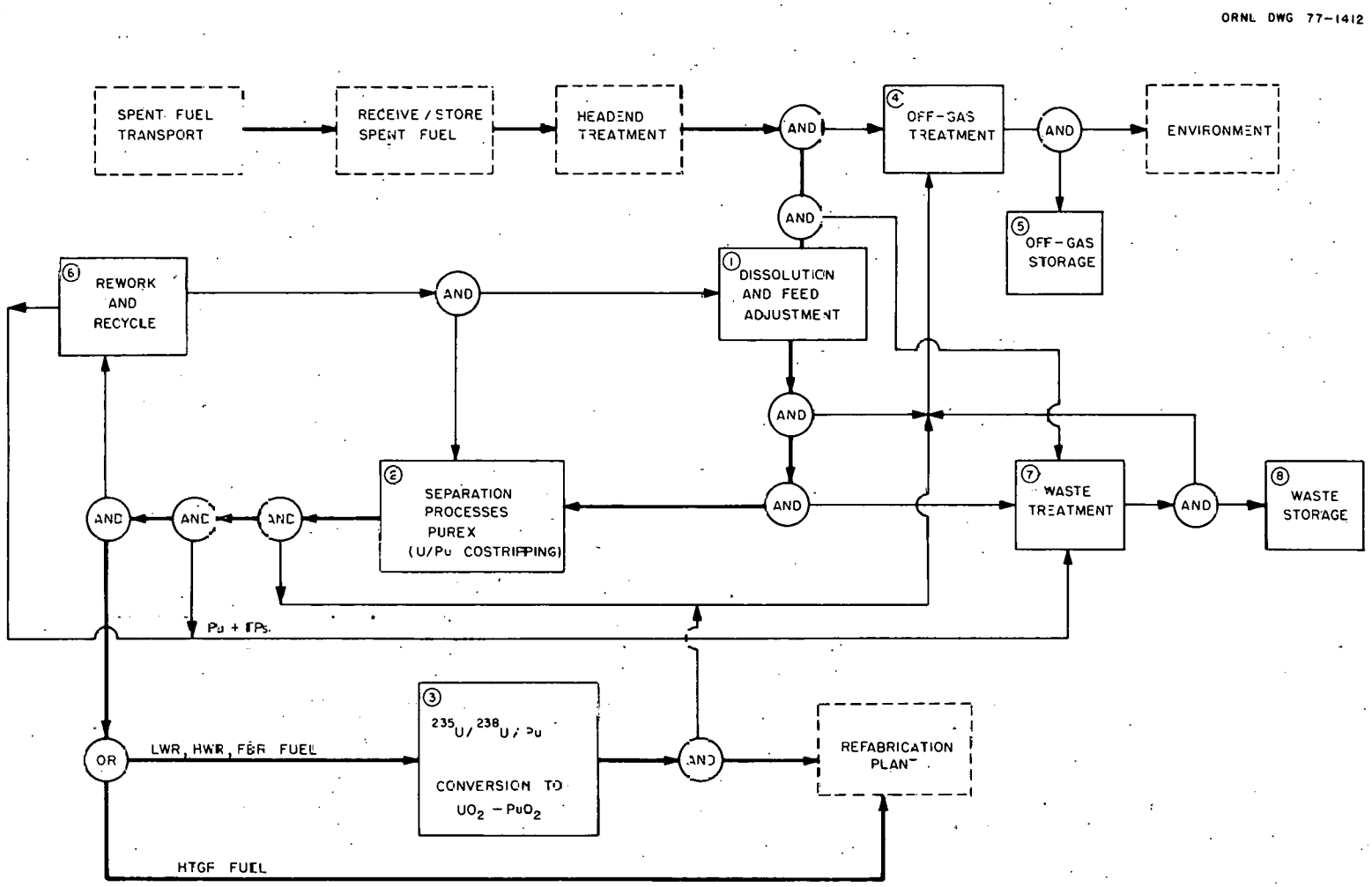


Fig. 4.10. Reprocessing Scheme 3 — Level 1 Functional Flow Diagram: Reprocessing (^{235}U, $^{238}U)O_2$-PuO_2 Fuel by the Purex Process. Costripping of Uranium and Plutonium.

Table 4.7. Analysis for Reprocessing Scheme 3 HEU or LEU — Factors Determined by Plutonium

Feed: $^{235,238}UO_2(NO_3)_2$-$Pu(NO_3)_4$* Product: $^{235,238}UO_2$-PuO_2

	Process Step and Operation	Development Needed	Material Location	Material Description	Convertibility	Radiation Hazard
1	Dissolution and feed adjustment	Cold engineering	Hot cell	Dissolved fuel material	c	High
2	Separations process	Prctotype	Hot cell	$UO_2(NO_3)_2$-$Pu(NO_3)_4$ solution	d	Low
3	Conversion to MOX	Cold engineering	Shielded alpha facility	UO_2-PuO_2	d	Low
4	Off-gas treatment	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	Medium
5	Gas storage	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	Medium
6	Rework and recycle	Hot engineering	Remote facility	Radioactive liquid	c to d	Medium to low
7	Waste treatment	Cold engineering	Remote facility	Radioactive liquid	c to d	High to medium
8	Waste storage	Hot engineering	Remote facility	Radioactive liquid	c	High

*The uranium can be any concentration of ^{235}U in ^{238}U.

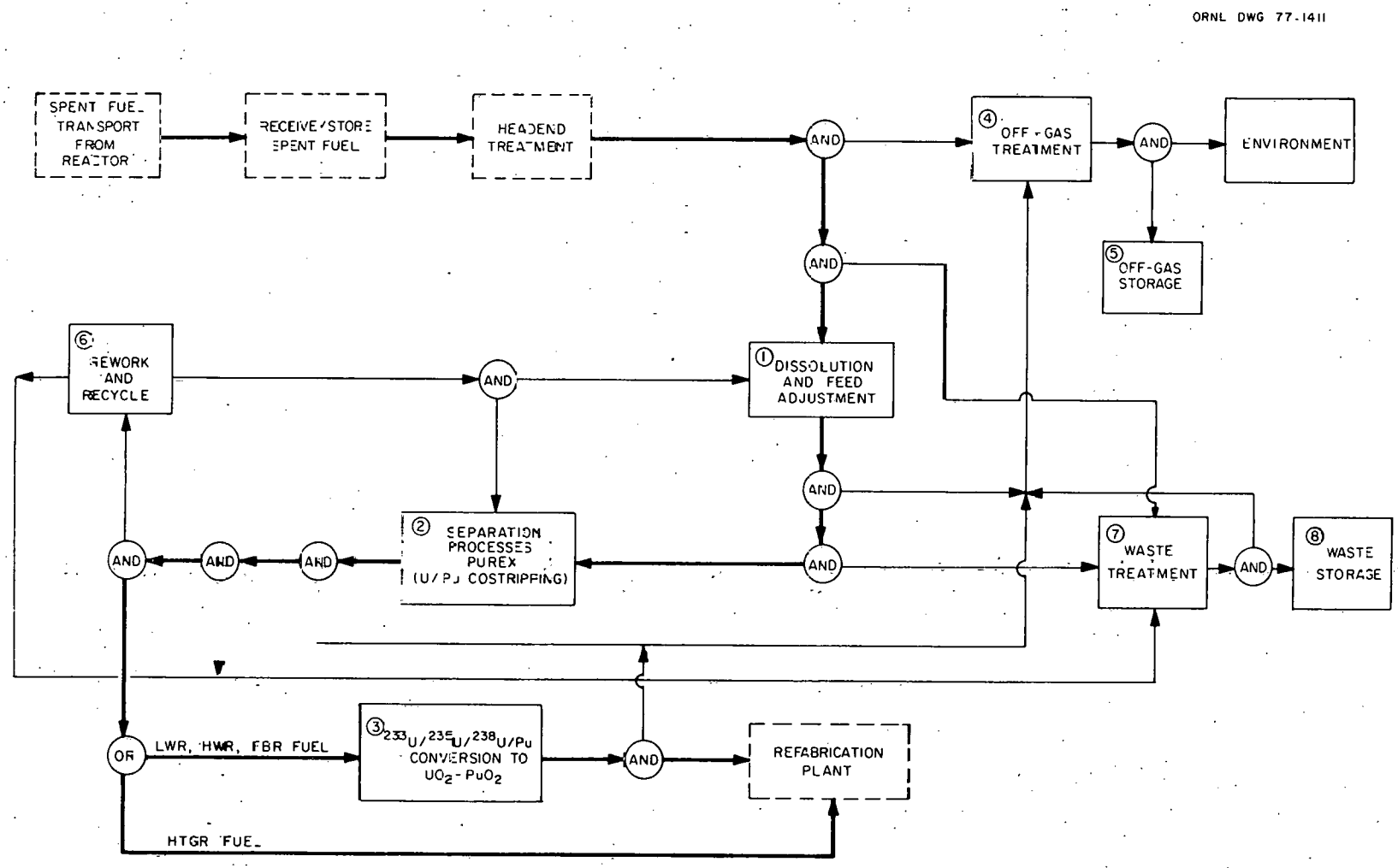


Fig. 4.11. Reprocessing Scheme 4 — Level 1 Functional Flow Diagram: Reprocessing (^{233}U, ^{235}U, ^{238}U)O_2-PuO_2 Fuel by the Purex Process, Costripping of Uranium and Plutonium.

Table 4.8. Analysis for Reprocessing Scheme 4 HEU or LEU — Factors Determined by Plutonium

Feed: $^{233,238}UO_2(NO_3)_2$-$Pu(NO_3)_4$* Product: $^{233,238}UO_2$-PuO_2

Process Step and Operation		Development Needed	Material Location	Material Description	Convertibility	Radiation Hazard
1	Dissolution and feed adjustment	Cold engineering	Hot cell	Dissolved fuel material	c	High
2	Separations process	Prototype	Hot cell	$UO_2(NO_3)_2$-$Pu(NO_3)_4$ solution	d	Low
3	MOX conversion	Cold engineering	Semiremote facility	UO_2-PuO_2	d	Low
4	Off-gas treatment	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	Medium
5	Gas storage	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	Medium
6	Rework and recycle	Hot engineering	Remote facility	Radioactive liquid	c to d	Medium to low
7	Waste treatment	Cold engineering	Remote facility	Radioactive liquid	c to d	High to medium
8	Waste storage	Hot engineering	Remote facility	Radioactive liquid	c	High

*The uranium can be any concentration of ^{233}U in ^{238}U.

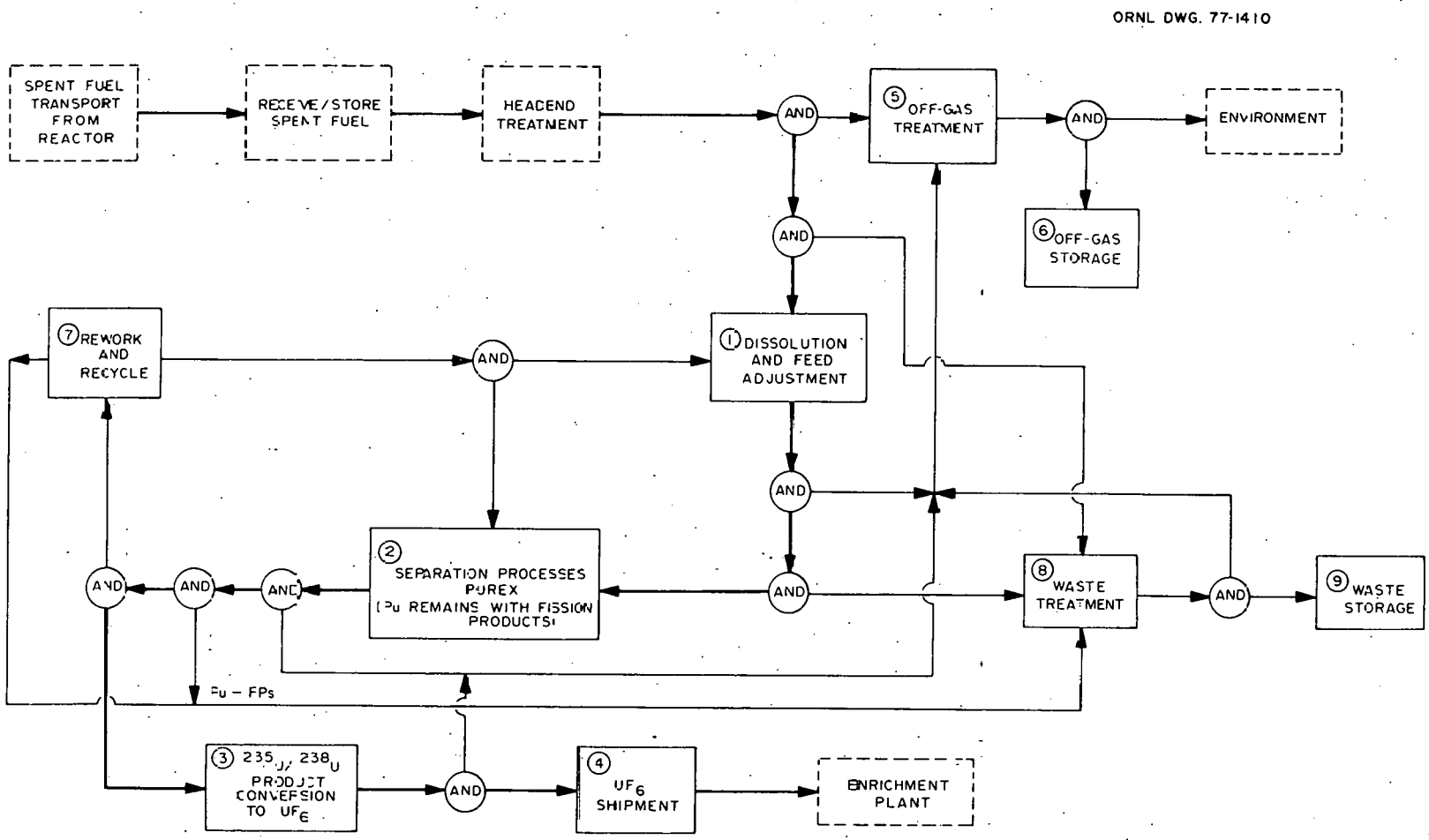


Fig. 4.12. Reprocessing Scheme 5 — Level 1 Functional Flow Diagram:- Reprocessing (^{235}U, ^{238}U-Pu)O_2 Fuel by the Purex Process. Plutonium stowaway with fission products.

Table 4.9. Analysis for Reprocessing Scheme 5 for <20%-Enriched ^{235}U

Feed: $^{235,238}UO_2(NO_3)_2$-$Pu(NO_3)_4$ (<20% ^{235}U)

Product: (1) $^{235,238}UF_6$ (<20% ^{235}U)
Plutionium to aqueous waste for storage

	Process Step and Operation	Development Needed	Material Location	Material Description	Convertibility	Radiation Hazard
1	Dissolution and feed adjustment	Cold engineering	Hot cell	Dissolved fuel material	c (Pu)	High
2	Separations process	Prototype	Hot cell	$UO_2(NO_3)_2$ solution	b*	Low
3	Uranium conversion	Developed	Semiremote facility	UO_2 and UF_6	b	Low
4	Ship UF_6 product	Developed	Gas cylinders	UF_6	b	Low
5	Off-gas treatment	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	Medium
6	Gas storage	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	Medium
7	Rework and recycle	Hot engineering	Remote facility	Radioactive liquid	b	Medium to low
8	Waste treatment	Cold engineering	Remote facility	Radioactive liquid	c to d	High to medium
9	Waste storage	Hot engineering	Remote facility	Radioactive liquid	c	High

*With relatively simple changes in the separations process flowsheet the plutonium could be recovered in a form that would enhance relative attractiveness.

Table 4.10. Analysis for Reprocessing Scheme 5 for >20%-Enriched ^{235}U

Feed: $^{235,238}UO_2(NO_3)_2$-$Pu(NO_3)_4$ (>20% ^{235}U)

Product: (1) $^{235,238}UF_6$ (>20% ^{235}U)
Plutonium to aqueous waste for storage

	Process Step and Operation	Development Needed	Material Location	Material Description	Convertibility	Radiation Hazard
1	Dissolution and feed adjustment	Cold engineering	Hot cell	Dissolved fuel material	c (Pu)	High
2	Separations process	Prototype	Hot cell	$UO_2(NO_3)_2$ solution	d*	Low
3	Uranium conversion	Developed	Semiremote facility	UO_2 and UF_6	d	Low
4	Ship UF_6 product	Developed	Gas cylinders	UF_6	d	Low
5	Off-gas treatment	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	Medium
6	Gas storage	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	Medium
7	Rework and recycle	Hot engineering	Remote facility	Radioactive liquid	c to d	Medium to low
8	Waste treatment	Cold engineering	Remote facility	Radioactive liquid	c to d	High to medium
9	Waste storage	Hot engineering	Remote facility	Radioactive liquid	c	High

*Plutonium could be recovered by making relatively simple flowsheet changes, but this would not increase the relative attractiveness.

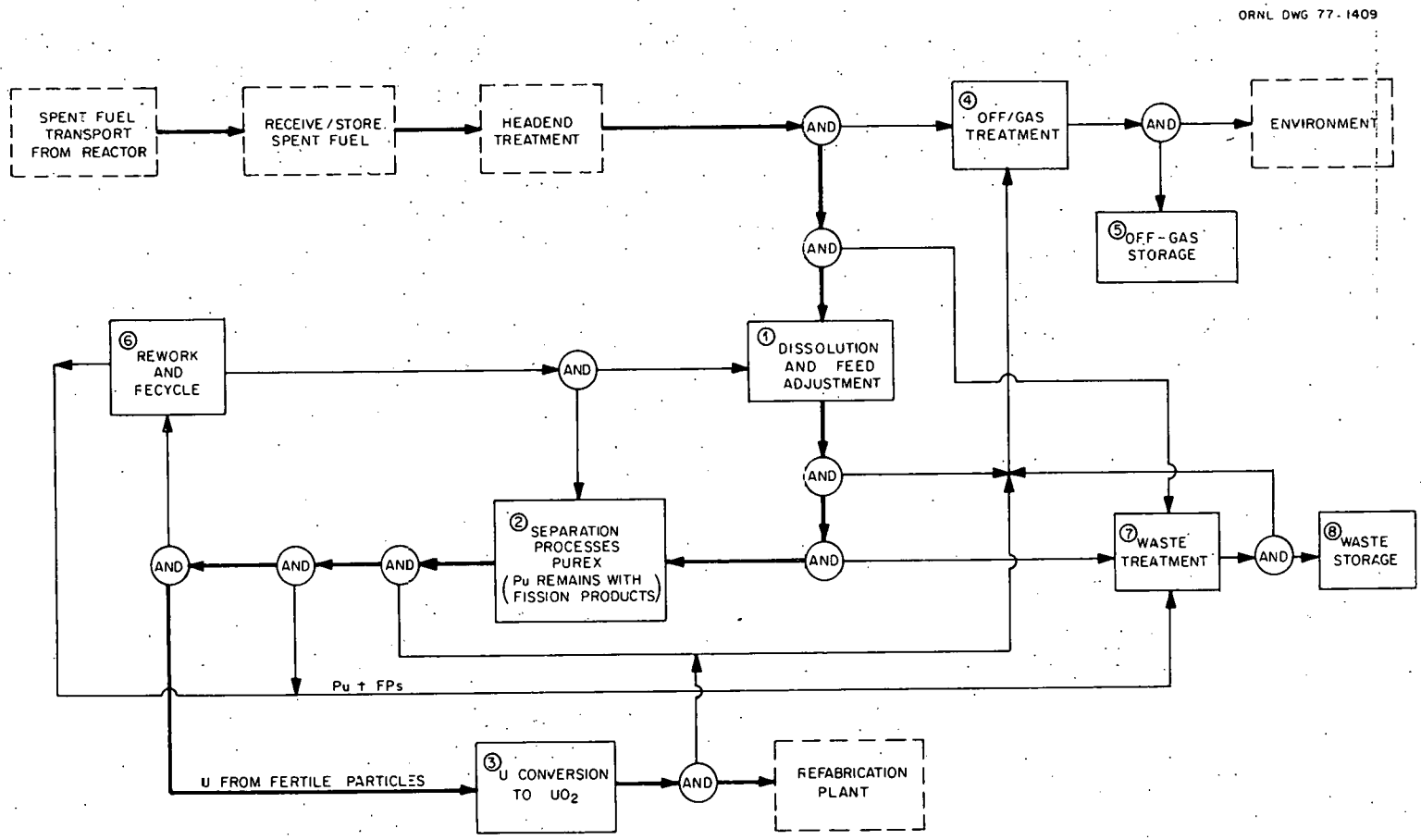


Fig. 4.13. Reprocessing Scheme 6 — Level 1 Functional Flow Diagram: Reprocessing (^{233}U, $^{238}U)O_2$-PuO_2 Fuel by the Purex Process. Plutonium stowaway with fission products.

Table 4.11. Analysis for Reprocessing Scheme 6 for <15%-Enriched ^{233}U

Feed: $^{233,238}UO_2(NO_3)_2$–$Pu(NO_3)_4$ (<15% ^{233}U)

Product: $^{233,238}UO_2(NO_3)_2$
Plutonium to aqueous waste for storage

Process Step and Operation		Development Needed	Material Location	Material Description	Convertibility	Radiation Hazard
1	Dissolution and feed adjustment	Cold engineering	Hot cell	Dissolved fuel material	c (Pu)	High
2	Separations process	Prototype	Hot cell	$UO_2(NO_3)_2$ solution	a*	Medium
3	Uranium conversion	Hot engineering	Remote alpha facility	UO_2	a*	Medium
4	Off-gas treatment	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	Medium
5	Gas storage	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	Medium
6	Rework and recycle	Hot engineering	Remote facility	Radioactive liquid	c (Pu)	High to medium
7	Waste treatment	Cold engineering	Remote facility	Radioactive liquid	c (Pu)	High to medium
8	Waste storage	Hot engineering	Remote facility	Radioactive liquid	c (Pu)	High

*With relatively simple changes in the separations process flowsheet the plutonium could be recovered.

Table 4.12. Analysis for Reprocessing Scheme 6 for >15%-Enriched ^{233}U

Feed: $^{233,238}UO_2(NO_3)_4$ (>15% ^{233}U)

Product: $^{233,238}UO_2(NO_3)_2$
Plutonium to aqueous waste for storage

Process Step and Operation		Development Needed	Material Location	Material Description	Convertibility	Radiation Hazard
1	Dissolution and feed adjustment	Cold engineering	Hot cell	Dissolved fuel material	c	High
2	Separations process	Prototype	Hot cell	$UO_2(NO_3)_2$ solution	c*	Medium
3	Uranium conversion	Hot engineering	Remote alpha facility	UO_2	d†	Medium
4	Off-gas treatment	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	Medium
5	Gas storage	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	Medium
6	Rework and recycle	Hot engineering	Remote facility	Radioactive liquid	c	High to medium
7	Waste treatment	Cold engineering	Remote facility	Radioactive liquid	c	High to medium
8	Waste storage	Hot engineering	Remote facility	Radioactive liquid	c	High

*With relatively simple changes in the separations process flowsheet the plutonium could be recovered.

†A 10 kg batch of ^{233}U in equilibrium with the daughters of the ^{232}U it contains would result in a 100 to 1000 R radiation field at one foot distance.

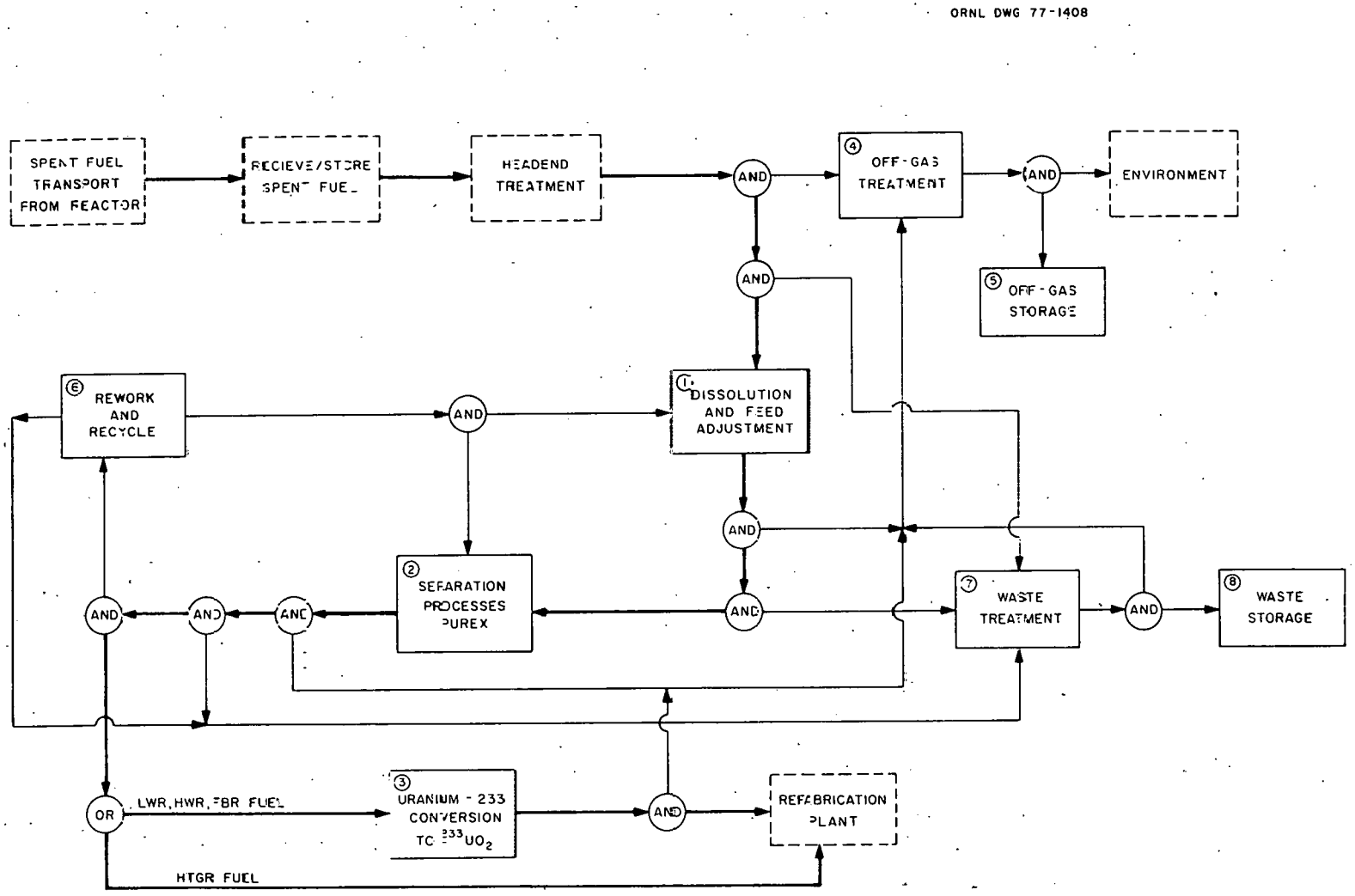


Fig. 4.14. Reprocessing Scheme 7 — Level 1 Functional Flow Diagram: Reprocessing $^{233}UO_2$ Fuel by the Purex Process.

Table 4.13. Analysis for Reprocessing Scheme 7

Feed: $^{233}UO_2(NO_3)_2$ Product: $^{233}UO_2(NO_3)_2$

Process Step and Operation		Development Needed	Material Location	Material Description	Convertibility	Radiation Hazard
1	Dissolution and feed adjustment	Cold engineering	Hot cell	Dissolved fuel material	c	High
2	Separations process	Prototype	Hot cell	$UO_2(NO_3)_2$ solution	d	Medium
3	Off-gas treatment	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	Medium
4	Gas storage	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	Medium
5	Rework and recycle	Hot engineering	Remote facility	Radioactive liquid	c to d	Medium
6	Waste treatement	Cold engineering	Remote facility	Radioactive liquid	c to d	High to medium
7	Waste storage	Hot engineering	Remote facility	Radioactive liquid	c	High

*A 10 kg batch of ^{233}U in equilibrium with the daughters of the ^{232}U it contains would result in about 100 to 1000 R radiation field at one foot distance.

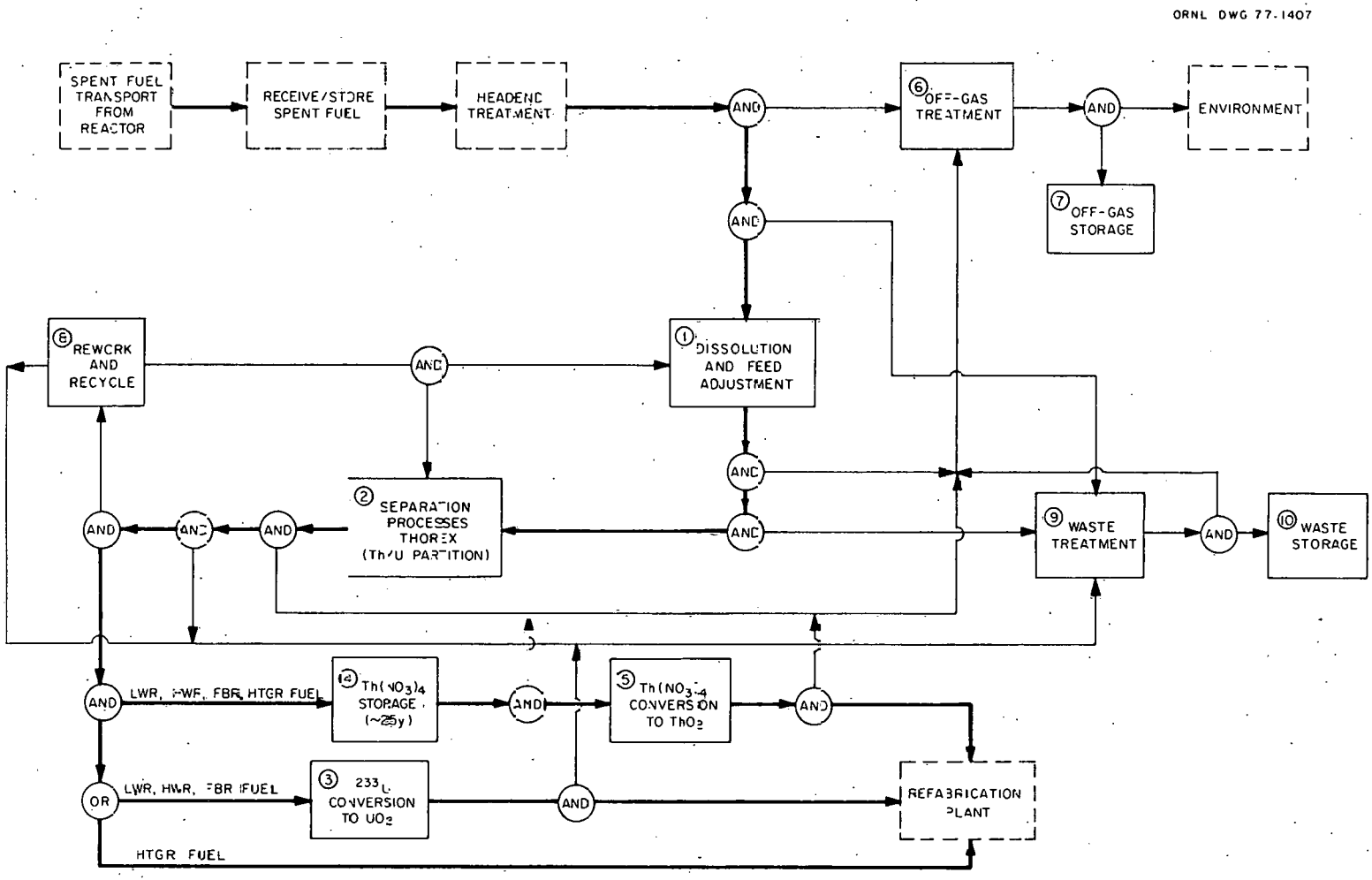


Fig. 4.15. Reprocessing Scheme 3 — Level 1 Functional Flow Diagram: Reprocessing $^{233}UO_2$-ThO_2 Fuel by the Thorex Process. Partition of thorium and uranium.

Table 4.14. Analysis for Reprocessing Scheme 8

Feed: $^{233}UO_2(NO_3)_2$-$Th(NO_3)_4$ Products: (1) $^{233}UO_2$(~100% ^{233}U) (2) ThO_2

	Process Step and Operation	Development Needed	Material Location	Material Description	Convertibility	Radiation Hazard
1	Dissolution and feed adjustment	Cold engineering	Hot cell	Dissolved fuel material	c	High
2	Separations process	Prototype	Hot cell	$UO_2(NO_3)_2$ and $Th(NO_3)_4$ solutions	c*	Medium
3	Uranium conversion	Hot engineering	Shielded alpha facility	UO_2	d*	Medium
4	Thorium product storage	Developed	Shielded tank	$Th(NO_3)_4$ solution	Nonfissionable	Medium
5	Thorium product conversion	Developed	Semiremote facility	ThO_2	Nonfissionable	Low
6	Off-gas treatment	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	Medium
7	Gas storage	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	Medium
8	Rework and recycle	Hot engineering	Remote facility	Radioactive liquid	c	High to medium
9	Waste treatment	Cold engineering	Remote facility	Radioactive liquid	c	High to medium
10	Waste storage	Hot engineering	Remote facility	Radioactive liquid	c	High

*A 10 kg batch of ^{233}U in equilibrium with the daughters of the ^{232}U it contains would result in about a 100 to 1000 R radiation field at one foot distance.

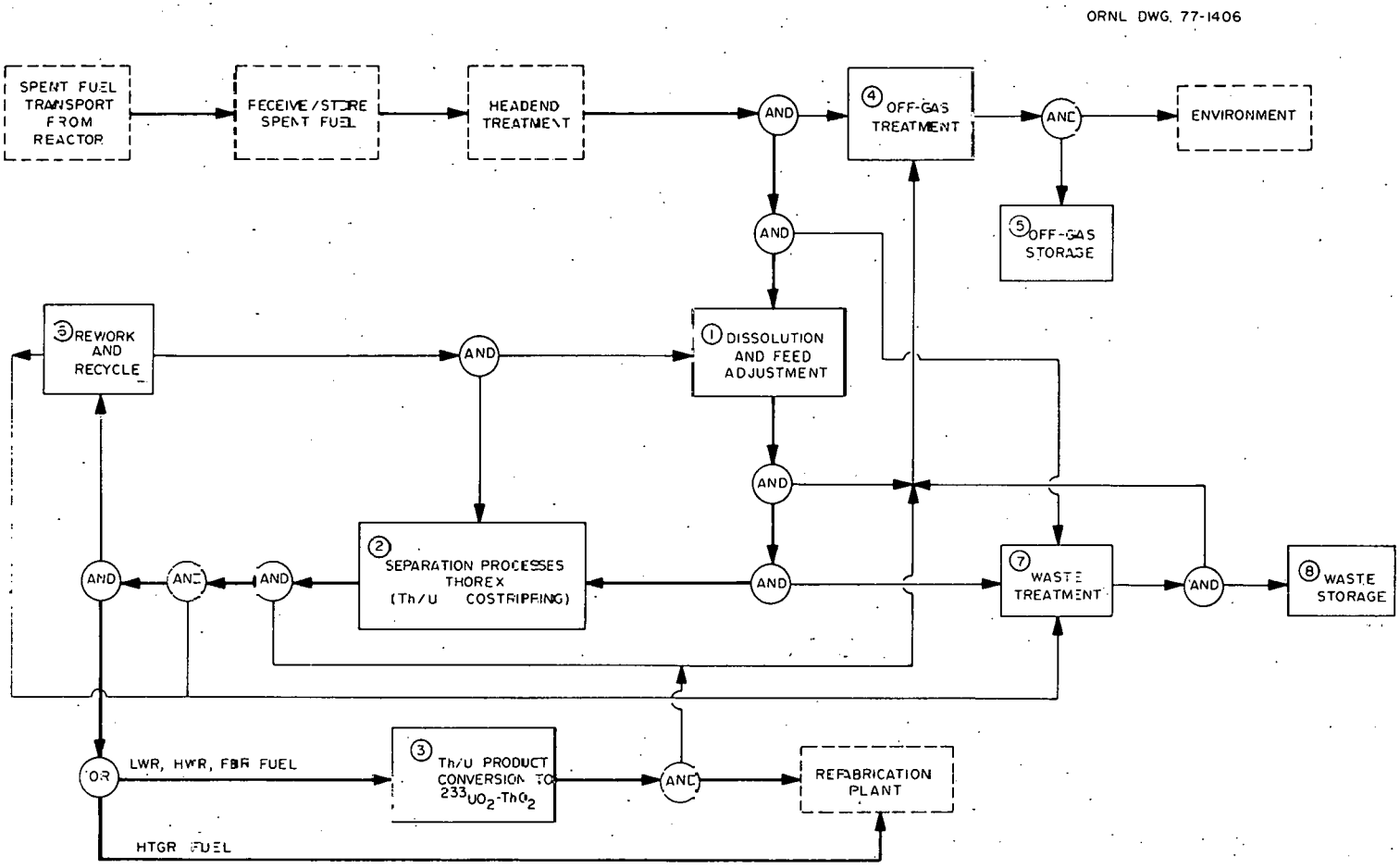


Fig. 4.16. Reprocessing Scheme 9 — Level 1 Functional Flow Diagram: Reprocessing ^{233}UO-ThO_2 Fuel by the Thorex Process. Costripping of Thorium and Uranium.

Table 4.15. Analysis for Reprocessing Scheme 9

Feed: $^{233}UO_2(NO_3)_2$-$Th(NO_3)_4$ solution

Product: (1) $^{233}UO_2$-ThO_2 (~100% ^{233}U)

Process Step and Operation		Development Needed	Material Location	Material Description	Convertibility	Radiation Hazard
1	Dissolution and feed adjustment	Cold engineering	Hot cell	Dissolved fuel material	c	High
2	Separations process	Prototype	Hot cell	$UO_2(NO_3)_2$-$Th(NO_3)_4$ solution	c*	Medium
3	Product conversion	Hot engineering	Shielded alpha facility	UO_2-ThO_2	c*	Medium
4	Off-gas treatment	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	Medium
5	Gas storage	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	Medium
6	Rework and recycle	Hot engineering	Remote facility	Radioactive liquid	c	High to medium
7	Waste treatment	Cold engineering	Remote facility	Radioactive liquid	c	High to medium
8	Waste storage	Hot engineering	Remote facility	Radioactive liquid	c	High

*A 10 kg batch of ^{233}U in equilibrium with the daughters of the ^{232}U it contains would result in about a 100 to 1000 R radiation field at one foot distance.

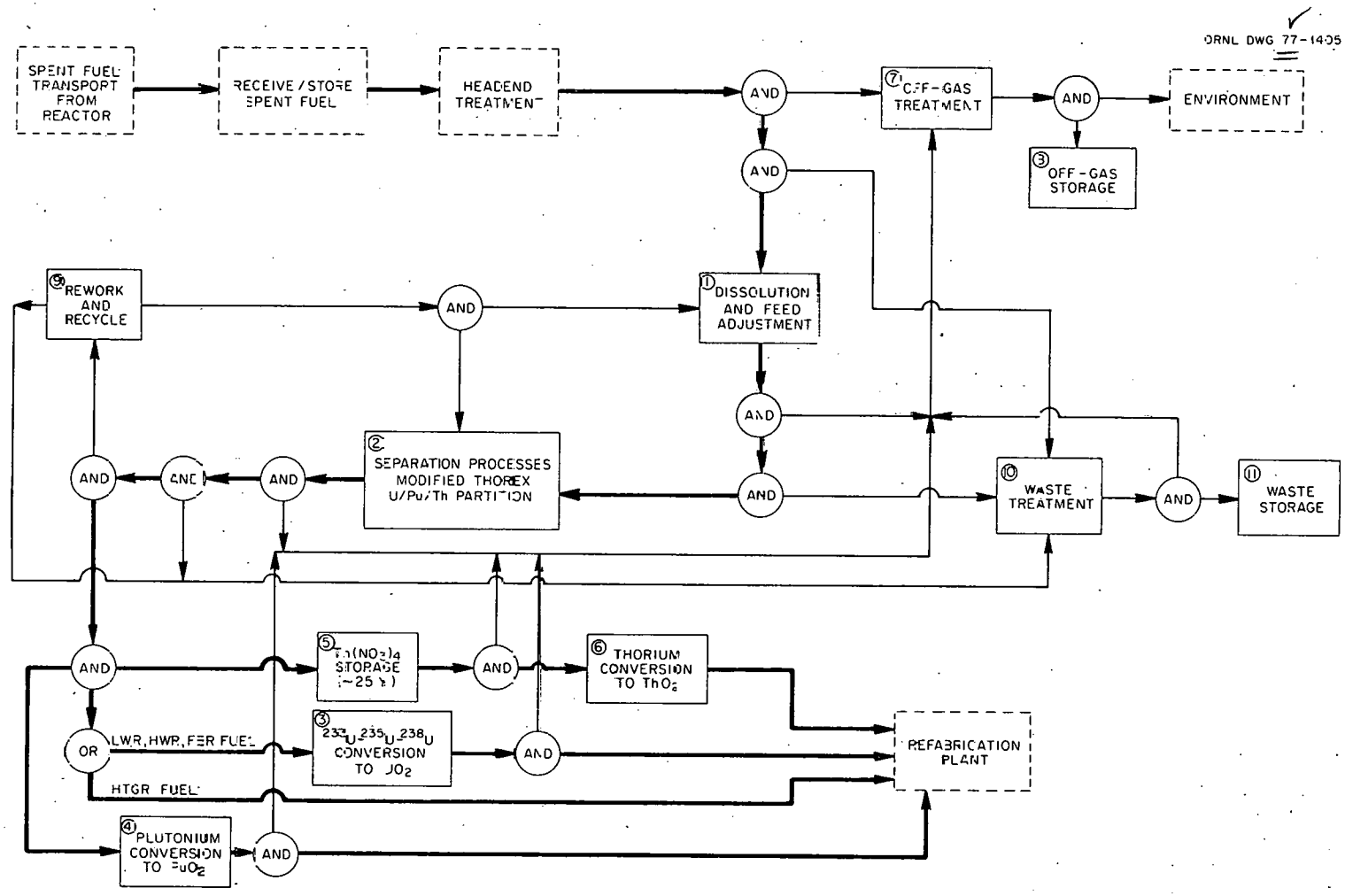


Fig. 4.17. Reprocessing Scheme 10 — Level 1 Functional Flow Diagram: Reprocessing (^{233}U, ^{235}U, ^{238}U)O_2-PuO_2-ThO_2 and $^{233}UO_2$-PuO_2-ThO_2 Fuel by the Modified Thorex Process. Partition of uranium, plutonium, and thorium.

Table 4.16. Analysis for Reprocessing Scheme 10 for <15%-Enriched $^{233}U + ^{235}U$

Feed: $^{233,235,238}UO_2(NO_3)_2$-$Pu(NO_3)_4$-$Th(NO_3)_4$ (<15% $^{233,235}U$)

Products: (1) $^{233,235,238}UO_2$ (<15% $^{233,235}U$)
(2) PuO_2
(3) ThO_2

Process Step and Operation		Development Needed	Material Location	Material Description	Convertibility	Radiation Hazard
1	Dissolution and feed adjustment	Cold engineering	Hot cell	Dissolved fuel material	c	High
2	Separations process	Prototype	Hot cell	$UO_2(NO_3)_2$, $Pu(NO_3)_4$ and $Th(NO_3)_4$ solutions	d (Pu)	Low
3	Uranium product conversion	Hot engineering	Shielded alpha facility	UO_2	a*	Medium
4	Plutonium product conversion	Cold engineering	Shielded alpha facility	PuO_2	d	Low
5	Thorium product storage	Developed	Shielded tank	$Th(NO_3)_4$	Nonfissionable	Medium
6	Thorium product conversion	Developed	Semiremote facility	ThO_2	Nonfissionable	Low
7	Off-gas treatment	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	Medium
8	Gas storage	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	Medium
9	Rework and recycle	Hot engineering	Remote facility	Radioactive liquid	c to d	Medium to Low
10	Waste treatment	Cold engineering	Remote facility	Radioactive liquid	c to d	High to medium
11	Waste storage	Hot engineering	Remote facility	Radioactive liquid	c to d	High

*A 10 kg batch of ^{233}U in equilibrium with the daughter of the ^{232}U it contains would result in a 100 to 1000 R radiation field at one foot distance.

Table 4.17. Analysis for Reprocessing Scheme 10 for >15%-Enriched $^{233}U + ^{235}U$

Feed: $^{233,235,238}UO_2(NO_3)_2$-$Pu(NO_3)_4$-$Th(NO_3)_4$ (>15% $^{233,235}U$)

Products: (1) $^{233,235,238}UO_2$ (>15% $^{233,235}U$)
(2) PuO_2
(3) ThO_2

Process Step and Operation		Development Needed	Material Location	Material Description	Convertibility	Radiation Hazard
1	Dissolution and feed adjustment	Cold	Hot cell	Dissolved fuel material	c	High
2	Separations process	Prototype	Hot cell	$UO_2(NO_3)_2$, $Pu(NO_3)_4$ $Th(NO_3)_4$ solutions	d	Low
3	Uranium product conversion	Hot engineering	Shielded alpha facility	UO_2	d*	Medium
4	Plutonium product conversion	Cold engineering	Shielded alpha facility	PuO_2	d	Low
5	Thorium product storage	Developed	Shielded tank	$Th(NO_3)_4$ solution	Nonfissionable	Medium
6	Thorium product conversion	Developed	Semiremote facility	ThO_2	Nonfissionable	Low
7	Off-gas treatment	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	Medium
8	Gas storage	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	Medium
9	Rework and recycle	Hot engineering	Remote facility	Radioactive liquid	c to d	Medium to low
10	Waste treatment	Cold engineering	Remote facility	Radioactive liquid	c to d	High to medium
11	Waste storage	Hot engineering	Remote facility	Radioactive liquid	c	High

*A 10 kg batch of ^{233}U in equilibrium with the daughters of the ^{232}U it contains would result in a 100 to 1000 R radiation field at one foot distance.

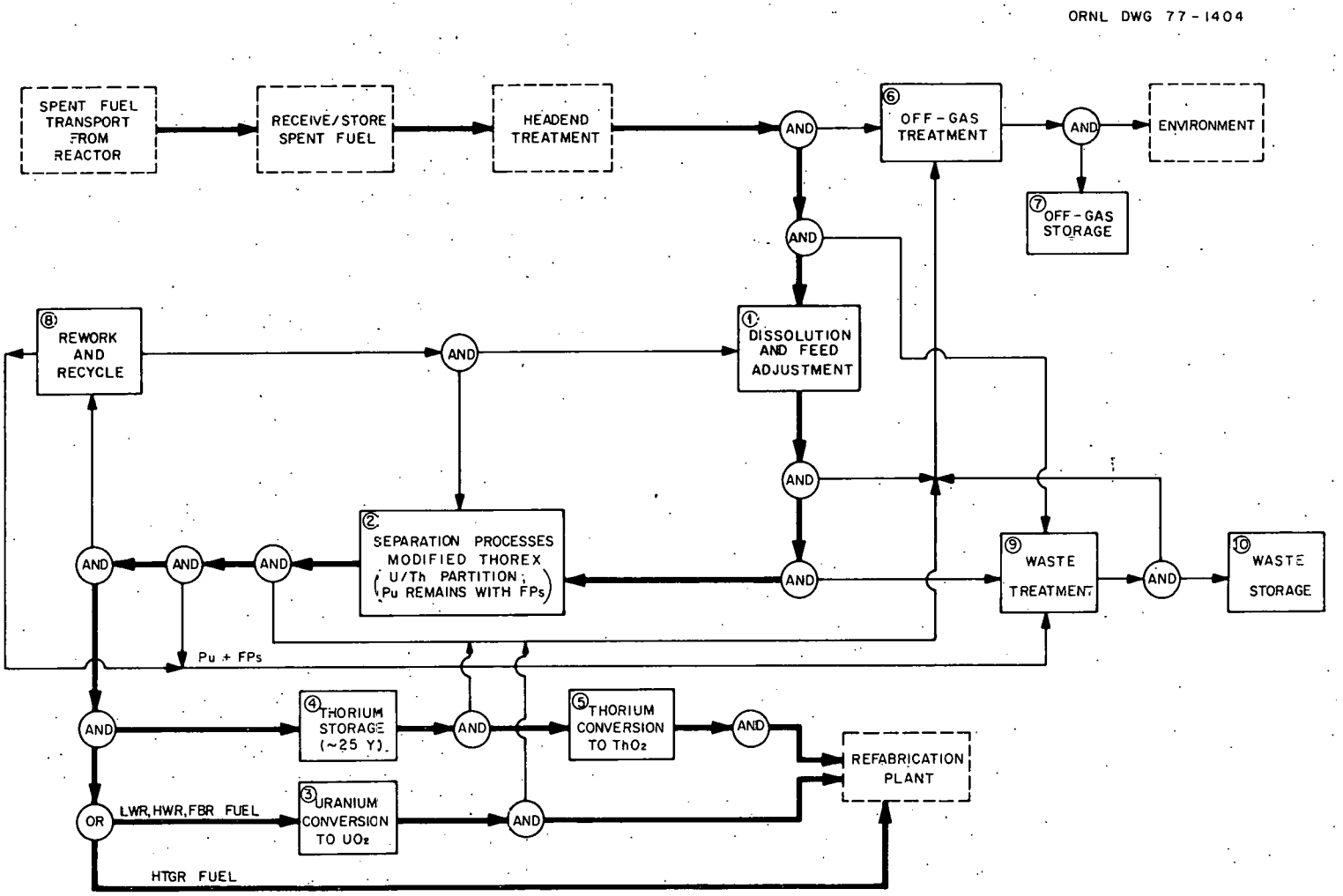


Fig. 4.18. Reprocessing Scheme 11 — Level 1 Functional Flow Diagram: Reprocessing (^{233}U, ^{235}U, ^{238}U)O_2-PuO_2-ThO_2 Fuel or $^{233}UO_2$-PuO_2-ThO_2 Fuel by the Modified Thorex Process. Partition of uranium and thorium. Plutonium stowaway with fission products.

Table 4.18. Analysis for Reprocessing Scheme 11 for <15%-Enriched $^{233}U + ^{235}U$

Feed: $^{233,235,238}UO_2(NO_3)_2$-$Pu(NO_3)_4$-$Th(NO_3)_4$ solution (<15% $^{233,235}U$)

Products: (1) $^{233,235,238}UO_2$ (<15% $^{233,235}U$)
(2) ThO_2
Plutonium remains in the aqueous phase with the fission products

	Process Step and Operation	Development Needed	Material Location	Material Description	Convertibility	Radiation Hazard
1	Dissolution and feed adjustment	Cold engineering	Hot cell	Dissolved fuel material	c	High
2	Separations process	Prototype	Hot cell	$UO_2(NO_3)_2$ and $Th(NO_3)_4$ solutions	a*	Medium
3	Uranium conversion	Hot engineering	Shielded alpha facility	UO_2	a	Medium
4	Thorium product storage	Developed	Shielded tank	$Th(NO_3)_4$ solution	Nonfissionable	Medium
5	Thorium product conversion	Developed	Semiremote facility	ThO_2	Nonfissionable	Low
6	Off-gas treatment	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	Medium
7	Gas storage	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	Medium
8	Rework and recycle	Hot engineering	Remote facility	Radioactive liquid	c to d	Medium to low
9	Waste treatment	Cold engineering	Remote facility	Radioactive liquid	c to d (Pu)	High to low
10	Waste storage	Hot engineering	Remote facility	Radioactive liquid	c	High

*With relatively simple changes in the separations process flowsheet the plutonium could be recovered.

Table 4.19. Analysis for Reprocessing Scheme 11 for >15%-Enriched $^{233}U + ^{235}U$

Feed: $^{233,235,238}UO_2(NO_3)_2$-$Pu(NO_3)_4$-$Th(NO_3)_4$ solution (>15% $^{233,235}U$)

Products: (1) $^{233,235,238}UO_2$ (>15% $^{233,235}U$)
(2) ThO_2

Plutonium remains in the aqueous phase with the fission products

Process Step and Operation		Development Needed	Material Location	Material Description	Convertibility	Radiation Hazard
1	Dissolution and feed adjustment	Cold engineering	Hot cell	Dissolved fuel material	c	High
2	Separations process	Prototype	Hot cell	$UO_2(NO_3)_2$ and $Th(NO_3)_4$ solutions	d*	Medium
3	Uranium conversion	Hot engineering	Shielded alpha facility	UO_2	d†	Medium
4	Thorium product storage	Developed	Shielded tank	$Th(NO_3)_4$	Nonfissionable	Medium
5	Thorium product conversion	Developed	Semiremote facility	ThO_2	Nonfissionable	Low
6	Off-gas treatment	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	Medium
7	Gas storage	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	Medium
8	Rework and recycle	Hot engineering	Remote facility	Radioactive liquid	c to d	Medium to low
9	Waste treatment	Cold engineering	Remote facility	Radioactive liquid	c to d (Pu)	High to low
10	Waste storage	Hot engineering	Remote facility	Radioactive liquid	c	High

*With relatively simple changes in the separations process flowsheet the plutonium could be recovered.

†A 10 kg batch of ^{233}U in equilibrium with the daughters of the ^{232}U it contains would result in a 100 to 1000 R radiation field at one foot distance.

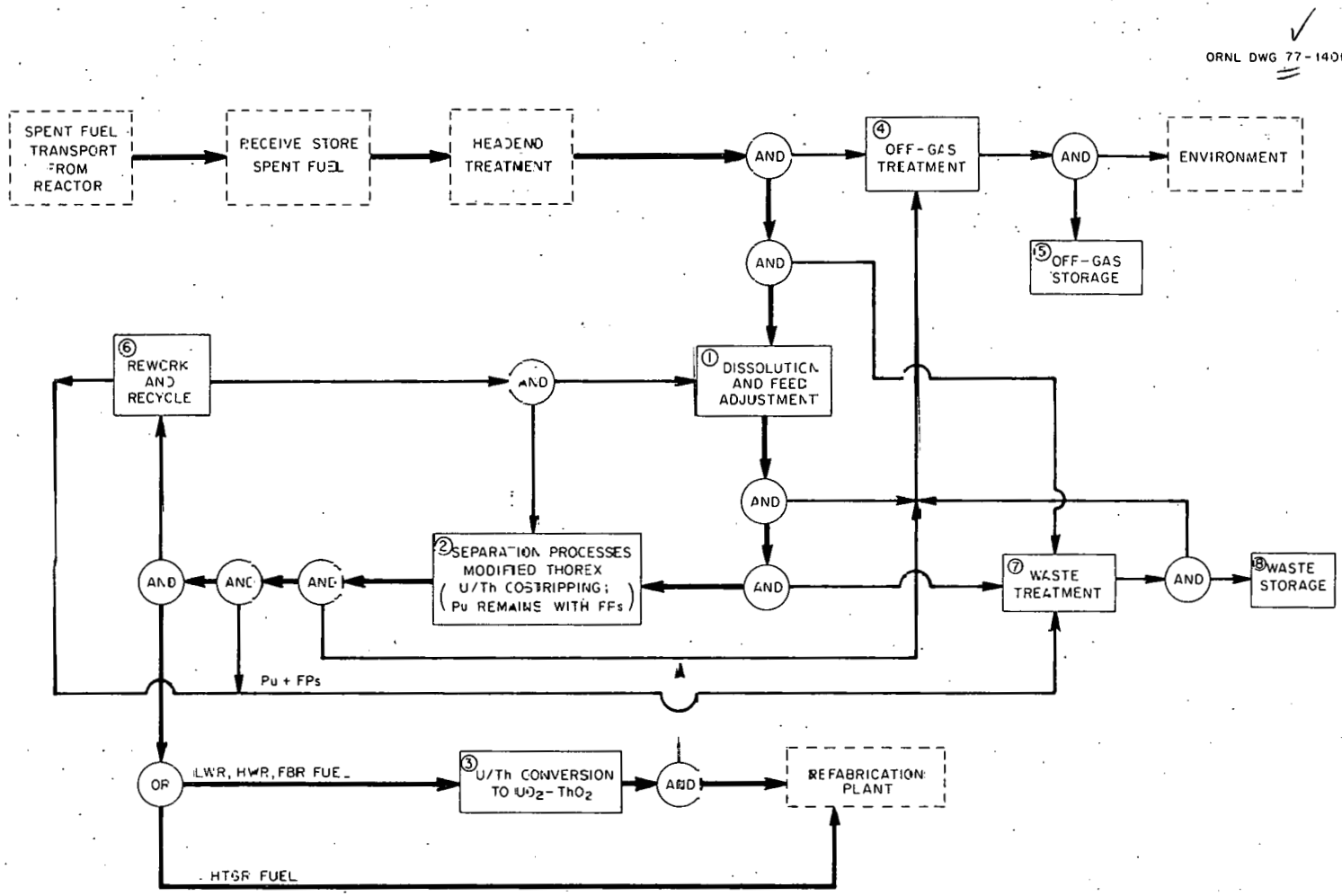


Fig. 4.19. Reprocessing Scheme 12 — Level 1 Functional Flow Diagram: Reprocessing (^{233}U, ^{235}U, ^{238}U)O_2-PuO_2-ThO_2 Fuel or $^{233}UO_2$-PuO_2-ThO_2 Fuel by the Modified Thorex Process. Costripping of Uranium and Thorium. Plutonium stowaway with fission products.

Table 4.20. Analysis for Reprocessing Scheme 12 for <15%-Enriched $^{233}U + ^{235}U$

Feed: $^{233,235,238}UO_2(NO_3)_2$-$Pu(NO_3)_4$-$Th(NO_3)_4$ solution (<15% $^{233,235}U$) Product: (1) $^{233,235,238}UO_2$-ThO_2 (<15% $^{233,235}U$) Plutonium remains with fission products in aqueous waste

Process Step and Operation		Development Needed	Material Location	Material Description	Convertibility	Radiation Hazard
1	Dissolution and feed adjustment	Cold engineering	Hot cell	Dissolved fuel material	c	High
2	Separations process	Prototype	Hot cell	$UO_2(NO_3)_2$-$Th(NO_3)_4$ solution	a*	Medium
3	Product conversion	Hot engineering	Shielded alpha facility	UO_2-ThO_2	a	Medium
4	Off-gas treatment	Hot engineering	Remote facility	Radiaoctive gases	Nonfissionable	Medium
5	Gas storage	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	Medium
6	Rework and recycle	Hot engineering	Remote facility	Radioactive liquid	c to d (Pu)	Medium to low
7	Waste treatment	Cold engineering	Remote facility	Radioactive liquid	c to d (Pu)	High to medium
8	Waste storage	Hot engineering	Remote facility	Radioactive liquid	c	High

*The plutonium could be recovered with the uranium and thorium by use of relatively simple process changes.

Table 4.21. Analysis for Reprocessing Scheme 12 for >15%-Enriched $^{233}U + ^{235}U$

Feed: $^{233,235,238}UO_2(NO_3)_2$-$Pu(NO_3)_4$-$Th(NO_3)_4$ solution (>15% $^{233,235}U$)

Product: (1) $^{233,235,238}UO_2$-ThO_2 (>15% $^{233,235}U$) Plutonium remains with fission products in aqueous waste

Process Step and Operation	Development Needed	Material Location	Material Description	Convertibility	Radiation Hazard
1 Dissolution and feed adjustment	Cold engineering	Hot cell	Dissolved fuel material	c	High
2 Separations process	Prototype	Hot cell	$UO_2(NO_3)_2$-$Th(NO_3)_4$ solution	c*	Medium
3 Product conversion	Cold engineering	Shielded alpha facility	UO_2-ThO_2	c†	Medium
4 Off-gas treatment	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	Medium
5 Gas storage	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	Medium
6 Rework and recycle	Hot engineering	Remote facility	Radioactive liquid	c to d (Pu)	Medium to low
7 Waste treatment	Cold engineering	Remote facility	Radioactive liquid	c to d (Pu)	High to medium
8 Waste storage	Hot engineering	Remote facility	Radioactive liquid	c	High

*The plutonium could be recovered with the uranium and thorium by use of relatively simple process changes.

†A 10 kg batch of ^{233}U in equilibrium with the daughters of the ^{232}U it contains would result in a 100 to 1000 R radiation field at one foot distance.

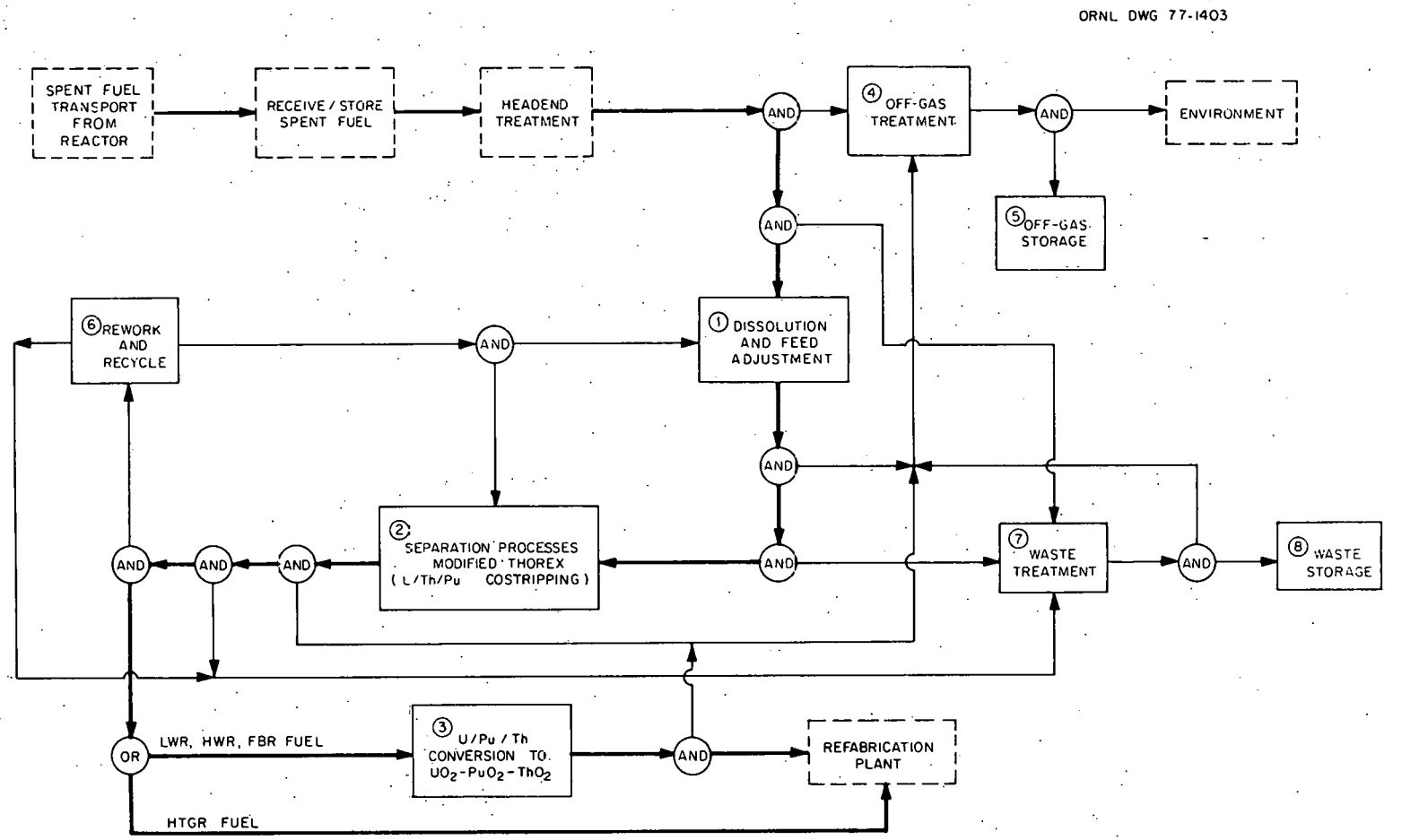


Fig. 4.20. Reprocessing Scheme 13 — Level 1 Functional Flow Diagram: Reprocessing (^{233}U, ^{235}U, ^{238}U)O_2-PuO_2-ThO_2 Fuel and ^{233}UO-PuO_2-ThO_2 Fuel by the Modified Thorex Process. Costripping of Uranium, Plutonium, and Thorium.

Table 4.22. Analysis for Reprocessing Scheme 13 for <15%-Enriched $^{233}U + ^{235}U$

Feed: $^{233,235,238}UO_2(NO_3)_2$-$Pu(NO_3)_4$-$Th(NO_3)_4$ solution (<15% $^{233,235}U$) Product: (1) $^{233,235,238}UO_2$-PuO_2-ThO_2 (mixed oxide)

Process Step and Operation		Development Needed	Material Location	Material Description	Convertibility	Radiation Hazard
1	Dissolution and feed adjustment	Hot engineering	Hot cell	Dissolved fuel material	c	High
2	Separations process	Prototype	Hot cell	$UO_2(NO_3)_2$-$Pu(NO_3)_4$ and $Th(NO_3)_4$ mixed sclution	c (Pu)*	Medium
3	Product conversion	Cold engineering	Shielded alpha facility	UO_2-PuO_2-ThO_2	c (Pu)*	Medium
4	Off-gas treatment	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	Medium
5	Gas storage	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	Medium
6	Rework and recycle	Hot engineering	Remote facility	Radioactive liquid	c to d (Pu)	Medium to low
7	Waste treatment	Cold engineering	Remote facility	Radioactive liquid	c to d (Pu)	High to medium
8	Waste storage	Hot engineering	Remote facility	Radioactive liquid	c	High

*The plutonium could be separated from the other components by use of relatively simple process changes.

Table 4.23. Analysis for Reprocessing Scheme 13 for >15%-Enriched $^{233,235}U$

Feed: $^{233,235,238}UO_2(NO_3)_2$-$Pu(NO_3)_4$-$Th(NO_3)_4$ solution Product: (1) $^{233,235,238}UO_2$-PuO_2-ThO_2 (mixed oxides) (>15% $^{233,235}U$)

Process Step and Operation		Development Needed	Material Location	Material Description	Convertibility	Radiation Hazard
1	Dissolution and feed adjustment	Cold engineering	Hot cell	Dissolved fuel material	c	High
2	Separations process	Prototype	Hot cell	$UO_2(NO_3)_2$-$Pu(NO_3)_4$-$Th(NO_3)_4$ mixed solutions	c (Pu)*	Medium
3	Product conversion	Cold laboratory	Shielded alpha facility	UO_2-PuO_2-ThO_2	c (Pu)*	Medium
4	Off-gas treatment	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	Medium
5	Gas storage	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	Medium
6	Rework and recycle	Hot engineering	Remote facility	Radioactive liquid	c to d	Medium to low
7	Waste treatment	Cold engineering	Remote facility	Radioactive liquid	c to d	High to medium
8	Waste storage	Hot engineering	Remote facility	Radioactive liquid	c	High

*The plutonium could be separated from the other components by use of relatively simple process changes.

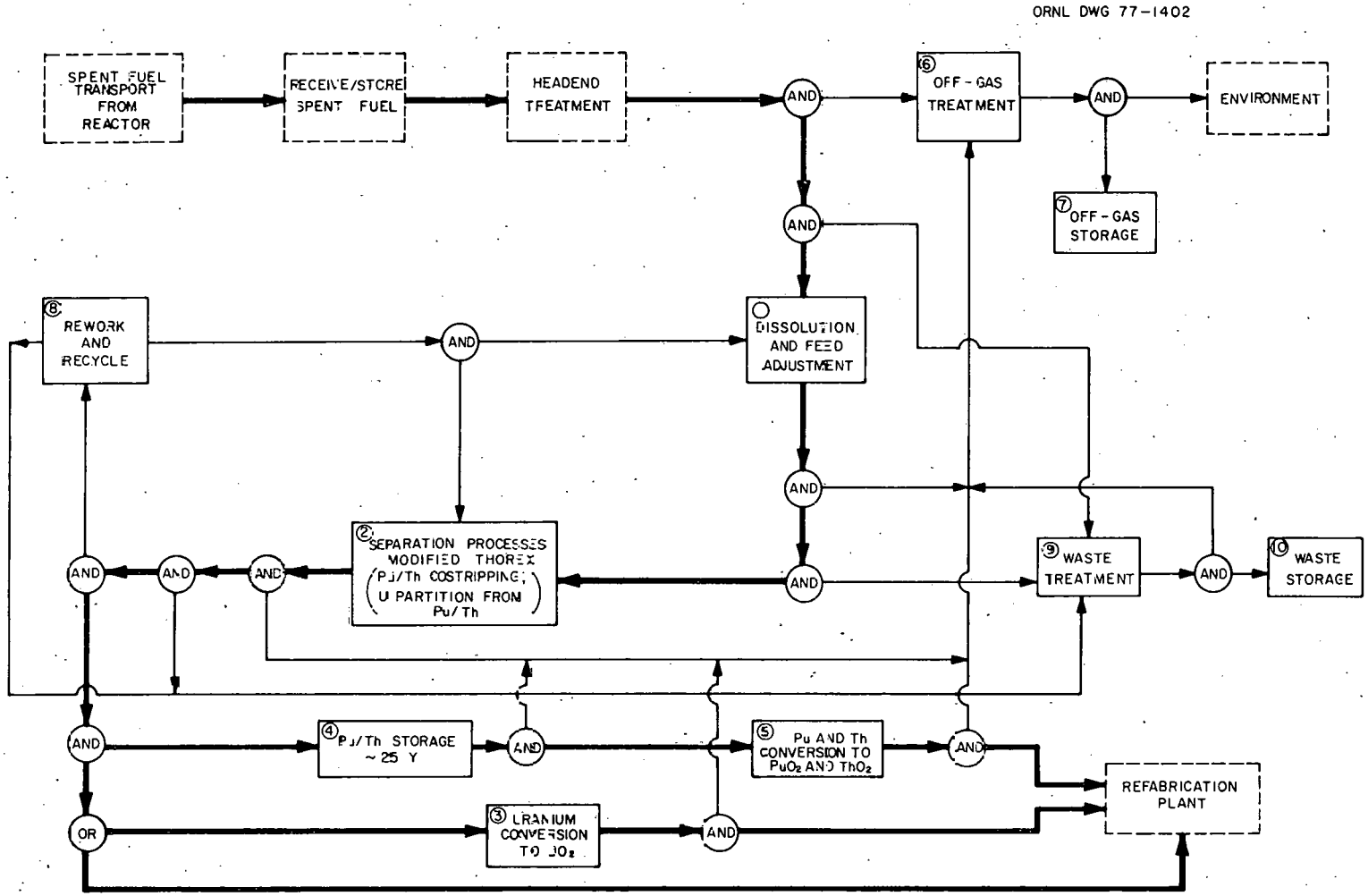


Fig. 4.21. Reprocessing Scheme 14 — Level 1 Functional Flow Diagram: Reprocessing (^{233}U, ^{235}U, ^{238}U)O_2-PuO_2-ThO_2 Fuel and $^{233}UO_2$-PuO_2-ThO_2 Fuel by the Modified Thorex Process. Costripping of plutonium and thorium. Partition of uranium from plutonium-thorium mixture.

Table 4.24. Analysis for Reprocessing Scheme 14 for <15%-Enriched $^{233}U + ^{235}U$

Feed: $^{233,235,238}UO_2(NO_3)_2$-$Pu(NC_3)_4$-$Th(NO_3)_4$ solution

Products: (1) $^{233,235,238}UO_2$ (<15% $^{233,235}U$)
(2) PuO_2-ThO_2 (mixed oxide)

Process Step and Operation		Development Needed	Material Location	Material Description	Convertibility	Radiation Hazard
1	Dissolution and feed adjustment	Cold engineering	Hot cell	Dissolved fuel material	c	High
2	Separations process	Prototype	Hot cell	$UO_2(NO_3)_2$ and $Pu(NO_3)_4$-$Th(NO_3)_4$ solutions	c	Medium
3	Uranium conversion	Cold engineering	Shielded alpha facility	UO_2	a	Medium
4	Thorium-Plutonium product storage	Hot engineering	Shielded tank	$Pu(NO_3)_4$-$Th(NO_3)_4$ solution	c*	Medium
5	Thorium-Plutonium product conversion	Hot engineering	Semiremote facility	PuO_2-ThO_2	d	Medium
6	Off-gas treatment	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	Medium
7	Gas storage	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	Medium
8	Rework and recycle	Hot engineering	Remote facility	Radioactive liquid	c to d	Medium to low
9	Waste treatment	Cold engineering	Remote facility	Radioactive liquid	c to d	High to medium
10	Waste storage	Hot engineering	Remote facility	Radioactive liquid	c	High

*The plutonium could be separated from the thorium by use of relatively simple process changes.

Table 4.25. Analysis for Reprocessing Scheme 14 for >15%-Enriched $^{233}U + ^{235}U$

Feed: $^{233,235,238}UO_2(NO_3)_2$-$Pu(NO_3)_4$-$Th(NO_3)_4$ solution

Product: (1) $^{233,235,238}UO_2$ (>15% $^{233,235}U$)
(2) PuO_2-ThO_2 (mixed oxide)

Process Step and Operation		Development Needed	Material Location	Material Description	Convertibility	Radiation Hazard
1	Dissolution and feed adjustment	Cold engineering	Hot cell	Dissolved fuel material	c	High
2	Separations process	Prototype	Hot cell	$UO_2(NO_3)_2$-$Pu(NO_3)_4$-$Th(NO_3)_4$ solutions	c*	Medium
3	Uranium conversion	Cold engineering	Shielded alpha facility	UO_2	d†	Medium
4	Thorium-Plutonium product storage	Hot engineering	Shielded tank	$Pu(NO_3)_4$-$Th(NO_3)_4$ solution	c*	Medium
5	Thorium-Plutonium product conversion	Hot laboratory	Semiremote facility	PuO_2-ThO_2	d	Medium
6	Off-gas treatment	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	Medium
7	Gas storage	Hot engineering	Remote facility	Radioactive gases	Nonfissionable	Medium
8	Rework and recycle	Hot engineering	Remote facility	Radioactive liquid	c to d	Medium to low
9	Waste treatment	Cold engineering	Remote facility	Radioactive liquid	c to d	High to medium
10	Waste storage	Hot engineering	Remote facility	Radioactive liquid	c	High

*The plutonium could be separated from the thorium by use of relatively simple process changes.

†A 10 kg batch of ^{233}U in equilibrium with the ^{232}U it contains would result in a 100 to 1000 R radiation field at one foot distance.

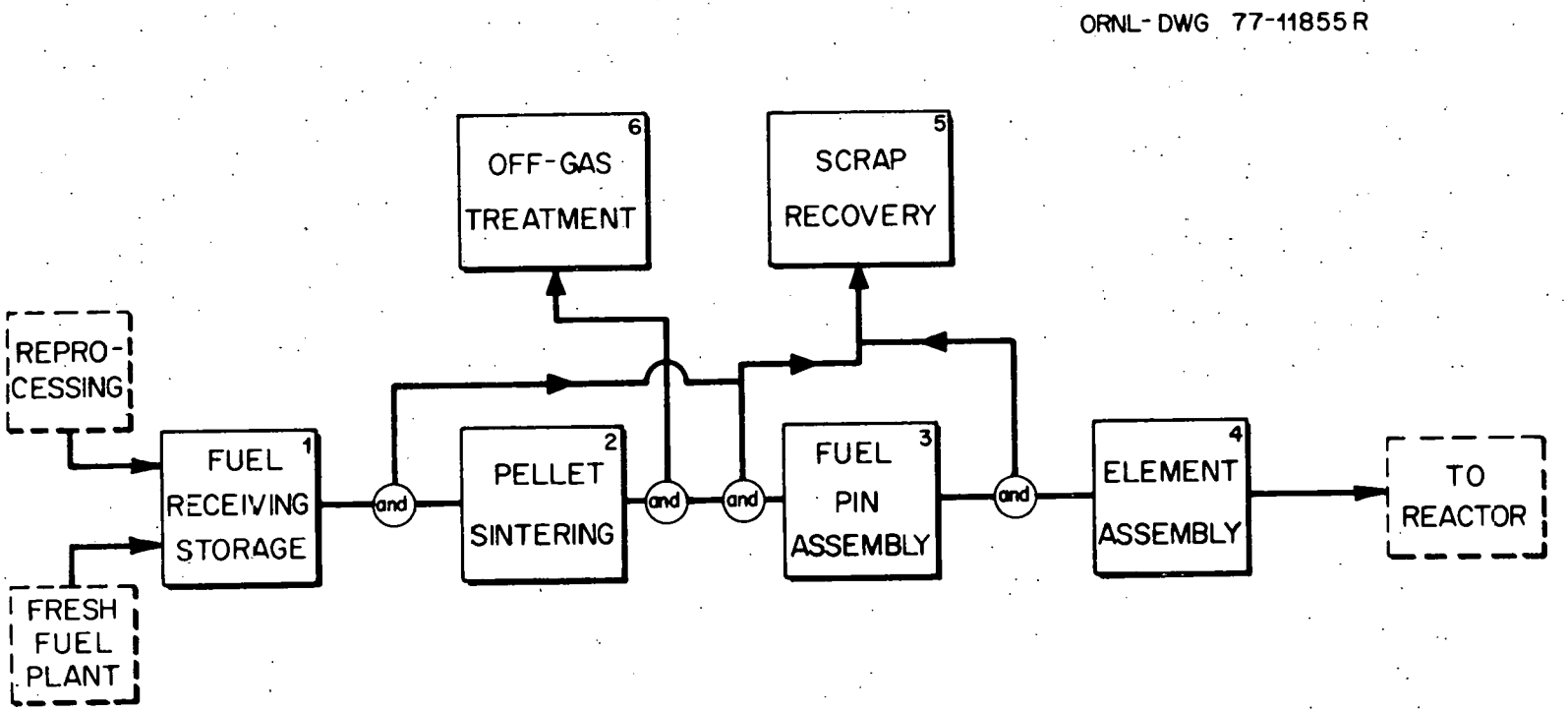


Fig. 4.22. Diagram 1, Level 1 Functional Flow Diagram: LWR-HWR-FBR Fuel Element Fabrication.

Table 4.26. Analysis of Fuel Refabrication Method 1, Flow Diagram 1

Feed: $^{233,235,238}UC_2$ Powder
PuO_2 Powder
ThO_2 Powder

Product: $(U,Pu,Th)O_2$ Pellets Welded into Fuel Pins

Process Step	Operation	Development Needed	Material Location	Material Description	Uranium Enrichment	Fissile Concentration	Convertability	Radiation Hazard
1	Fuel receiving storage	Cold prototype	Shipping cask and hot cell	UO_2 powder	LEU	Low	a	Medium
				PuO_2 powder		High	d	Low
				ThO_2 powder			Non-fissionable	Low
2	Pellet sintering	Hot laboratory	Hot cell	UO_2 makeup	MEU	Medium	a (^{233}U) or b (^{235}U)	Medium (^{233}U) or low (^{235}U)
3	Fuel pin assembly	Cold engineering	Hot cell	Sintered $(U,Pu,Th)O_2$ pellets		Low	c	Medium
4	Element assembly	Cold engineering	Hot cell and shipping cask	Welded fuel pins			c	Medium
5	Scrap recovery	Hot laboratory	Hot cell	Miscellaneous off-spec material			d	Low
6	Off-gas treatment	Hot engineering	Hot cell	Sintering furnace effluent			Non-fissionable	Low

Table 4.27. Analysis of Fuel Refabrication Method 2, Flow Diagram 1

Feed: Mixed Oxides of $^{233,235,238}U$, Pu, and Th

Product: Sintered Pellets of $(U,Pu,Th)O_2$ Welded into Fuel Pins

Process Step	Operation	Development Needed	Material Location	Material Description	Uranium Enrichment	Fissile Concentration	Convertability	Radiation Hazard
1	Fuel receiving storage	Cold prototype	Shipping cask and hot cell	$(U,Pu,Th)O_2$ powder	LEU	Low	c	Medium
2	Pellet sintering	Hot laboratory	Hot cell	UO_2 makeup	MEU	Medium	a (^{233}U) or b (^{235}U)	Medium (^{233}U) or Low (^{235}U)
3	Fuel pin assembly	Cold engineering	Hot cell	Sintered $(U,Pu,Th)O_2$ pellets			c	Medium
4	Element assembly	Cold engineering	Hot cell and shipping cask	Welded fuel pins			c	Medium
5	Scrap recovery	Hot laboratory	Hot cell	Miscellaneous off-spec material			c	Medium
6	Off-Gas Treatment	Hot engineering	Hot cell	Sintering furnace effluent			Non-fissionable	Low

Table 4.28. Analysis of Fuel Refabrication Method 3, Flow Diagram 1

Feed: $^{233,235,238}UO_2$ and $(Pu,Th)O_2$

Product: $(U,Pu,Th)O_2$ Sintered Pellets Welded in Fuel Pins

Process Step	Operation	Development Needed	Material Location	Material Description	Uranium Enrichment	Fissile Concentration	Convertability	Radiation Hazard
1	Fuel receiving	Cold prototype	Shipping container and hot cell	$^{233,235,238}UO_2$	LEU	Low	a	Medium
				$(Pu,Th)O_2$		Low	d	Low
2	Pellet sintering	Hot laboratory	Hot cell	Makeup UO_2	MEU	Medium	a (^{233}U) or b (^{235}U)	Medium (^{233}U) or Low (^{235}U)
3	Fuel pin assembly	Cold engineering	Hot cell	$(U,Pu,Th)O_2$ sintered pellets		Low	a	Medium
4	Element assembly	Cold engineering	Hot cell and shipping cask	Welded fuel pins		Low	a	Medium
5	Scrap recovery	Hot Laboratory	Hot cell	Miscellaneous off-spec material			d (Pu,Th)	Low
6	Off-gas treatment	Hot engineering	Hot cell	Sintering furnace effluent			Non-fissionable	Low

Table 4.29. Analysis of Fuel Refabrication Method 4, Flow Diagram 1

Feed: PuO_2 Powder
$^{238}UO_2$ Powder

Product: (Pu,^{238}U)O_2 Pellets Welded into Fuel Pins

Process Step	Operation	Development Needed	Material Location	Material Description	Uranium Enrichment	Fissile Concentration	Convertability	Radiation Hazard
1	Fuel receiving storage	Developed	Powder sealed in cans	PuO_2 powder		High	d	Low
2	Pellet sintering	Developed	Glove box	$^{238}UO_2$ powder	Low	Low	b	Negligible
3	Fuel pin assembly	Developed	Glove box	(Pu,U)O_2 pellets		Low (LWR) Medium (LMFBR)	d	Low
4	Element assembly	Developed	Hands-on	Welded fuel pins			d	Low
5	Scrap recovery	Developed	Glove box	Miscellaneous off-spec material			d	Low
6	Off-gas treatment	Developed		Sintering furnace effluent			Non-fissionable	Low

Table 4.30. Analysis of Fuel Refabrication Method 5, Flow Diagram 1

Feed: $^{233}UO_2$ Powder
ThO_2 Powder

Product: $(U,Th)O_2$ Pellets Welded into Fuel Pins

Process Step	Operation	Development Needed	Material Location	Material Description	Uranium Enrichment	Fissile Concentration	Convertability	Radiation Hazard
1	Fuel receiving storage	Cold prototype	Shipping cask and hot cell	$^{233}UO_2$ powder	HEU	High	d	Medium
2	Fuel pellet sintering	Cold engineering	Hot cell	ThO_2 powder			Non-fissionable	Low
3	Fuel pin assembly	Cold engineering	Hot cell	$(U,Th)O_2$ pellets		Low (LWR) Medium (LMFBR)	c	Medium
4	Element assembly	Cold engineering	Hot cell and shipping cask	Welded fuel pins			c	Medium
5	Scrap recovery	Hot laboratory	Hot cell	Miscellaneous off-spec material			d	Medium
6	Off-gas treatment	Hot engineering	Hot cell	Sintering furnace effluent			Non-fissionable	Low

Table 4.31. Analysis of Fuel Refabrication Method 6, Flow Diagram 1

Feed: $(^{233}U,Th)O_2$ Powder

Product: $(^{233}U,Th)O_2$ Pellets Welded into Fuel Pins

Process Step	Operation	Development Needed	Material Location	Material Description	Uranium Enrichment	Fissile Concentration	Convertability	Radiation Hazard
1	Fuel receiving storage	Cold prototype	Shipping cask and hot cell	$(^{233}U,Th)O_2$ powder	HEU	Low	c	Medium
2	Pellet sintering	Cold engineering	Hot cell	UO_2 powder for makeup	MEU	Medium	a (^{233}U) or b (^{235}U)	Medium (^{233}U) or Low (^{235}U)
3	Fuel pin assembly	Cold engineering	Hot cell	$(U,Th)O_2$ pellets		Low	c	Medium
4	Element assembly	Cold engineering	Hot cell and shipping cask	Welded fuel pins			c	Medium
5	Scrap recovery	Cold laboratory	Hot cell	Miscellaneous off-spec material			c	Medium
6	Off-gas treatment	Hot engineering	Hot cell	Sintering furnace effluent			Non-fissionable	Low

Table 4.32. Analysis of Fuel Refabrication Method 7, Flow Diagram 1

Feed: $(^{235,238}U,Pu)C_2$ Powder

Product: $(U,Pu)O_2$ Pellets Welded into Fuel Pins

Process Step	Operation	Development Needed	Material Location	Material Description	Uranium Enrichment	Fissile Concentration	Convertability	Radiation Hazard
1	Fuel receiving storage	Developed	Powder sealed in cans	$(U,Pu)O_2$ powder	LEU	Low	d	Low
2	Pellet sintering	Developed	Glove box	$^{235}UO_2$ powder for makeup	MEU	Medium	b	Low
3	Fuel pin assembly	Developed	Glove box	$(U,Pu)O_2$ pellets		Low	d	Low
4	Element assembly	Developed	Contact facility	Welded fuel pins			d	Low
5	Scrap recovery	Developed	Glove box	Miscellaneous off-spec material			d	Low
6	Off-gas treatment	Developed	Glove box	Sintering furnace effluent			Non-fissionable	Low

Table 4.33. Analysis of Fuel Refabrication Method 8, Flow Diagram 1

Feed: $^{233,235,238}UO_2$ Powder

Product: UO_2 Pellets Welded into Fuel Pins

Process Step	Operation	Development Needed	Material Location	Material Description	Uranium Enrichment	Fissile Concentration	Convertability	Radiation Hazard
1	Fuel receiving storage	Cold prototype	Shipping cask and hot cell	UO_2 powder UO_2 makeup	LEU MEU	Low Medium	a a (^{233}U) or b (^{235}U)	Medium Medium (^{233}U) or low (^{235}U)
2	Pellet sintering	Cold engineering	Hot cell					
3	Fuel pin assembly	Cold engineering	Hot cell	Sintered UO_2 pellets	LEU	Low	a	Medium
4	Element assembly	Cold engineering	Hot cell and shipping cask	Welded fuel pins			a	Medium
5	Scrap recovery	Cold engineering	Hot cell	Miscellaneous off-spec material			a or b	Medium or low
6	Off-gas treatment	Hot engineering	Hot cell	Sintering furnace effluent			Non-fissionable	Low

Table 4.34. Analysis of Fuel Refabrication Method 9, Flow Diagram 1

Feed: PuO_2 Powder
ThO_2 Powder

Product: $(Pu,Th)O_2$ Pellets Welded into Fuel Pins

Process Step	Operation	Development Needed	Material Location	Material Description	Uranium Enrichment	Fissile Concentration	Convertability	Radiation Hazard
1	Fuel receiving storage	Developed	Powder sealed in cans	PuO_2 powder		High	d	Low
2	Pellet sintering	Hot laboratory	Glove box	ThO_2 powder			Non-fissionable	Low
3	Fuel pin assembly	Developed	Glove box	$(Pu,Th)O_2$ pellets		Low	d	Low
4	Element assembly	Developed	Contact facility	Welded fuel pins		Low	d	Low
5	Scrap recovery	Hot laboratory	Glove box	Miscellaneous off-spec material			d	Low
6	Off-gas treatment	Developed	Glove box	Sintering furnace effluent			Non-fissionable	Low

Table 4.35. Analysis of Fuel Refabrication Method 10, Flow Diagram 1

Feed: $(Pu, Th)O_2$ Powder

Product: $(Pu,Th)O_2$ Pellets Welded into Fuel Pins

Process Step	Operation	Development Needed	Material Location	Material Description	Uranium Enrichment	Fissile Concentration	Convertability	Radiation Hazard
1	Fuel receiving storage	Developed	Powder sealed in cans	$(Pu,Th)O_2$ powder		Low	d	Low
2	Pellet sintering	Hot laboratory	Glove box	PuO_2 makeup		High	d	Low
3	Fuel pin assembly	Developed	Glove box	$(Pu,Th)O_2$ pellets		Low	d	Low
4	Element assembly	Developed	Contact facility	Welded fuel pins		Low	d	Low
5	Scrap recovery	Hot laboratory	Glove box	Miscellaneous off-spec material			d	Low
6	Off-gas treatment	Developed	Glove box	Sintering furnace effluent			Non-fissionable	Low

Table 4.36. Analysis of Fuel Refabrication Method 11, Flow Diagram 1

Feed: $^{233,235,238}UO_2$ Powder
ThO_2 Powder

Product: $(U,Th)O_2$ Pellets Welded into Fuel Pins

Process Step	Operation	Development Needed	Material Location	Material Description	Uranium Enrichment	Fissile Concentration	Convertability	Radiation Hazard
1	Fuel receiving storage	Cold prototype	Shipping cask and hot cell	UO_2 powder	LEU	Low	a	Medium
				ThO_2 powder			Non-fissionable	Low
2	Fuel pellet sintering	Cold engineering	Hot cell	UO_2 makeup	MEU	Medium	a (^{233}U) or b (^{235}U)	Medium (^{233}U) or low (^{235}U)
3	Fuel pin assembly	Cold engineering	Hot cell	Sintered $(U,Th)O_2$ pellets	LEU	Low	a	Medium
4	Element assembly	Cold engineering	Hot cell and shipping cask	Welded fuel pins			a	Medium
5	Scrap recovery	Cold laboratory	Hot cell	Miscellaneous off-spec material			a or b	Medium or low
6	Off-gas treatment	Hot engineering	Hot cell	Sintering furnace effluent			Non-fissionable	Low

Table 4.37. Analysis of Fuel Refabrication Method 12, Flow Diagram 1

Feed: $(^{233,235,238}U,Th)O_2$ Powder

Product: $(U,Th)O_2$ Pellets Welded into Fuel Pins

Process Step	Operation	Development Needed	Material Location	Material Description	Uranium Enrichment	Fissile Concentration	Convertability	Radiation Hazard
1	Fuel receiving storage	Cold prototype	Shipping cask and hot cell	$(U,Th)O_2$ powder	LEU	Low	a	Medium
2	Pellet sintering	Cold engineering	Hot cell	UO_2 makeup	MEU	Medium	a (^{233}U) or b (^{235}U)	Medium or low
3	Fuel pin assembly	Cold engineering	Hot cell	Sintered $(U,Th)O_2$ pellets	LEU	Low	a	Medium
4	Element assembly	Cold engineering	Hot cell and shipping cask	Welded fuel pins			a	Medium
5	Scrap recovery	Cold laboratory	Hot cell	Miscellaneous off-spec material			a or b	Medium or low
6	Off-gas treatment	Hot engineering	Hot cell	Sintering furnace effluent			Non-fissionable	Low

Table 4.38. Analysis of Fuel Refabrication Method 13, Flow Diagram 1

Feed: $^{233}UO_2$ Powder, $^{238}UO_2$ Powder

Product: UO_2 Pellets Welded into Fuel Pins

Process Step	Operation	Development Needed	Material Location	Material Description	Uranium Enrichment	Fissile Concentration	Convertability	Radiation Hazard
1	Fuel receiving storage	Cold prototype	Shipping cask and hot cell	$^{233}UO_2$ powder	HEU	High	d	Medium
				$^{238}UO_2$ powder	LEU	Low	b	Low
2	Fuel pellet sintering	Cold engineering	Hot cell					
3	Fuel pin assembly	Cold engineering	Hot cell	UO_2 pellets	LEU	Low (LWR) Medium (LMFBR)	a	Medium
4	Element assembly	Cold engineering	Hot cell and shipping cask	Welded fuel pins			a	Medium
5	Scrap recovery	Hot laboratory	Hot cell	Miscellaneous off-spec material			d	Medium
6	Off-gas treatment	Hot engineering	Hot cell	Sintering furnace effluent			Non-fissionable	Low

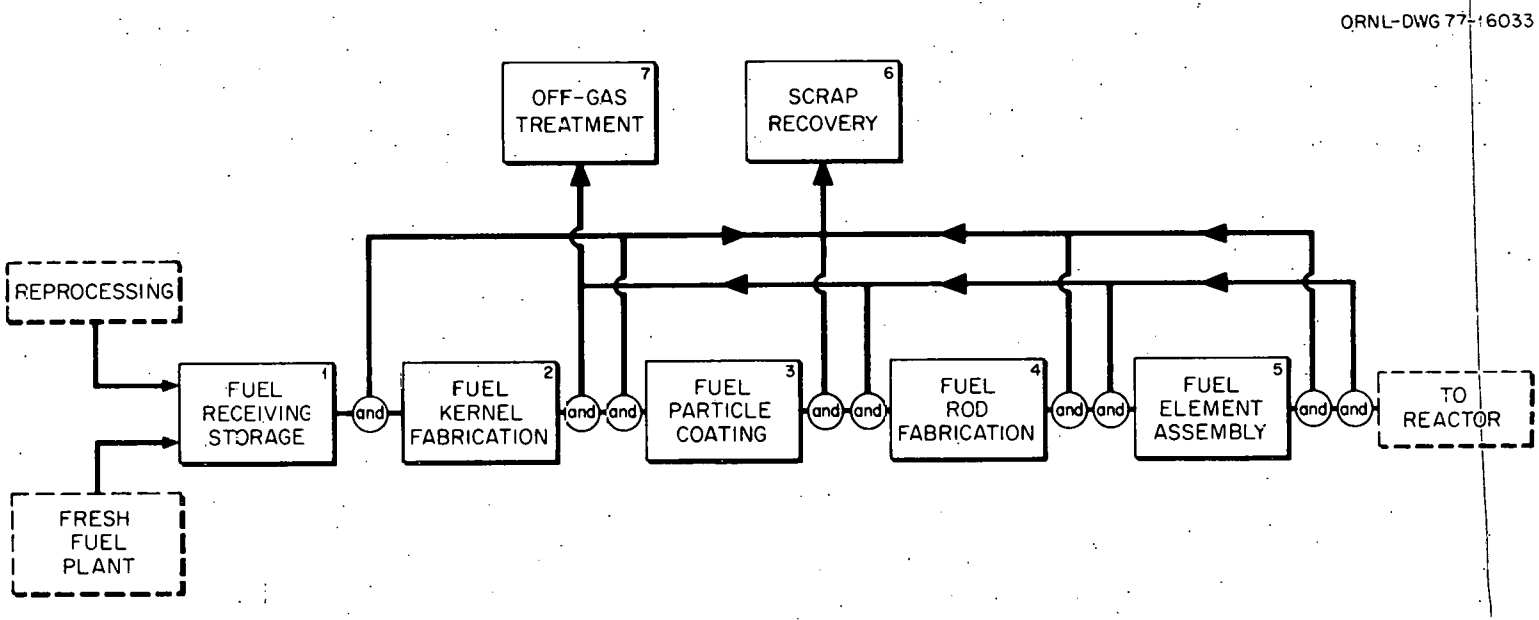


Fig. 4.23. Diagram 2. Level 1 Functional Flow Diagram: HTGR Fuel Element Fabrication.

Table 4.39. Analysis of Fuel Fabrication Method 14, Flow Diagram 2

Feed: $Pu(NO_3)_4$ Solution
$^{235,238}UO_2(NO_3)_2$ Solution

Product: Coated $(U,Pu)O_2$ Fuel Particles Within a Graphite Fuel Element

Process Step	Operation	Development Needed	Material Location	Material Description	Uranium Enrichment	Fissile Concentration	Convertability	Radiation Hazard
1	Fuel receiving storage	Hot engineering	Glove box	$Pu(NO_3)_4$ solution		High	d	Low
2	Fuel kernel fabrication	Hot engineering	Glove box	$UO_2(NO_3)_2$ solution	LEU	Low	b	Low
3	Fuel particle coating	Hot engineering	Glove box	$(U,Pu)O_2$ fuel kernels	LEU	Medium	d	Low
4	Fuel rod fabrication	Cold prototype	Glove box	Coated fuel particles			d	Low
5	Fuel element assembly	Cold prototype	Glove box	Pitch bonded fuel rods			d	Low
6	Scrap recovery	Hot engineering	Glove box	Miscellaneous off-spec material			d	Low
7	Off-gas treatment	Developed	Glove box	Radioactive gases			Non-fissionable	Low

Table 4.40. Analysis of Fuel Fabrication Method 15, Flow Diagram 2

Feed: $(^{233}UO_2)(NO_3)_2$ Product: Coated UO_2 Fuel Particles Within a Graphite Fuel Element

Process Step	Operation	Development Needed	Material Location	Material Description	Uranium Enrichment	Fissile Concentration	Convertability	Radiation Hazard
1	Fuel receiving storage	Cold prototype and hot engineering	Hot cell	$(^{233}UO_2)(NO_3)_2$	HEU	High	d	Medium
2	Fuel kernel fabrication	Cold prototype and hot engineering	Hot cell	$(^{238}UO_2)(NO_3)_2$	LEU	Low	b	Low
3	Fuel particle coating	Cold prototype	Hot cell	UC_2 microspheres	MEU	Medium	a	Medium
4	Fuel rod fabrication	Cold prototype	Hot cell	Coated microspheres			a	Medium
5	Fuel element assembly	Cold prototype	Hot cell	Pitch-blended fuel rods			a	Medium
6	Scrap recovery	Hot laboratory	Hot cell	Miscellaneous off-spec material			d	Medium
7	Off-gas treatment	Hot engineering	Hot cell	Radioactive gases			Non-fissionable	Low

Table 4.41. Analysis of Fuel Fabrication Method 16, Flow Diagram 2

Feed: Mixed Solutions of Th and ^{233}U nitrate

Product: Coated $(U,Th)O_2$ Fuel Particles Within a Graphite Fuel Element

Process Step	Operation	Development Needed	Material Location	Material Description	Uranium Enrichment	Fissile Concentration	Convertability	Radiation Hazard
1	Fuel receiving storage	Cold prototype	Hot cell	$x(UO_2)(NO_3)_2 \cdot y Th(NO_3)_4$ Aqueous solution	HEU	Medium	c	Medium
2	Fuel kernel fabrication	Cold laboratory	Hot cell					
3	Fuel particle coating	Cold prototype	Hot cell	$(U,Th)O_2$ fuel microspheres	HEU	Medium	c	Medium
4	Fuel rod fabrication	Cold prototype	Hot cell	Coated fuel particles			c	Medium
5	Fuel element assembly	Cold prototype	Hot cell	Pitch bonded fuel rods			c	Medium
6	Scrap recovery	Hot laboratory	Hot cell	Miscellaneous off-spec material			c	Medium
7	Off-gas treatment	Hot engineering	Hot cell	Radioactive gases			Non-fissionable	Low

Table 4.42. Analysis of Fuel Fabrication Method 17, Flow Diagram 2

Feed: Mixed Nitrate Solutions of $^{235,238}U$ and Pu

Product: Coated $(U,Pu)O_2$ Microspheres Within a Graphite Fuel Element

Process Step	Operation	Development Needed	Material Location	Material Description	Uranium Enrichment	Fissile Concentration	Convertability	Radiation Hazard
1	Fuel receiving storage	Developed	Glove box	$(UO_2)(NO_3)_2$ — $Pu(NO_3)_4$ solution	MEU	Medium	d	Low
2	Fuel kernel fabrication	Hot laboratory	Glove box					
3	Fuel particle coating	Hot engineering	Glove box	(U,Pu)O microspheres	MEU	Medium	d	Low
4	Fuel rod fabrication	Cold prototype	Glove box	Coated microspheres			d	Low
5	Fuel element assembly	Cold prototype	Glove box	Pitch bonded fuel rods			d	Low
6	Scrap recovery	Hot engineering	Glove box	Miscellaneous off-spec material			d	Low
7	Off-gas treatment	Developed	Glove box	Radioactive gases			Non-fissionable	Low

Table 4.43. Analysis of Fuel Fabrication Method 18, Flow Diagram 2

Feed: $Pu(NO_3)_4$ Solution

Product: Coated PuO_2 Microspheres Combined with Coated ThO_2 Microspheres in a Graphite Fuel Element

Process Step	Operation	Development Needed	Material Location	Material Description	Uranium Enrichment	Fissile Concentration	Convertability	Radiation Hazard
1	Fuel receiving storage	Developed	Glove box	$Pu(NO_3)_4$ solution		High	d	Low
2	Fuel kernel fabrication	Hot laboratory	Glove box					
3	Fuel particle coating	Hot engineering	Glove box	PuO_2 microspheres		High	d	Low
4	Fuel rod fabrication	Cold prototype	Glove box	Coated PuO_2 microspheres			d	Low
5	Fuel element assembly	Cold prototype	Glove box	Pitch bonded fuel rods			d	Low
6	Scrap recovery	Hot laboratory	Glove box	Miscellaneous off-spec material			d	Low
7	Off-gas treatment	Developed	Glove box	Radioactive gases			Non-fissionable	Low

Table 4.44. Analysis of Fuel Fabrication Method 19, Flow Diagram 2

Feed: $^{233}UO_2(NO_3)_2$ Solutions, HEU and LEU

Product: Coated UO_2 Microspheres in a Graphite Fuel Element

Process Step	Operation	Development Needed	Material Location	Material Description	Uranium Enrichment	Fissile Concentration	Convertability	Radiation Hazard
1	Fuel receiving storage	Cold prototype	Hot cell	$^{233}UO_2(NO_3)_2$	HEU	High	d	Medium
2	Fuel kernel fabrication	Cold engineering	Hot cell	$^{233}UO_2(NO_3)_2$	LEU	Low	a	Medium
3	Fuel particle coating	Cold prototype	Hot cell	UO_2 microspheres	MEU	Medium	a	Medium
4	Fuel rod fabrication	Cold prototype	Hot cell	Coated UO_2 microspheres			a	Medium
5	Fuel element assembly	Cold prototype	Hot cell	Pitch bonded fuel rods			a	Medium
6	Scrap recovery	Hot laboratory	Hot cell	Miscellaneous off-spec material			d	Medium
7	Off-gas treatment	Hot engineering	Hot cell	Radioactive gases			Non-fissionable	Low

Table 4.45. Analysis of Fuel Fabrication Method 20, Flow Diagram 2

Feed: $Pu(NO_3)_4 - Th(NC_3)_4$ Blended Solutions

Product: Coated $(Pu,Th)O_2$ Microspheres in a Graphite Fuel Element

Process Step	Operation	Development Needed	Material Location	Material Description	Uranium Enrichment	Fissile Concentration	Convertability	Radiation Hazard
1	Fuel received storage	Developed	Glove box	$Pu(NO_3)_4 - Th(NO_3)_4$		Medium — High	d	Low
2	Fuel kernel fabrication	Hot laboratory	Glove box					
3	Fuel particle coating	Hot engineering	Glove box	$(Pu,Th)O_2$ microspheres		Medium — High	d	Low
4	Fuel rod fabrication	Cold prototype	Glove box	Coated fuel microspheres			d	Low
5	Fuel element assembly	Cold prototype	Glove box	Pitch-Bonded fuel rods			d	Low
6	Scrap recovery	Hot laboratory	Glove box	Miscellaneous off-spec material			d	Low
7	Off-gas treatment	Developed	Glove box	Radioactive gases			Non-fissionable	Low

Table 4.46. Analysis of Fuel Fabrication Method 21, Flow Diagram 2

Feed: $^{233}UO_2(NO_3)_2$ Solution

Product: Coated UO_2 Microspheres Within Spherical Graphite Fuel Elements

Process Step	Operation	Development Needed	Material Location	Material Description	Uranium Enrichment	Fissile Concentration	Convertability	Radiation Hazard
1	Fuel receiving storage	Cold prototype	Hot cell	$^{233}UO_2(NO_3)_2$	HEU	High	d	Medium
2	Fuel kernel fabrication	Cold prototype	Hot cell					
3	Fuel particle coating	Cold prototype	Hot cell	UO_2 microspheres	HEU	High	d	Medium
4	Fuel rod fabrication	NA	NA	NA	NA	NA	NA	NA
5	Fuel element assembly	Cold engineering	Hot cell	Coated fuel particles			d	Medium
6	Scrap recovery	Hot laboratory	Hot cell	Miscellaneous off-spec material			d	Medium
7	Off-gas treatment	Hot engineering	Hot cell	Radioactive gases			Non-fissionable	Low

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5. FUEL CYCLE EVALUATIONS FOR URANIUM-PLUTONIUM FUELS IN LWRs (Savannah River Laboratory)

Savannah River Laboratory

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Aiken, South Carolina

F. R. Field and F. E. Driggers

5.1 INTRODUCTION

The President's Nuclear Policy Statement of October 28, 1976, identified activities by the U.S. Government to reduce the risk of nuclear weapon proliferation while providing for the continued use of nuclear power worldwide. The United States Energy Research and Development Administration (ERDA) is developing a Nonproliferation Alternative Systems Assessment Program (NASAP) to implement the Nuclear Policy Statement. As shown in Figure 5.1, a cooperative program to identify and evaluate alternative nuclear fuel cycles is under the direction of the Division of Nuclear Research and Applications (NRA); the Division of Waste Management, Production, and Reprocessing (WPR); and the Division of Safeguards and Security (DSS). The objective of this program is to define viable fuel cycles which have the most potential for reducing the risks of nuclear weapon proliferation. Laboratories currently participating in the WPR program are:

Argonne National Laboratory (ANL)
Hanford Engineering Development Laboratory (HEDL)
Oak Ridge National Laboratory (ORNL)
Savannah River Laboratory (SRL)

The study of alternative fuel cycles is proceeding in phases; in Phase 1, the interest is to consider a wide range of possible fuel cycles. In Phase 2, the number of fuel cycles will be reduced for detailed study.

* The information contained in this article was developed during the course of work under Contract No. AT(07-2)-1 with the U.S. Energy Research and Development Administration.

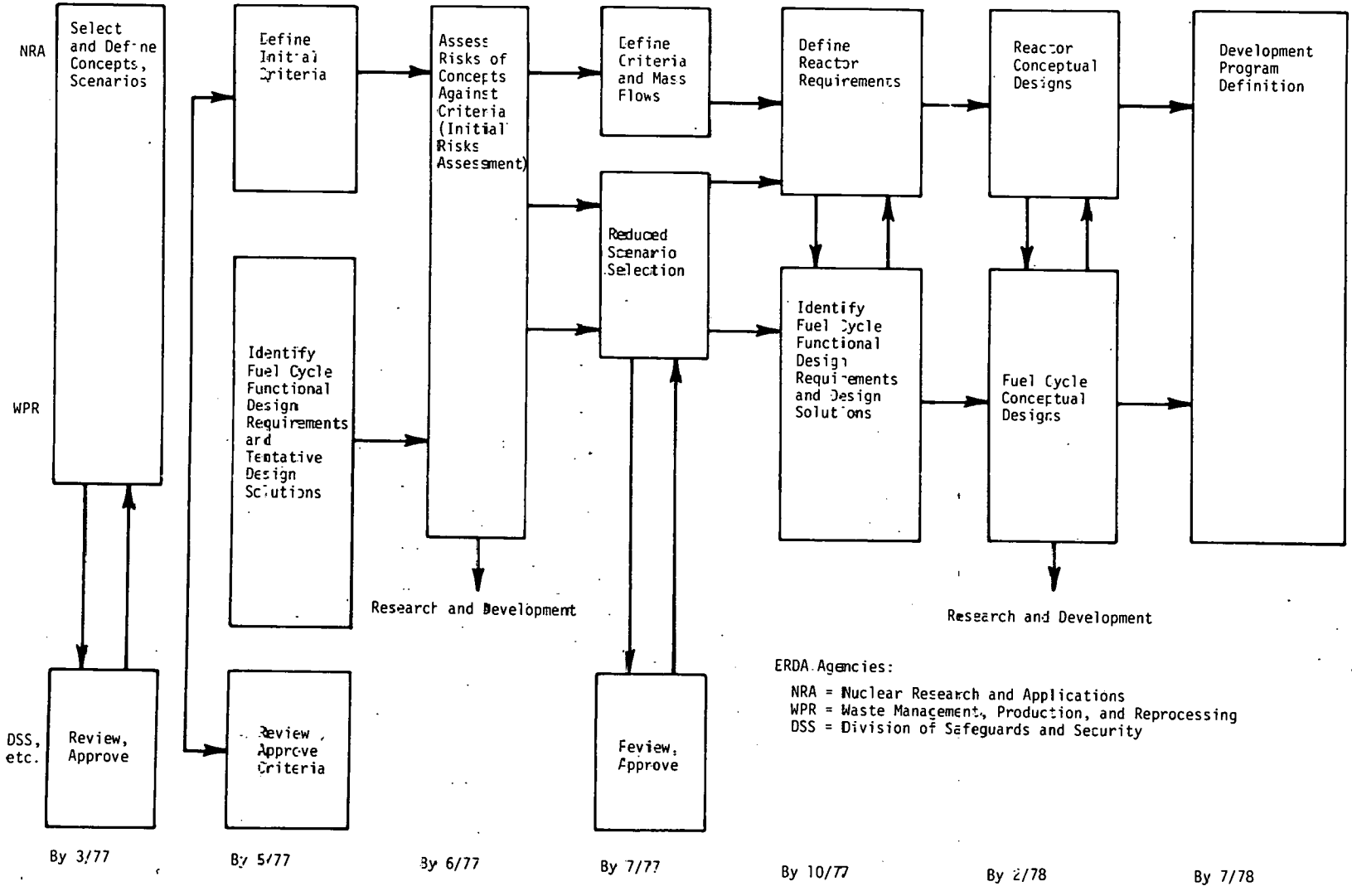


Fig. 5.1. Schedule for Assessment of Alternative Fuel Cycles

This report combines three previous Phase 1 SRL reports into a single document describing 16 fuel cycles (Table 5.1) and completes Phase 1 for SRL. The diagrams for the 16 fuel cycles cover two levels, Level 0 and Level 1. Level 0 is defined to include the major segments of a fuel cycle such as Enrichment, Reprocessing, Terminal Waste Storage, Mixed Oxide (MOX) Fuel Fabrication, etc., while Level 1 identifies major process steps within each segment; i.e., shearing fuel and solvent extraction are process steps for the Reprocessing segment. In Phase 2, Level 2 diagrams will expand Level 1 processes to detailed equipment operations. Costs and resource demands will be included.

Each Level 1 diagram is complemented by a table that describes the development status of each process and some of the relevant characteristics of the material in process (fissionable content, radiation level, etc.). The development status will be used to estimate the cost and time frame for development of such a fuel cycle. The material characteristics are intended to be used to analyze the relative resistance to proliferation of a fuel cycle or to compare fuel cycles to other non-fuel cycle paths to proliferation (e.g., research reactor or enrichment plant). The methods to analyze fuel cycles are to be developed by Science Applications, Inc., under contract to the NRA.

5.2 DISCUSSION

The 16 cases (Table 5.1) are discussed individually. Seven cases are associated with the light water reactor (LWR). Two cases concerning the gas-cooled reactor (GCR) are derived from generic LWR cases. Three cases concern the heavy water reactor (HWR), three cases involve hybrid LWR-HWR operation, and the final case, energy center (EC), is a combination of fuel cycle cases.

5.2.1 Case 1.1.1 (LWR): Recycle of Uranium and Plutonium (Base)

The major segments of this fuel cycle (Level 0) are shown schematically in Figure 5.2. Major process steps (Level 1) are shown in Figures 5.3, 5.4, and 5.5. The state-of-the-art and material characteristics for the chemical

Table 5.1. Fuel Cycle Alternatives

Case Description	Case Number	Tables	Process Figures
A. LWR			
Recycle of Uranium and Plutonium (Reference)	1.1.1	5.2	5.2 -5.5
Recycle of Uranium and Spiked Plutonium	1.1.2	5.3	5.6 -5.9
Recycle of Uranium and Spiked Uranium - Plutonium Oxide (Coprocessed)	1.1.3	5.4	5.10-5.13
Recycle of Uranium; Plutonium Diverted to High Activity Waste Stream	1.1.4	5.5	5.14-5.16
Fuel Throwaway - Oxide Fuel	1.1.5	5.6	5.17-5.19
Fuel Throwaway - Metal Fuel	1.1.6	5.7	5.20 5.22
Recycle of Uranium and Uranium-Plutonium Oxide (Coprocessed)	1.1.7	5.9	5.23-5.26
B. Gas-Cooled Reactors[a]			
Fuel Throwaway - Metal Fuel (Magnox)	2.1.6	5.10	5.27-5.29
Recycle of Uranium and Plutonium in Advanced Gas Reactors (Oxide Fuel)	2.1.7	5.11	5.30-5.33
C. HWR			
Recycle of Plutonium[b]	3.1.1	5.12	5.34-5.37
Recycle of Spiked Plutonium[b]	3.1.2	5.13	5.38-5.41
Fuel Throwaway - Oxide Fuel[b]	3.1.3	5.14	5.42-5.44
D. LWR-HWR Hybrids			
Tandem Cycle - Reconstituted LWR Fuel	3.3.1	5.15	5.45-5.48
Tandem Cycle - Fuel not Reconstituted[c] (LWR fuel compatible with HWR operation)	3.3.2	5.16	5.49-5.51
Spectral Shift Reactor - Throwaway of Oxide Fuel (LWR-HWR hybrid)[d]	3.4.1	5.17	5.52-5.53
E. Energy Center[e]			
U-Pu LWR with Recycle of Pu Fuel Inside Center; U Fueled LWR Outside Center, Irradiated Fuel Transferred to Energy Center	7.1.1	5.18	5.54-5.58

[a]Gas-cooled reactor cases are derived from generic LWR cases.

[b]HWR cases are derived from generic LWR cases, except uranium is not recycled because of low ^{235}U content.

[c]Derived from the previous tandem case.

[d]Derived from Case 1.1.5 for LWR fuel.

[e]Energy center cases are combinations of other fuel cycle cases.

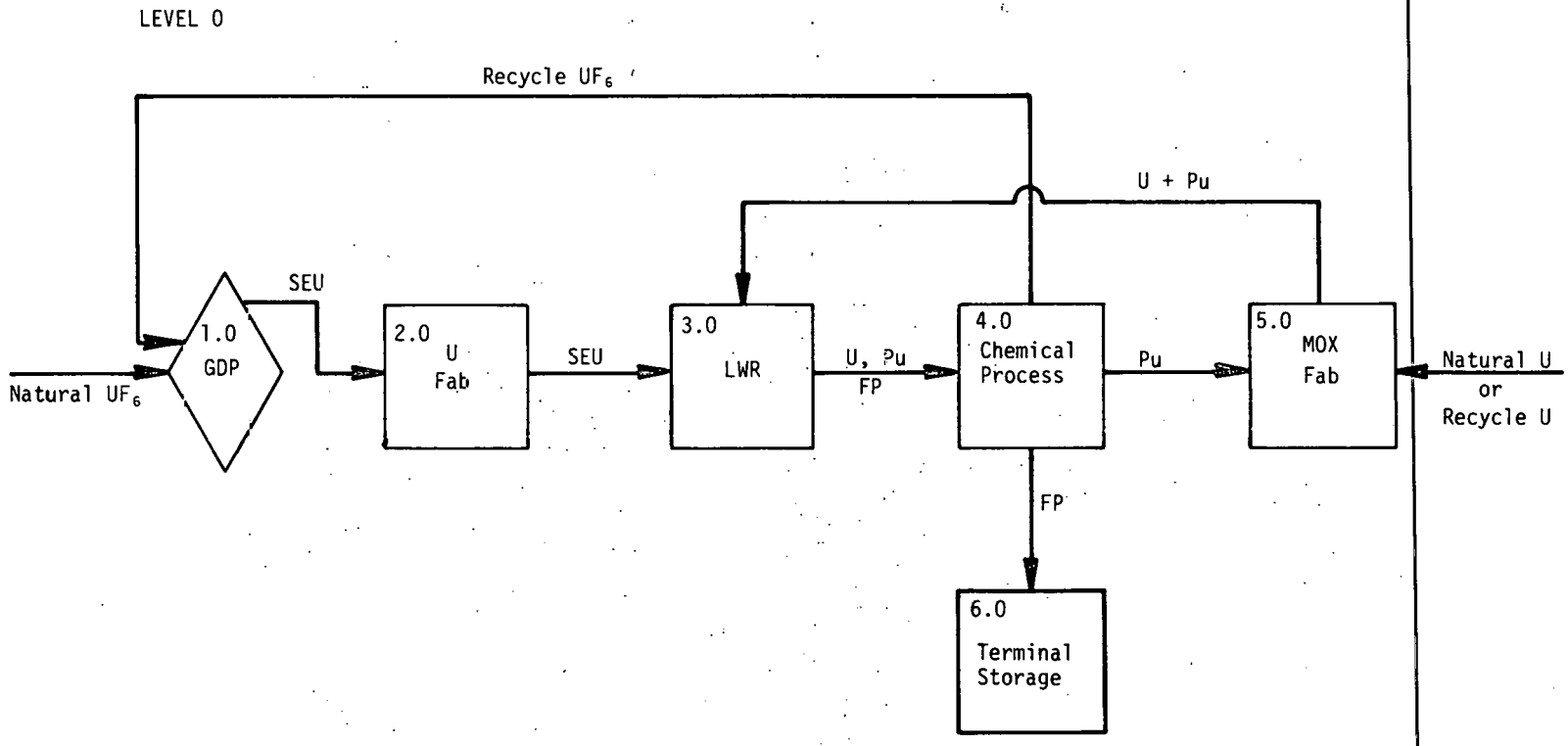
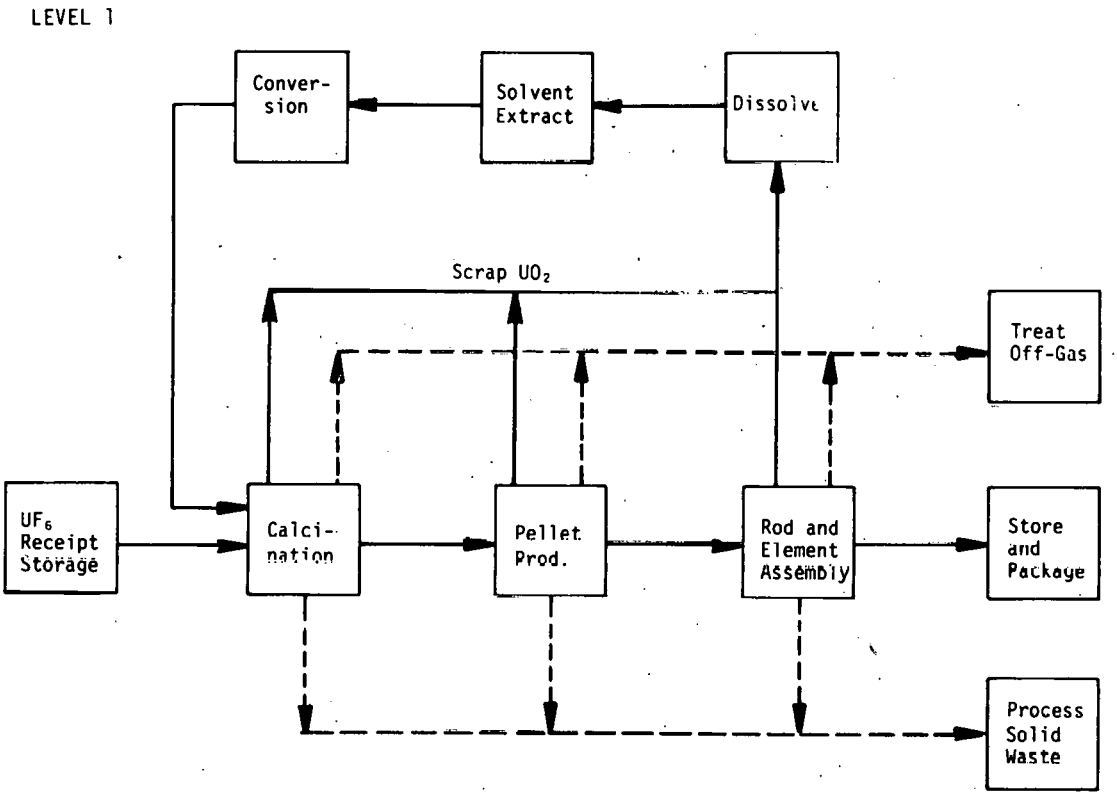


Fig. 5.2. Case 1.1.1 — Recycle of Uranium and Plutonium



All functions are developed state of the art; material has negligible radiation level.

Material convertibility is Category B for all steps except "Treat Off-Gas" and "Process Solid Waste."

Fig. 5.3. Case 1.1.1 — Recycle of Uranium and Plutonium; Slightly Enriched (2-4% ^{235}U) Uranium Fabrication (2.0)

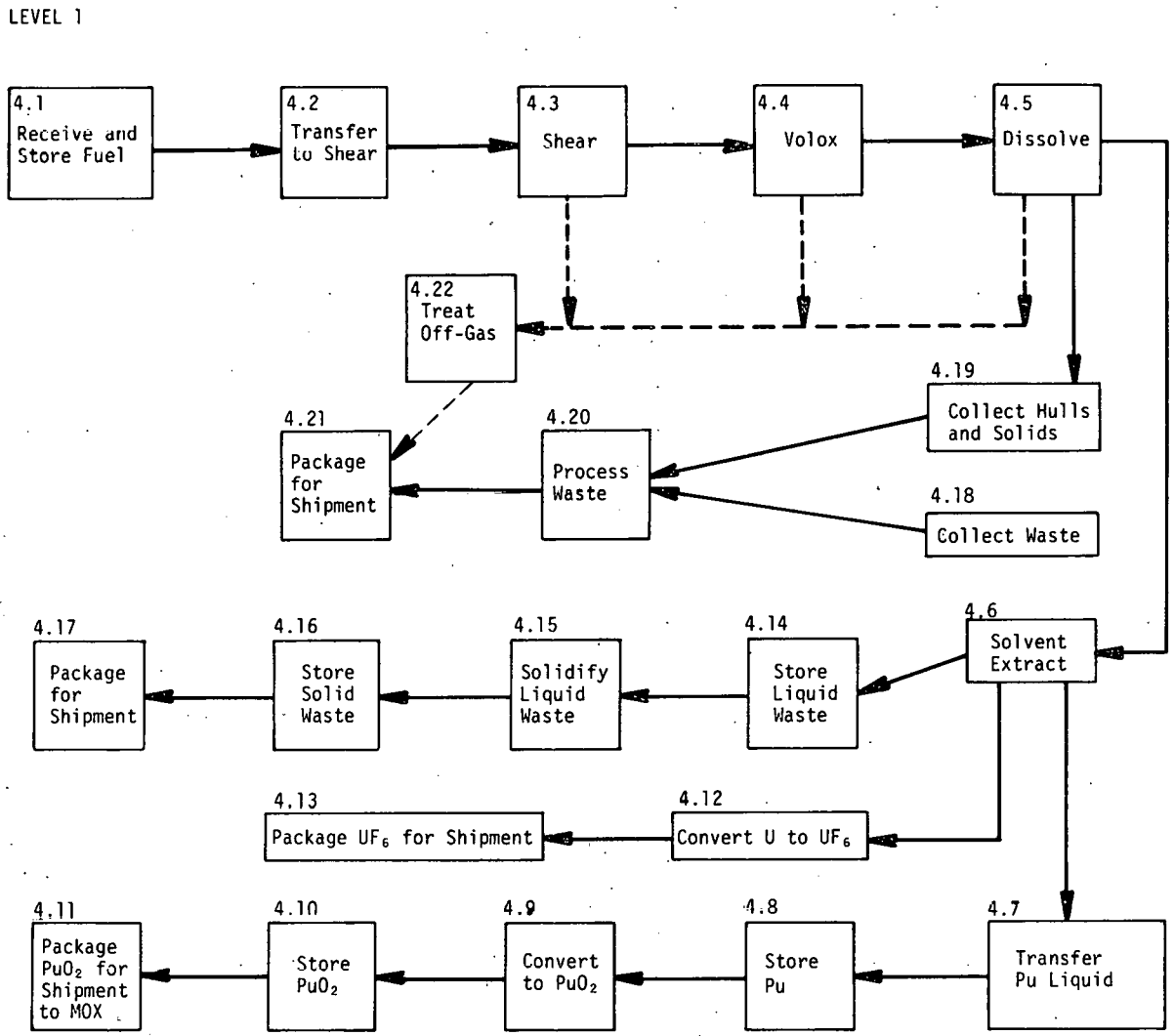


Fig. 5.4. Case 1.1.1 — Recycle of Uranium and Plutonium; Chemical Reprocessing (4.0)

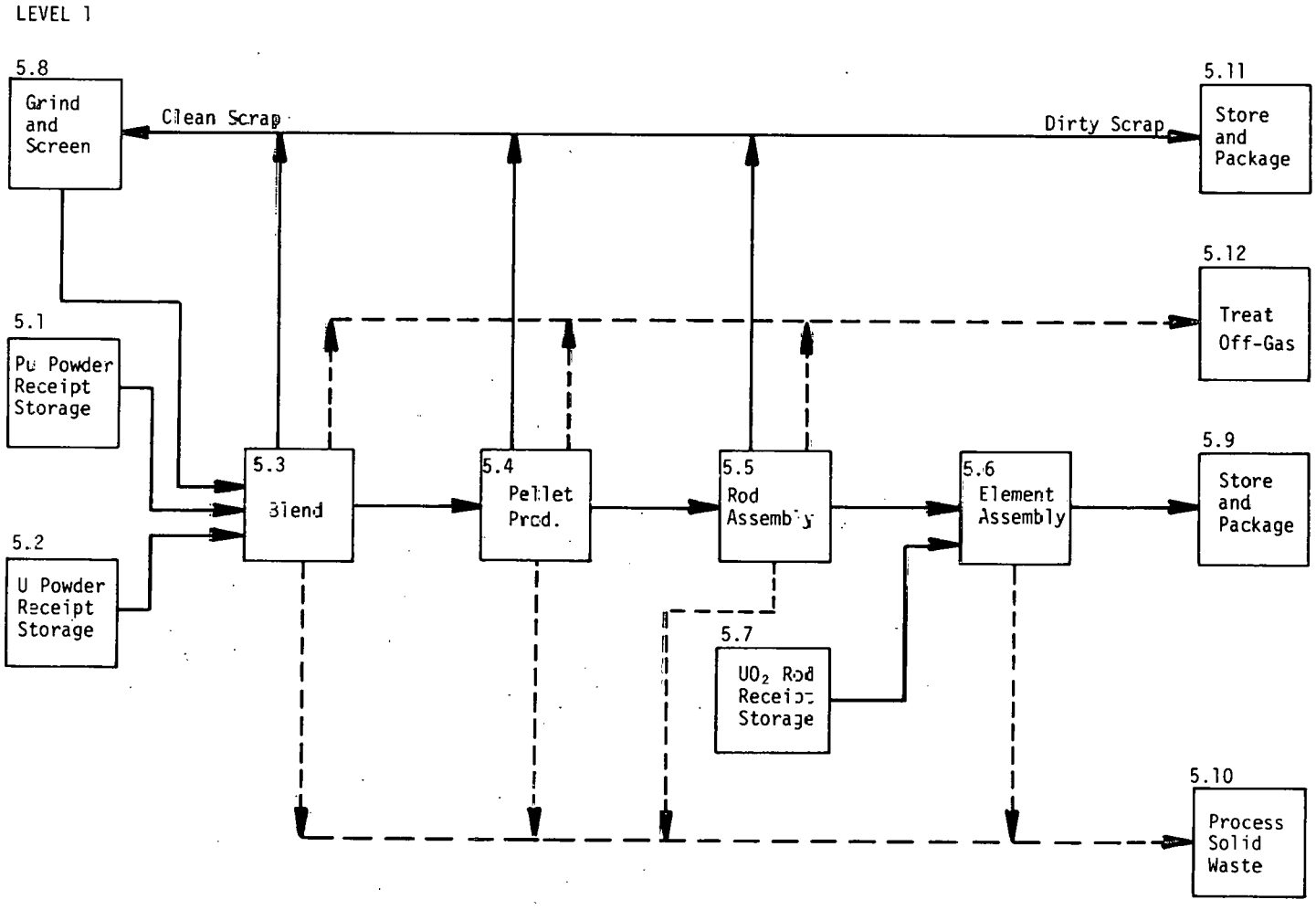


Fig. 5.5. Case 1.1.1 — Recycle of Uranium and Plutonium; MOX Fabrication (5.0)

reprocessing step (4.0) and the MOX fabrication step (5.0) are shown in Table 5.2. This fuel cycle is probably the most well-defined and will be useful in the study as a reference cost case.

The MOX plant is assumed to be located at a separate site from the reprocessing plant (also in Cases 1.1.2, 1.1.3, and 1.1.7). A co-located MOX plant could be another case for evaluation.

5.2.2 Case 1.1.2 (LWR): Recycle of Uranium and Spiked Plutonium

Level 0 segments are shown schematically in Figure 5.6; Level 1 steps, in Figures 5.7, 5.8, and 5.9. The state-of-the-art and material characteristics for the chemical reprocessing step (4.0) and refabrication step (5.0) are shown in Table 5.3.

Fission products in the plutonium may interfere with proven fabrication methods and quality control inspection of MOX fuel and thereby affect the fuel power guarantees that are customary for fuel fabricators to offer utilities.

The optimum amount of radiation to be emitted by the plutonium is a balance of the benefits gained in resistance to unauthorized use to the cost penalties in processing. Pending more study (see Section 5.4, "Spiking Plutonium"), the plutonium stream from the first Purex cycle of solvent extraction is assumed to contain the desired level of fission product content in the plutonium in Cases 1.1.2, 1.1.3, and 3.1.2. Radiation emitted will vary with fuel power, fuel decay period, chemical processing techniques, and other factors.

KEY to Symbols in Tables 5.2 through 5.18

A. State of the Art (8 stages)

S = Study Concept	HE = Hot Engineering
CL = Cold Laboratory	CP = Cold Prototype
HL - Hot Laboratory	HDF = Hot Demonstration Facility
CE = Cold Engineering	D = Developed

B. Radiation Hazard (4 ranges)

High:	$>10^4$ R/hr
Medium:	10 to 10^4 R/hr
Low:	<10 R/hr
Negligible:	$\lesssim$100 mR/hr

C. Material Convertibility (4 classes of fissionable material)

A = Material needs isotope separation for enrichment to weapons grade and is radioactive.

B = Material needs isotope separation for enrichment to weapons grade.

C = Highly radioactive material mixed with fissionable material, but chemically separable.

D = Minor impurities or minor chemical changes needed to convert to weapons-grade material.

Table 5.2. Case 1.1.1 (LWR): Recycle of Uranium and Plutonium (Reference Process)

	Process Step	State-of-the-Art	Material Description	Material Location	Radiation Hazard	Material Convertibility
A.	Chemical Reprocessing (4.0)					
4.1	Receive and Store Fuel	D	Irradiated Fuel	Basin	High	C
4.2	Transfer to Shear	D	Irradiated Fuel	Basin	High	C
4.3	Shear	D	Fuel Elements	Canyon	High	C
4.4	Voloxidation	HL	Fuel Element Pieces	Canyon	High	C
4.5	Dissolve	D	Fuel in Solution	Canyon	High	C
4.6	Solvent Extraction	D	Solution of U, Pu, Fission Products	Canyon	High	C
4.7	Transfer Pu Liquid	D	Pu Nitrate	Canyon	Low	D
4.8	Store Pu	D	Pu Nitrate	Canyon	Low	D
4.9	Convert to PuO_2	HE	PuO_2	Cell	Low	D
4.10	Store PuO_2	HE	PuO_2	Cell	Low	D
4.11	Package for Shipment	HE	PuO_2	Cell	Low	D
4.12	Convert U to UF_6	D	U Nitrate	Cell	Negligible	B
4.13	Package UF_6	D	UF_6	Warehouse	Negligible	B
4.14	Store Liquid Waste	D	Radioactive Liquids	Canyon	High	
4.15	Solidify Liquid Waste	HDF	Radioactive Solids	Canyon	High	
4.16	Store Solid Waste	HDF	Radioactive Solids	Canyon	High	
4.17	Package for Shipment	CE	Radioactive Solids	Canyon	High	
4.18	Collect Waste	D	Radioactive Solids	Canyon	Variable	
4.19	Collect Hulls and Solids	D	Radioactive Solids	Canyon	High	
4.20	Process Waste	CE	Radioactive Solids	Canyon	High	
4.21	Package for Shipment	CE	Radioactive Solids	Cell	High	
4.22	Treat Off-Gas	HL	Radioactive Gases	Canyon	High	
B.	MOX Fabrication (5.0)					
5.1	Receive and Store Pu Powder	HDF	PuO_2	Cell	Low	D
5.2	Receive and Store U Powder	HDF	UO_2	Cell	Negligible	B
5.3	Blend	HDF	PuO_2/UO_2	Cell	Low	D
5.4	Produce Pellets	HDF	PuO_2/UO_2	Cell	Low	D
5.5	Assemble Rods	HDF	PuO_2/UO_2/Cladding	Cell	Low	D
5.6	Prepare Assemblies	HDF	PuO_2/UO_2/Cladding	Cell	Low	D
5.7	UO_2 Rod Storage	D	UO_2/Cladding	Warehouse	Negligible	B
5.8	Regrind and Screen Scrap	HDF	PuO_2/UO_2	Cell	Low	B
5.9	Store and Package	HDF	PuO_2/UO_2	Warehouse	Low	B
5.10	Process Solid Waste	HDF	Miscellaneous	Cell	Low	
5.11	Store and Package Scrap	HDF	Miscellaneous	Cell		
5.12	Treat Off-Gas	HDF	Pu	Cell		

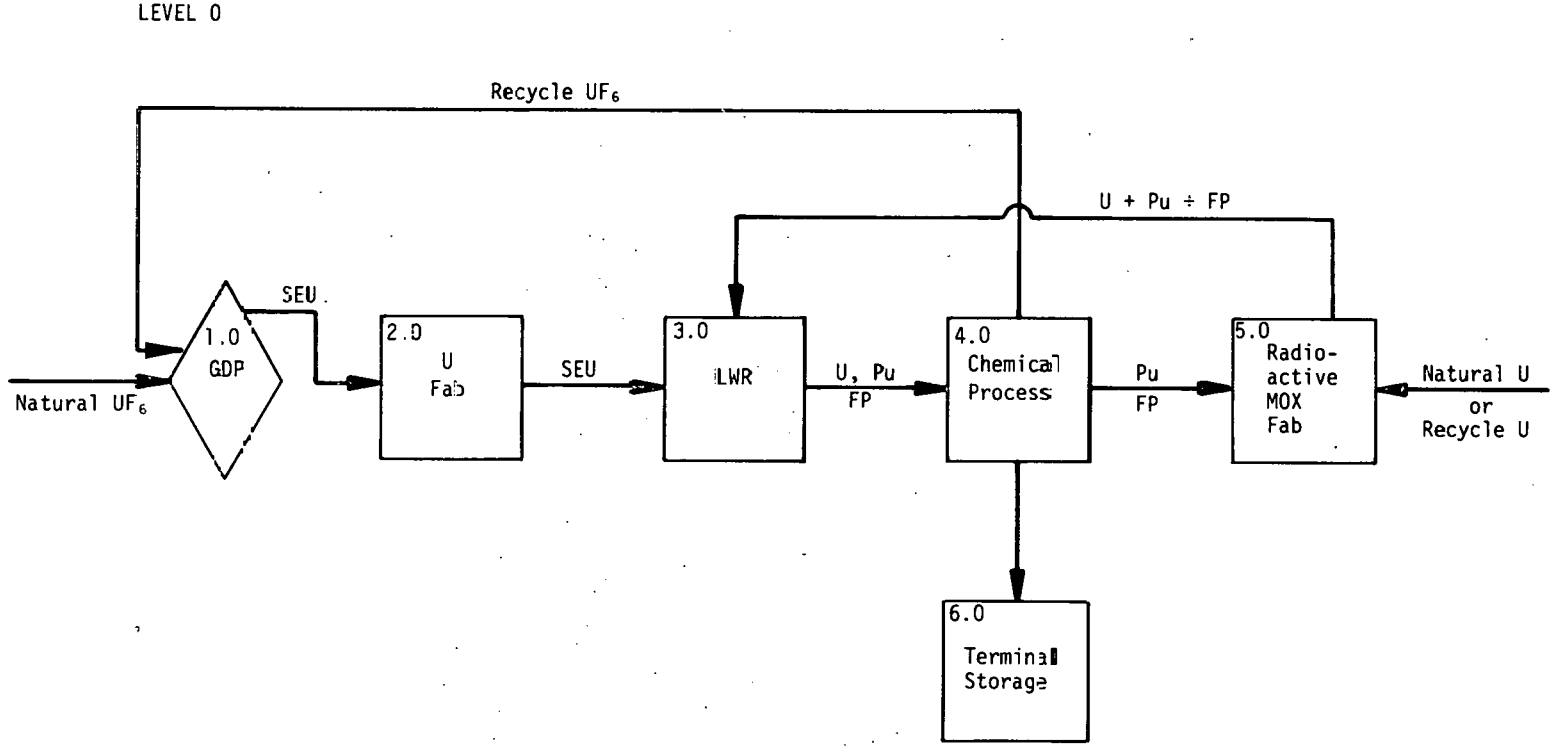
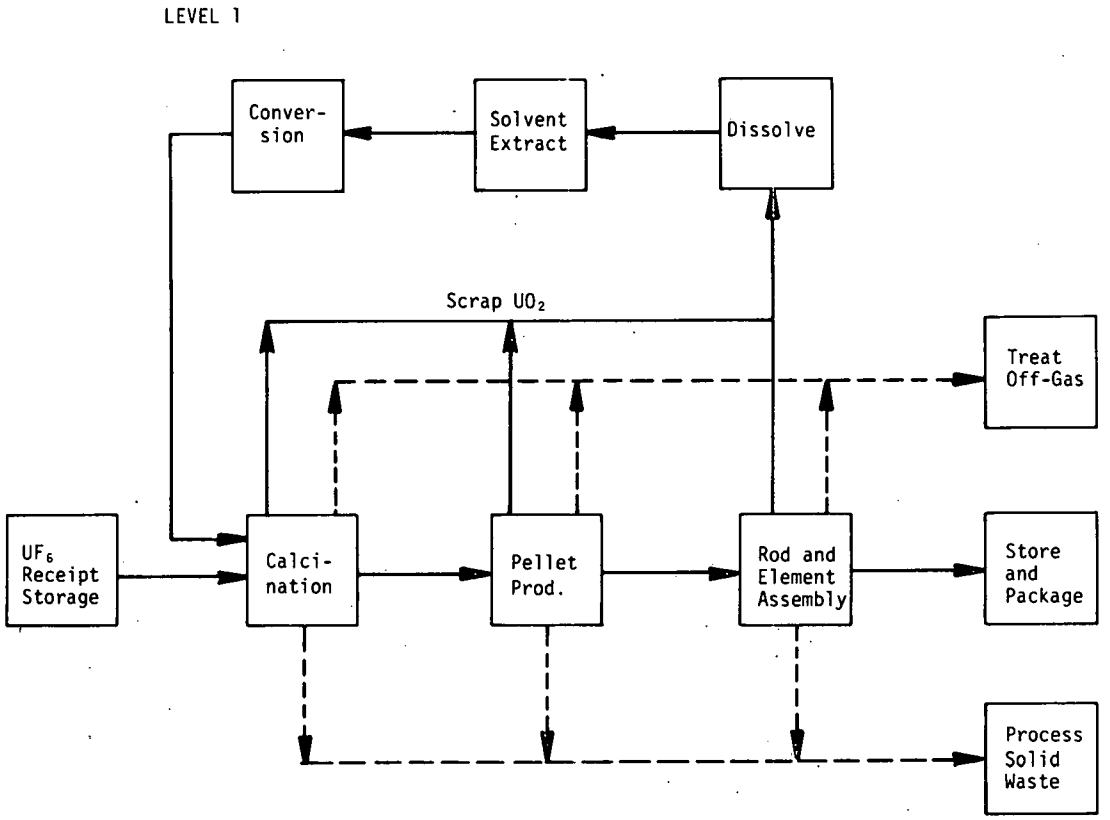


Fig. 5.6. Case 1.1.2 – Recycle of Uranium and Plutonium with Fission Products



All functions are developed state of the art; material is a negligible radiation hazard.

Material convertibility is Category B for all steps except "Treat Off-Gas" and "Process Solid Waste."

Fig. 5.7. Case 1.1.2 — Recycle of Uranium and Plutonium with Fission Products; Slightly Enriched (2-4% ^{235}U) Uranium Fabrication (2.0)

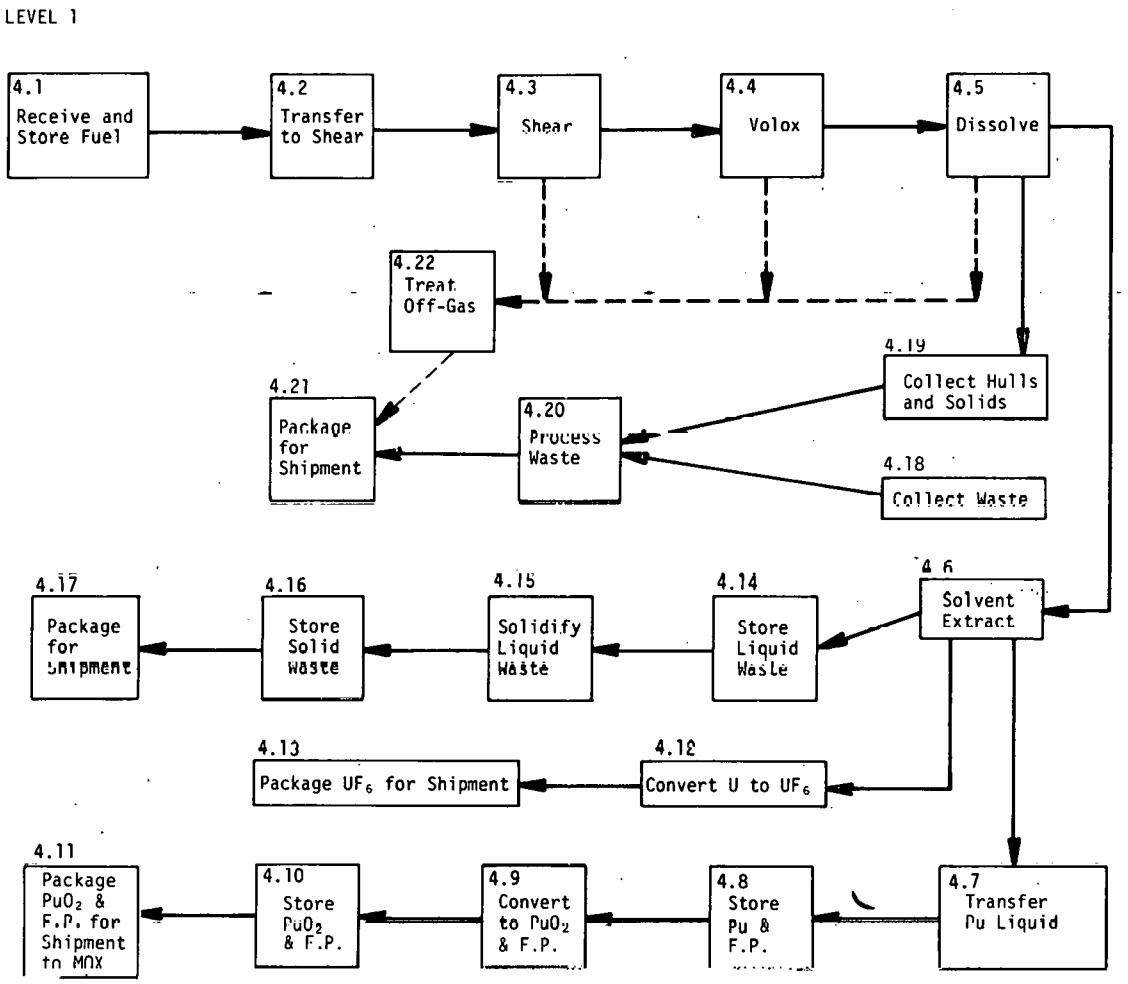


Fig. 5.8. Case 1.1.2 — Recycle of Uranium and Plutonium with Fission Products; Chemical Reprocessing (4.0)

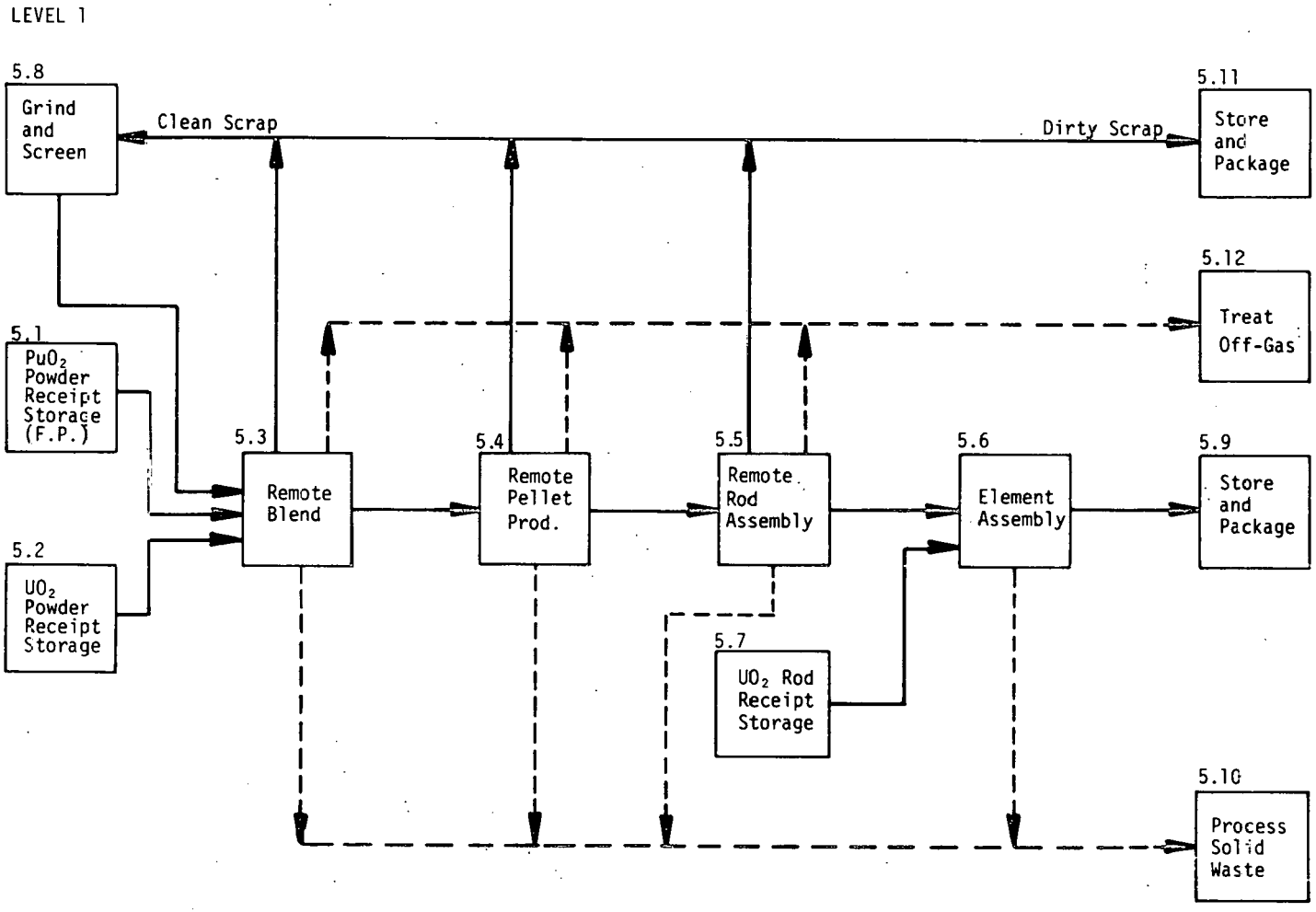


Fig. 5.9. Case 1.1.2 — Recycle of Uranium and Plutonium with Fission Products; MOX Fabrication (5.0)

Table 5.3. Case 1.1.2 (LWR): Recycle of Uranium and Spiked Plutonium

	Process Step	State-of-the-Art	Material Description	Material Location	Radiation Hazard	Material Convertibility
A.	Chemical Reprocessing (4.0)					
4.1	Receive and Store Fuel	D	Irradiated Fuel	Basin	High	C
4.2	Transfer to Shear	D	Irradiated Fuel	Basin	High	C
4.3	Shear	D	Fuel Elements	Canyon	High	C
4.4	Voloxidation	HL	Fuel Element Pieces	Canyon	High	C
4.5	Dissolve	D	Fuel in Solution	Canyon	High	C
4.6	Solvent Extraction	D	Solution of U, Pu, Fission Products	Canyon	High	C
4.7	Transfer Pu Liquid	CP	Pu Nitrate, Fission Prod.	Canyon	Medium	C
4.8	Store Pu	CP	Pu Nitrate, Fission Prod.	Canyon	Medium	C
4.9	Convert to PuO_2	CP	PuO_2, Fission Prod.	Cell	Medium	C
4.10	Store PuO_2	CP	PuO_2, Fission Prod.	Cell	Medium	C
4.11	Package for Shipment	CP	PuO_2, Fission Prod.	Cell	Medium	C
4.12	Convert U to UF_6	D	U Nitrate	Cell	Negligible	B
4.13	Package UF_6	D	UF_6	Warehouse	Negligible	B
4.14	Store Liquid Waste	D	Radioactive Liquids	Canyon	High	
4.15	Solidify Liquid Waste	HDF	Radioactive Solids	Canyon	High	
4.16	Store Solid Waste	HDF	Radioactive Solids	Canyon	High	
4.17	Package for Shipment	CE	Radioactive Solids	Canyon	High	
4.18	Collect Waste	D	Radioactive Solids	Canyon	Variable	
4.19	Collect Hulls and Solids	D	Radioactive Solids	Canyon	High	
4.20	Process Waste	CE	Radioactive Solids	Canyon	High	
4.21	Package for Shipment	CE	Radioactive Solids	Cell	High	
4.22	Treat Off-Gas	HL	Radioactive Gases	Canyon	High	
B.	MOX Fabrication (5.0)					
5.1	Receive and Store Pu Powder	CP	PuO_2/F.P.	Cell	Medium	C
5.2	Receive and Store U Powder	HDF	UO_2	Cell	Negligible	B
5.3	Blend	CP	PuO_2/UO_2/F.P.	Cell	Medium	C
5.4	Produce Pellets	CP	PuO_2/UO_2/F.P.	Cell	Medium	C
5.5	Assemble Rods	CP	PuO_2/UO_2/Cladding/F.P.	Cell	Medium	C
5.6	Prepare Assemblies	CP	PuO_2/UO_2/Cladding/F.P.	Cell	Medium	C
5.7	UO_2 Rod Storage	CP	UO_2/Cladding	Cell	Negligible	B
5.8	Regrind and Screen Scrap	CP	PuO_2/UO_2/F.P.	Cell	Medium	C
5.9	Store and Package	CP	PuO_2/UO_2/F.P.	Cell	Medium	C
5.10	Process Solid Waste	HDF	Miscellaneous	Cell		
5.11	Store and Package Scrap	HDF	Miscellaneous	Cell		
5.12	Treat Off-Gas	HDF	Pu	Cell		

5.2.3 Case 1.1.3 (LWR): Recycle of Uranium and Spiked Uranium - Plutonium Oxide (Coprocessed)

Level 0 segments are shown schematically in Figure 5.10; Level 1 steps, in Figures 5.11, 5.12, and 5.13. The state-of-the-art and material characteristics for the chemical processing step (4.0) and the MOX fabrication step (5.0) are shown in Table 5.4.

The coprocessed product is about 10% Pu - 90% U with excess uranium separated and converted to UF_6. The fission product content is assumed to vary as previously discussed. See also the section "Spiking Plutonium."

5.2.4 Case 1.1.4 (LWR): Recycle of Uranium (Plutonium Diverted to High Activity Waste Stream)

Level 0 segments are shown schematically in Figure 5.14; Level 1 steps, in Figures 5.15 and 5.16. The state-of-the-art and material description for the chemical reprocessing step (4.0) are shown in Table 5.5.

This fuel cycle complicates the waste processing steps and yields none of the economic benefit from plutonium recycle as in Reference Case 1.1.1.

5.2.5 Case 1.1.5 (LWR): Fuel Throwaway - Oxide Fuel

Level 0 segments are shown schematically in Figure 5.17; Level 1 steps, in Figures 5.18 and 5.19. The state-of-the art and material characteristics for the major processing steps are shown in Table 5.6.

A portion of this fuel cycle is the current mode of LWR operation in the U.S. and thus is a present-day reference similar to Case 1.1.1. Irradiated fuel is currently being stored at LWR sites in the U.S. ERDA has programs to define the away-from-reactor storage systems and the terminal waste disposal systems for irradiated LWR fuel.

5.2.6 Case 1.1.6 (LWR): Fuel Throwaway - Metal Fuel

Level 0 segments are shown schematically in Figure 5.20; Level 1 steps, in Figures 5.21 and 5.22. The state-of-the-art and material characteristics for the fuel encapsulation steps (8.0) are shown in Table 5.7.

The process description is similar to Case 1.1.5, except that no metal fuel fabrication industry exists. Uranium metal fuel for LWR's would

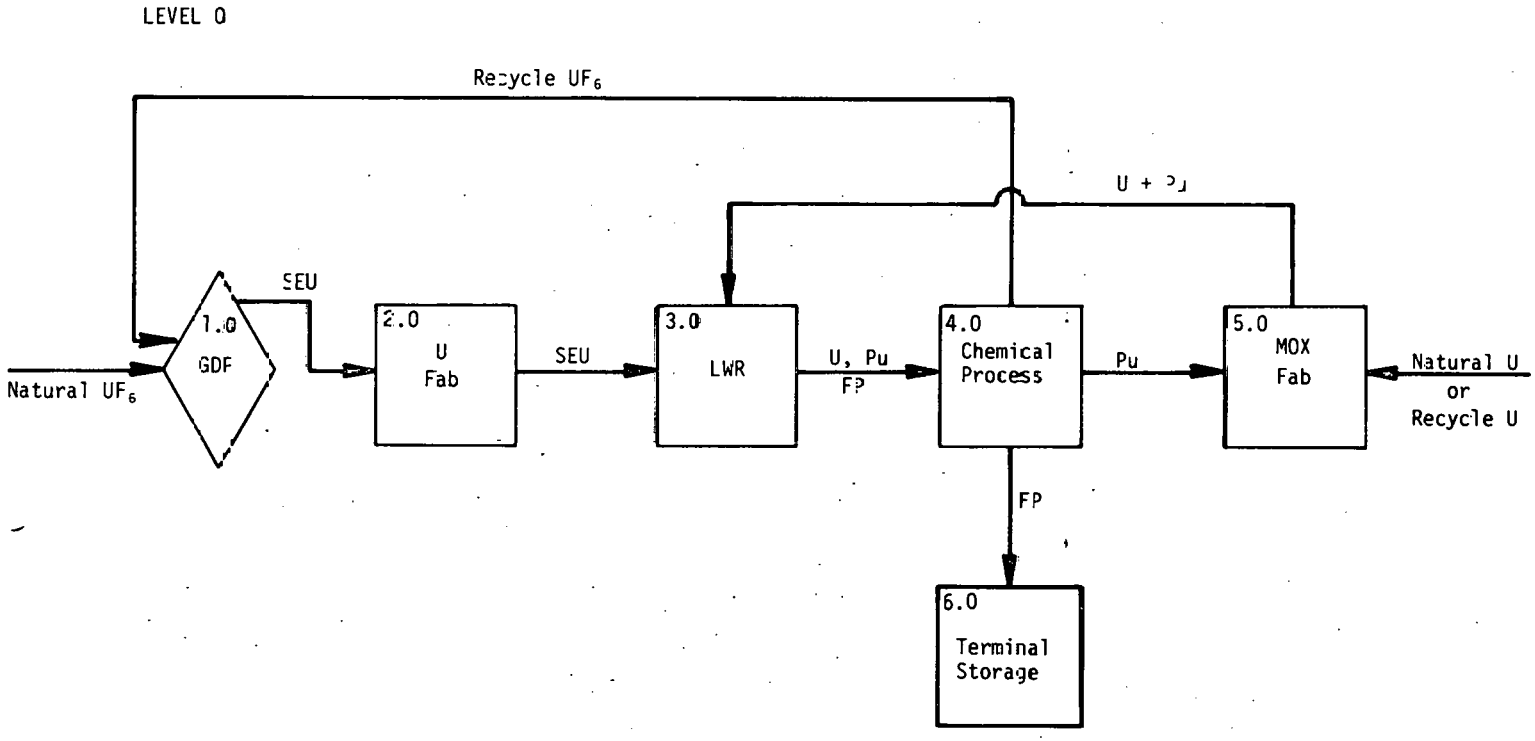
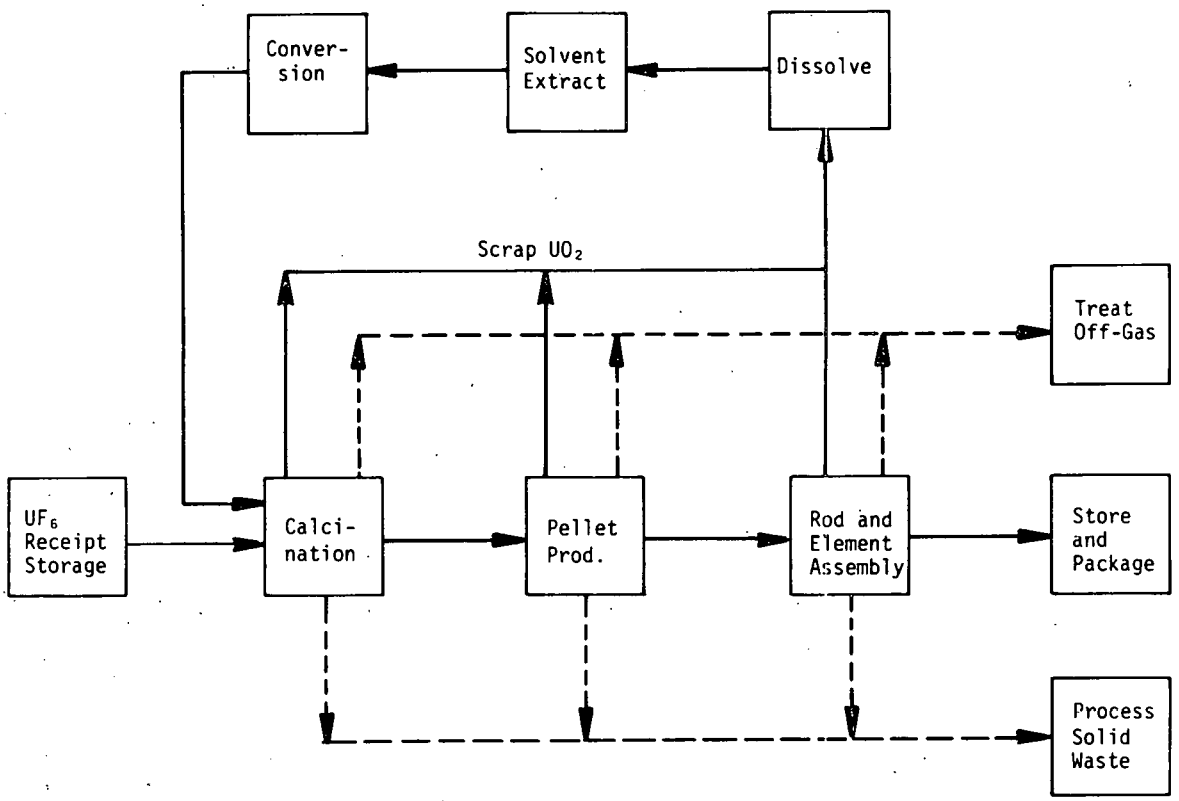


Fig. 5.10. Case 1.1.3 — Recycle of Uranium and Spiked Uranium-Plutonium Oxide (Coprocessed)

LEVEL 1



All functions are developed state of the art; material is a negligible radiation hazard.

Material convertibility is Category B for all steps except "Treat Off-Gas" and "Process Solid Waste."

Fig. 5.11. Case 1.1.3 — Recycle of Uranium and Spiked Uranium-Plutonium Oxide (Coprocessed); Slightly Enriched (2-4% ^{235}U) Uranium Fabrication (2.0)

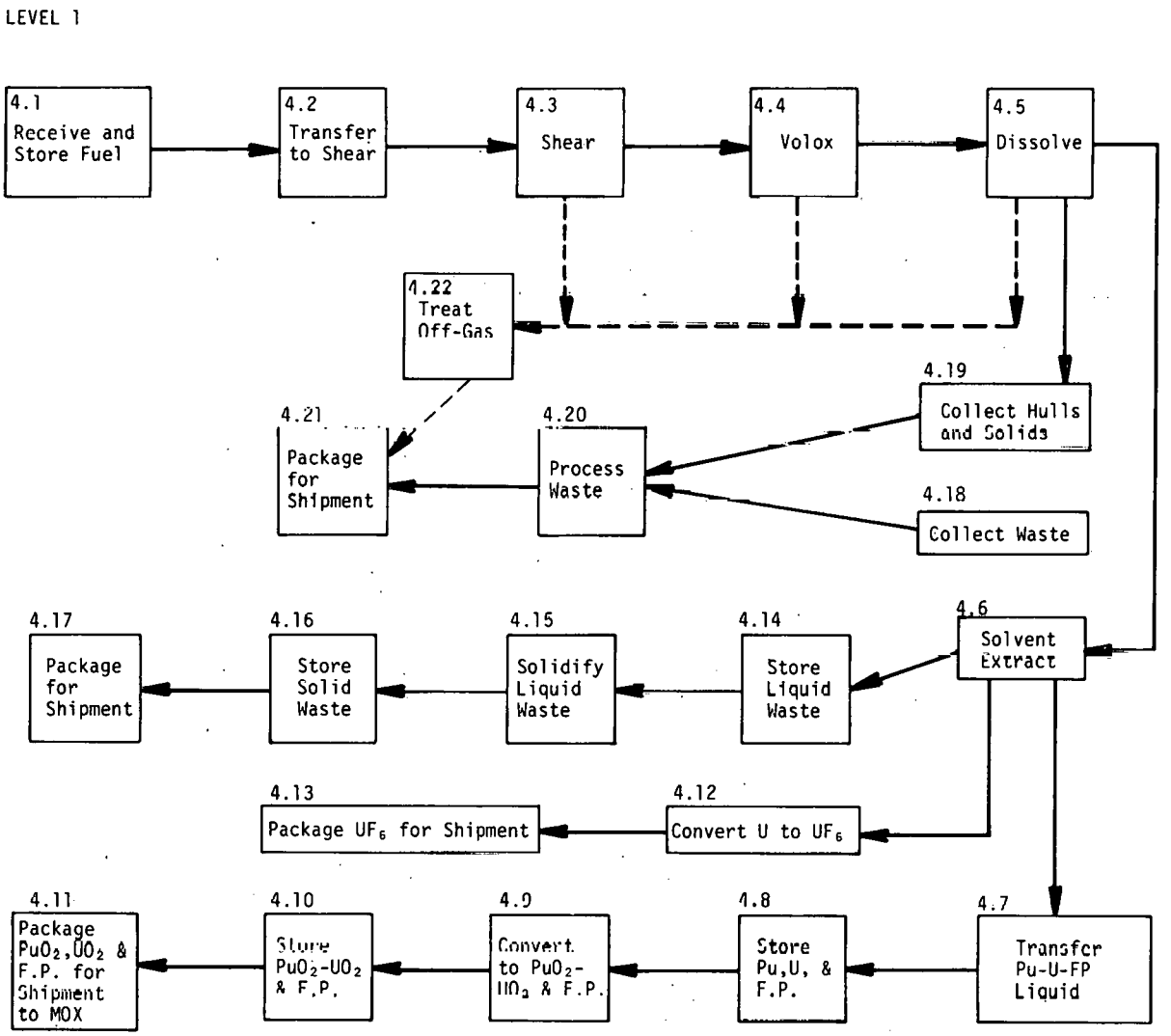


Fig. 5.12 Case 1.1.3 — Recycle of Uranium and Spiked Uranium - Plutonium Oxide (Coprocessed); Chemical Reprocessing (4.0)

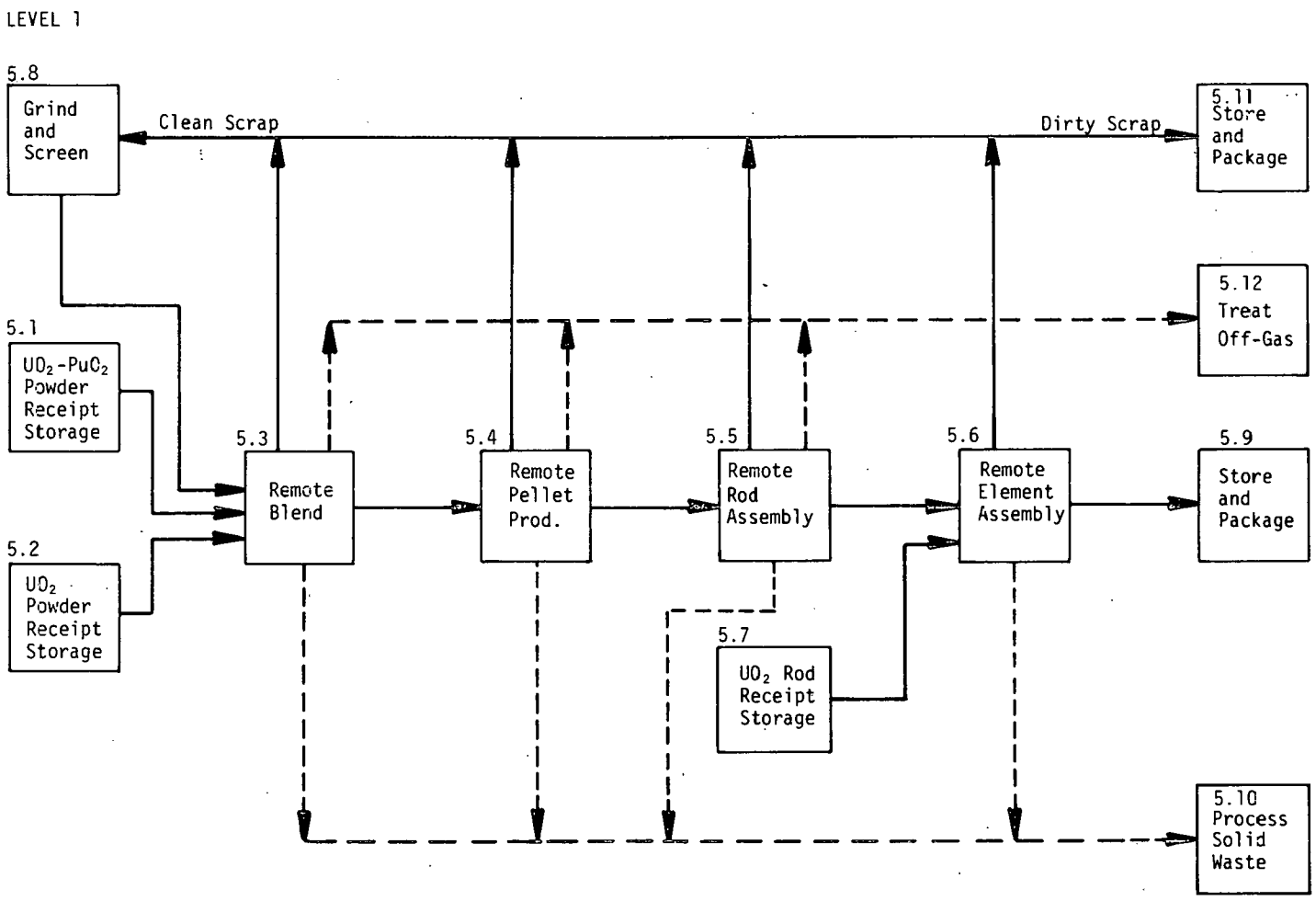


Fig. 5.13. Case 1.1.3 – Recycle of Uranium and Spiked Uranium-Plutonium Oxide (Coprocessed); MOX Fabrication (5.0)

Table 5.4. Case 1.1.3 (LWR): Recycle of Uranium and Spiked Uranium-Plutonium Oxide (Coprocessed)

	Process Step	State-of-the-Art	Material Description	Material Location	Radiation Hazard	Material Convertibility
A.	Chemical Reprocessing (4.0)					
4.1	Receive and Store Fuel	D	Irradiated Fuel	Basin	High	C
4.2	Transfer to Shear	D	Irradiated Fuel	Basin	High	C
4.3	Shear	D	Fuel Elements	Canyon	High	C
4.4	Voloxidation	HL	Fuel Element Pieces	Canyon	High	C
4.5	Dissolve	D	Fuel in Solution	Canyon	High	C
4.6	Solvent Extraction	D	Solution of U, Pu, Fission Products	Canyon	High	C
4.7	Transfer U-Pu Liquid	CP	U-Pu Nitrate/F.P.	Canyon	Medium	C
4.8	Store U-Pu Liquid	CP	U-Pu Nitrate/F.P.	Canyon	Medium	C
4.9	Convert to PuO_2-UO_2	CP	PuO_2/UO_2/F.P.	Cell	Medium	C
4.10	Store PuO_2 UO_2	CP	PuO_2/UO_2/F.P.	Cell	Medium	C
4.11	Package for Shipment	CP	PuO_2/UO_2/F.P.	Cell	Medium	C
4.12	Convert U to UF_6	D	U Nitrate	Cell	Negligible	B
4.13	Package UF_6	D	UF_6	Warehouse	Negligible	B
4.14	Store Liquid Waste	D	Radioactive Liquids	Canyon	High	
4.15	Solidify Liquid Waste	HDF	Radioactive Solids	Canyon	High	
4.16	Store Solid Waste	HDF	Radioactive Solids	Canyon	High	
4.17	Package for Shipment	CE	Radioactive Solids	Canyon	High	
4.18	Collect Waste	D	Radioactive Solids	Canyon	Variable	
4.19	Collect Hulls and Solids	D	Radioactive Solids	Canyon	High	
4.20	Process Waste	CE	Radioactive Solids	Canyon	High	
4.21	Package for Shipment	CE	Radioactive Solids	Cell	High	
4.22	Treat Off-Gas	HL	Radioactive Gases	Canyon	High	
B.	MOX Fabrication (5.0)					
5.1	Receive and Store Pu Powder	CP	PuO_2/F.P.	Cell	Medium	C
5.2	Receive and Store U Powder	HDF	UO_2	Cell	Negligible	B
5.3	Blend	CP	PuO_2/UO_2/F.P.	Cell	Medium	C
5.4	Produce Pellets	CP	PuO_2/UO_2/F.P.	Cell	Medium	C
5.5	Assemble Rods	CP	PuO_2/UO_2/Cladding	Cell	Medium	C
5.6	Prepare Assemblies	CP	PuO_2/UO_2/Cladding	Cell	Medium	C
5.7	UO_2 Rod Storage	CP	UO_2/Cladding	Cell	Negligible	B
5.8	Regrind and Screen Scrap	CP	PuO_2/UO_2	Cell	Medium	C
5.9	Store and Package	CP	PuO_2/UO_2	Cell	Medium	C
5.10	Process Solid Waste	HDF	Miscellaneous	Cell		
5.11	Store and Package Scrap	HDF	Miscellaneous	Cell		
5.12	Treat Off-Gas	HDF	Pu	Cell		

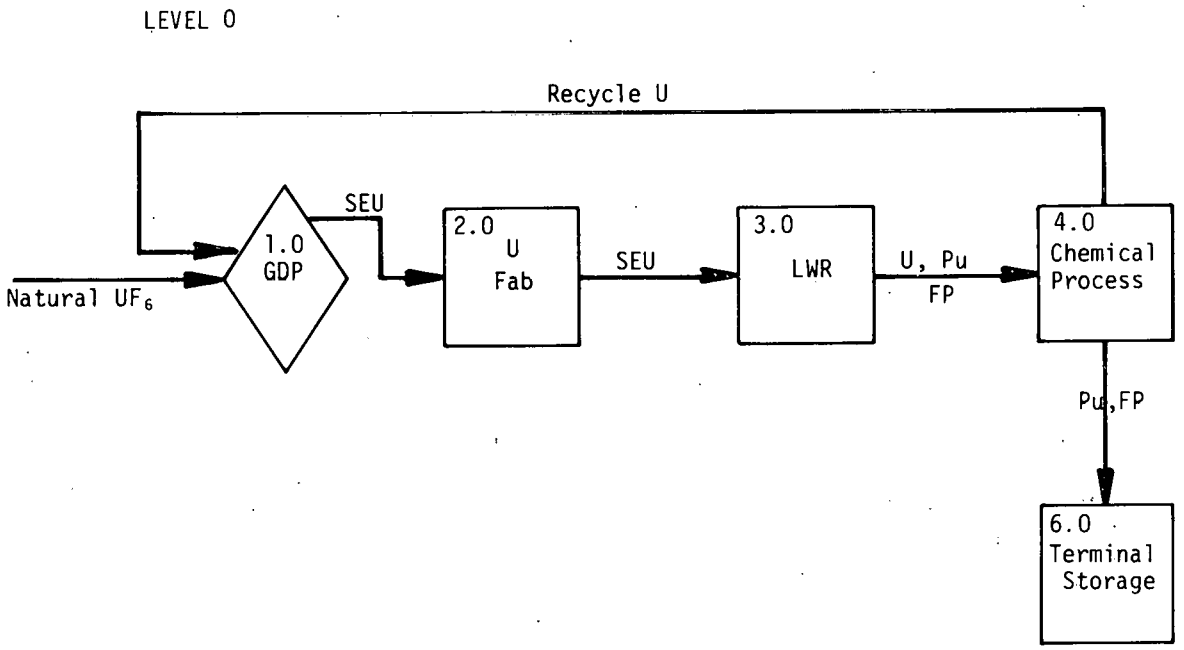
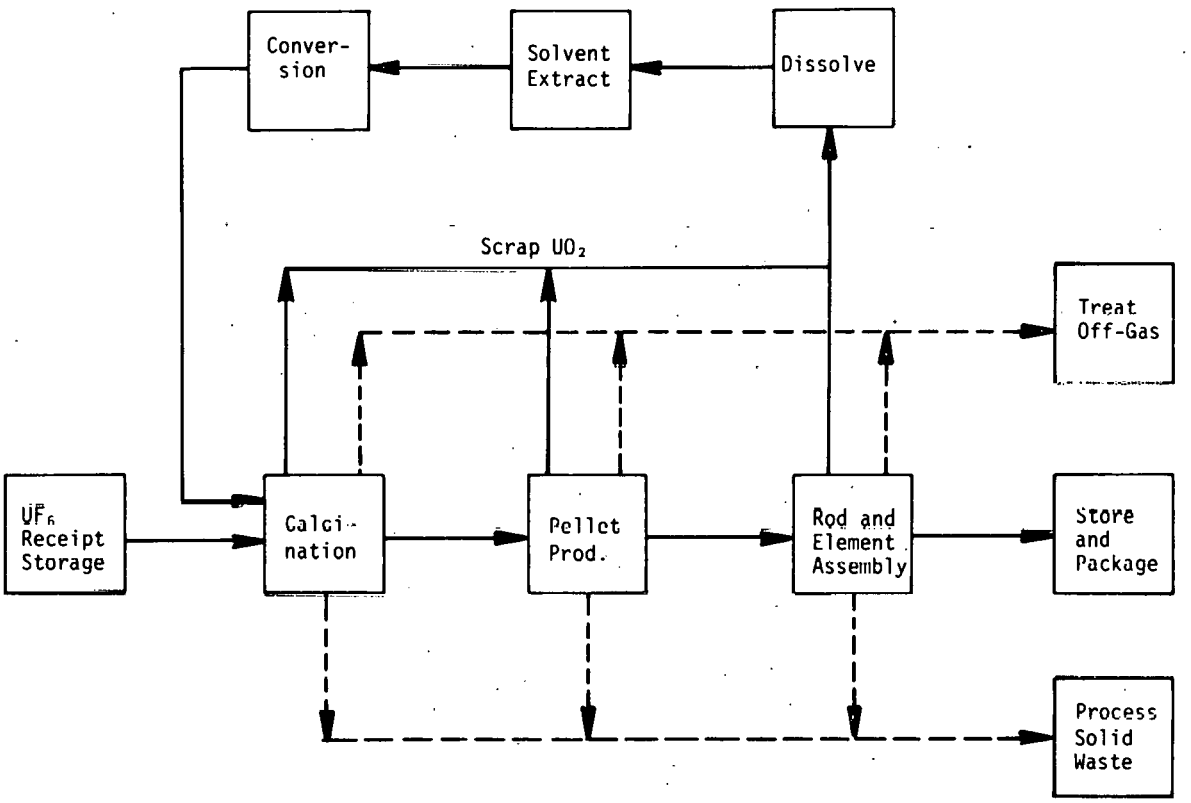


Fig. 5.14. Case 1.1.4 — Recycle of Uranium (Diversion of Plutonium to HAW Stream)

LEVEL 1



All functions are developed state of the art; material is a negligible radiation hazard.

Material convertibility is Category B for all steps except "Treat Off-Gas" and "Process Solid Waste."

Fig. 5.15. Case 1.1.4 — Recycle of Uranium (Diversion of Plutonium to HAW Stream); Slightly Enriched (2-4% ^{235}U) Uranium Fabrication (2.0)

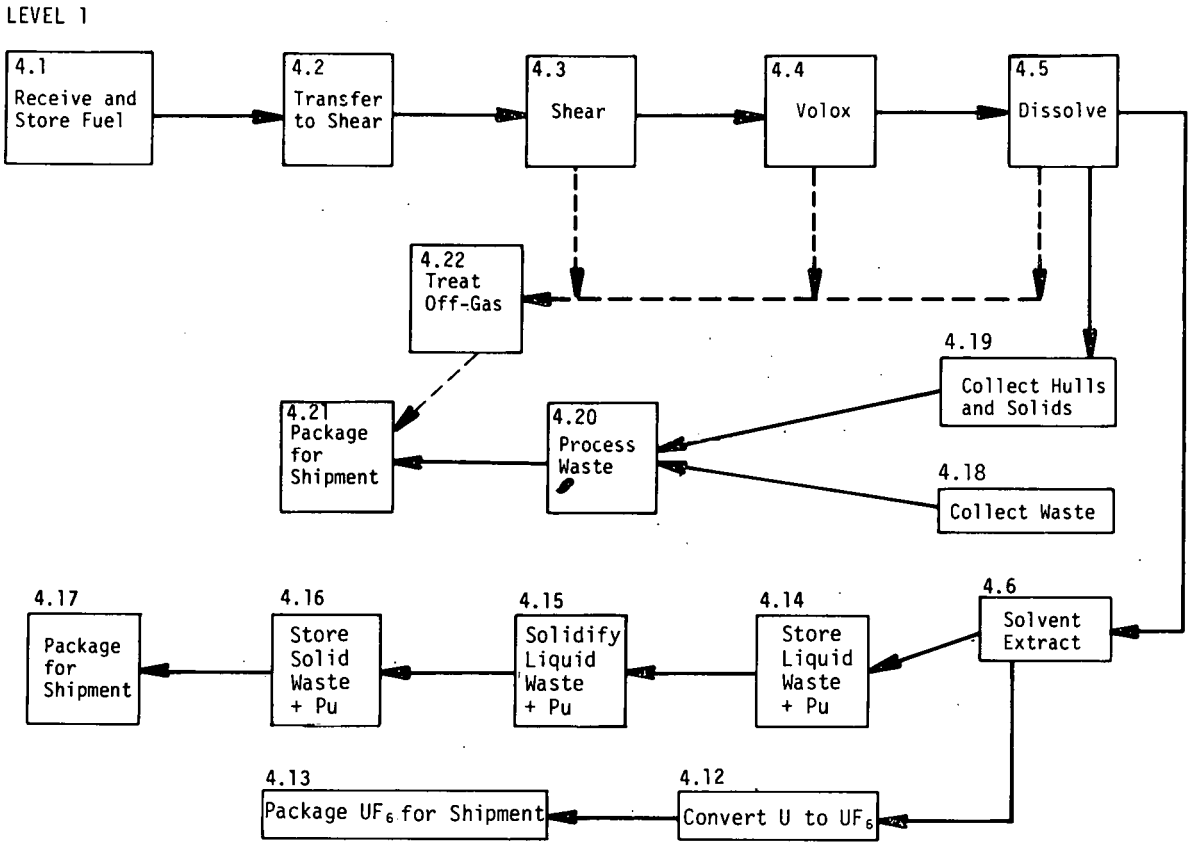


Fig. 5.16. Case 1.1.4 — Recycle of Uranium (Diversion of Plutonium to HAW Stream); Chemical Reprocessing (4.0)

Table 5.5. Case 1.1.4 (LWR): Recycle of Uranium (Plutonium Diverted to Waste)

		Process Step	State-of-the Art	Material Description	Material Location	Radiation Hazard	Material Convertibility
A.	Chemical Reprocessing (4.0)						
	4.1	Receive and Store Fuel	D	Irradiated Fuel	Basin	High	C
	4.2	Transfer to Shear	D	Irradiated Fuel	Basin	High	C
	4.3	Shear	D	Fuel Elements	Canyon	High	C
	4.4	Voloxidation	HL	Fuel Element Pieces	Canyon	High	C
	4.5	Dissolve	D	Fuel in Solution	Canyon	High	C
	4.6	Solvent Extraction	D	Solution of U, Pu, Fission Products	Canyon	High	C
	4.12	Convert U to UF_6	D	U Nitrate	Cell	Negligible	B
	4.13	Package UF_6	D	UF_6	Warehouse	Negligible	B
	4.14	Store Liquid Waste	D	Radioactive Liquids/Pu	Canyon	High	C
	4.15	Solidify Liquid Waste	HDF	Radioactive Solids/Pu	Canyon	High	C
	4.16	Store Solid Waste	HDF	Radioactive Solids/Pu	Canyon	High	C
	4.17	Package for Shipment	CE	Radioactive Solids/Pu	Canyon	High	C
	4.18	Collect Waste	D	Radioactive Solids	Canyon	Variable	
	4.19	Collect Hulls and Solids	D	Radioactive Solids	Canyon	High	
	4.20	Process Waste	CF	Radioactive Solids	Canyon	High	
	4.21	Package for Shipment	CE	Radioactive Solids	Cell	High	
	4.22	Treat Off-Gas	HL	Radioactive Gases	Canyon	High	

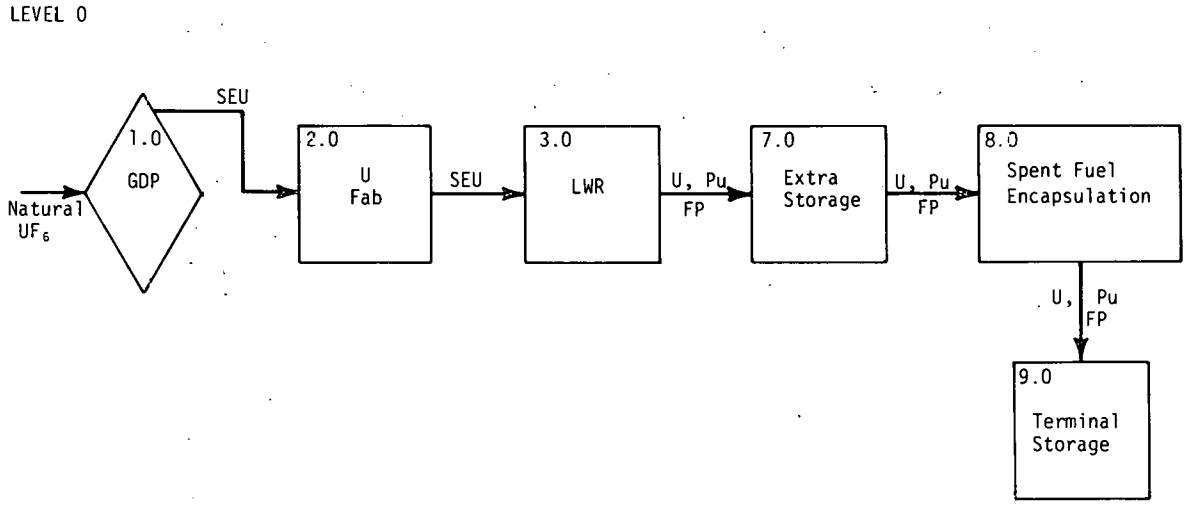
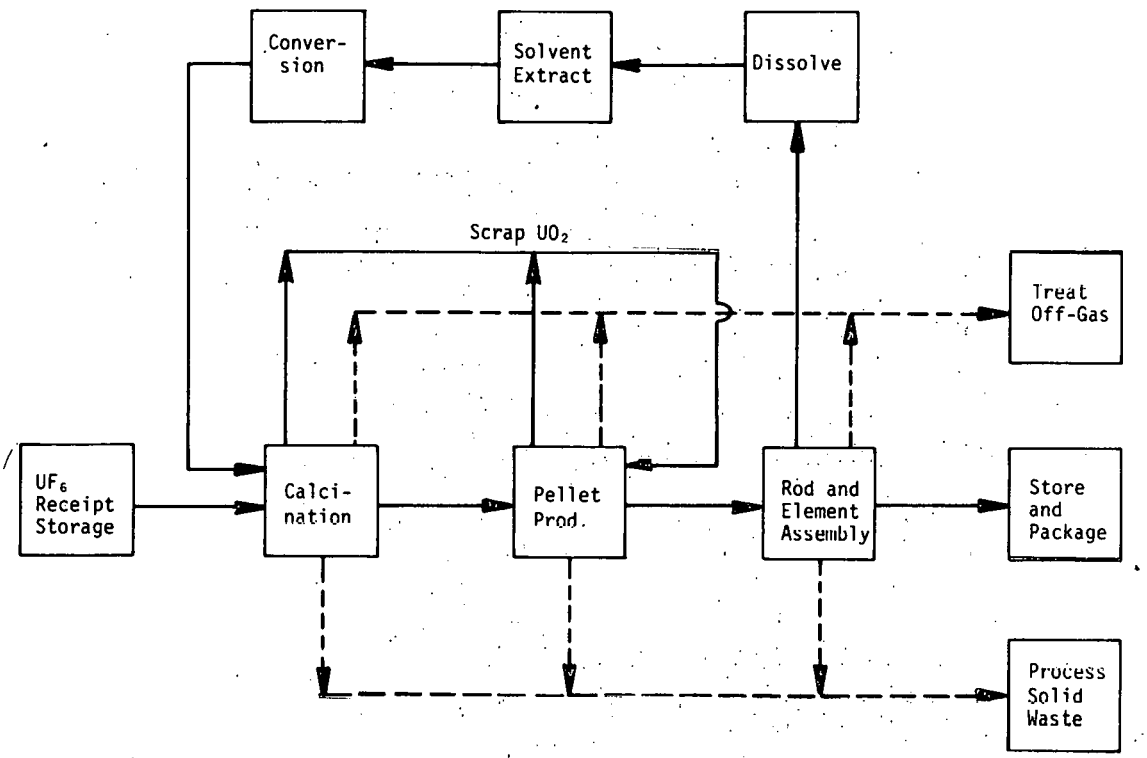


Fig. 5.17. Case 1.1.5 — Fuel Throwaway (Oxide Fuel)

LEVEL 1



All functions are developed state of the art; material is a negligible radiation hazard.

Material convertibility is Category B for all steps except "Treat Off-Gas" and "Process Solid Waste."

Fig. 5.18. Case 1.1.5 — Fuel Throwaway (Oxide Fuel); Slightly Enriched (2-4% ^{235}U) Uranium Fabrication (2.0)

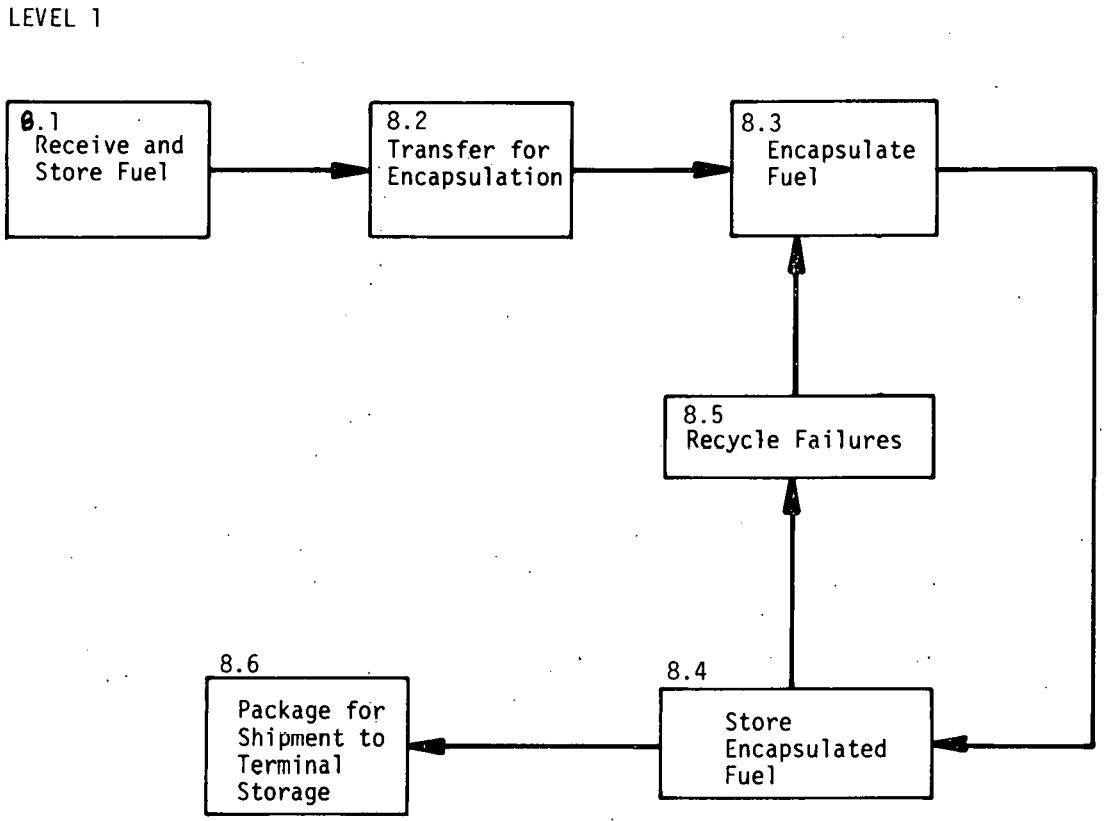


FIG. 5.19. Case 1.1.5 — Fuel Throwaway (Oxide Fuel); Encapsulate Fuel (5.0)

Table 5.6. Case 1.1.5 (LWR): Fuel Throwaway (Oxide Fuel)

Process Step	State-of-the-Art	Material Description	Material Location	Radiation Hazard	Material Convertibility
A. Encapsulation (8.0)					
8.1 Receive Fuel	D	Irradiated LWR Fuel	Basin	High[a]	C
8.2 Transfer to Encapsulation	D	Irradiated Fuel	Basin	High	C
8.3 Encapsulate Fuel Remove Gas	S[b]	Irradiated Fuel	Canyon	High	C
8.4 Store Encapsulated Fuel	D	Irradiated Fuel	Basin or Vault	High	C
8.5 Recycle Failures	S	Failed Irradiated Fuel	Canyon	High	C
8.6 Package for Shipment to Storage	CE	Irradiated Fuel	Basin or Vault	High	C

[a]Varies with age of fuel.

[b]Some Canadian development work for CANDU fuel may be applicable.

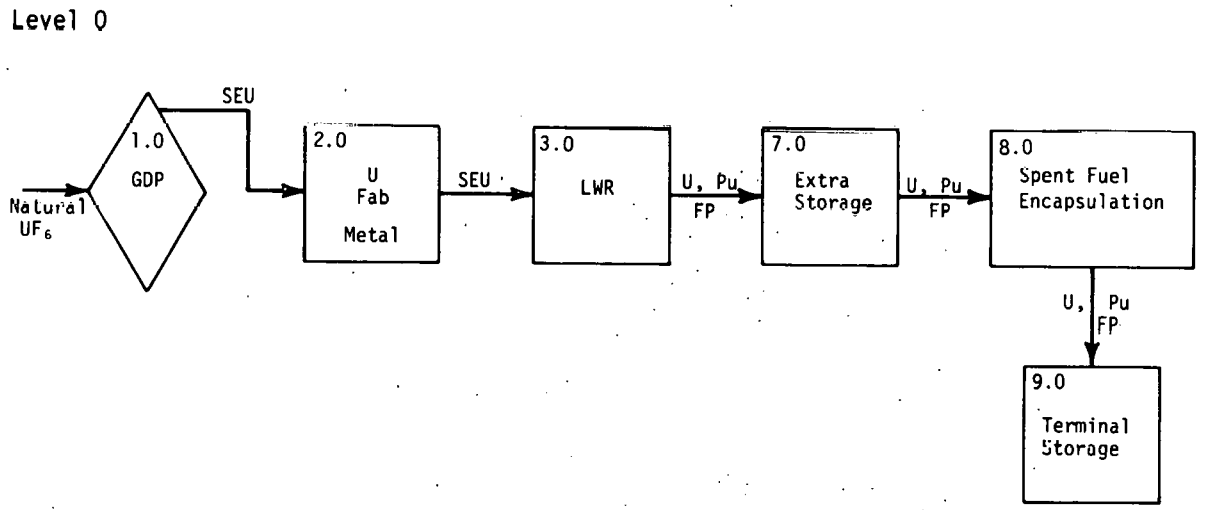
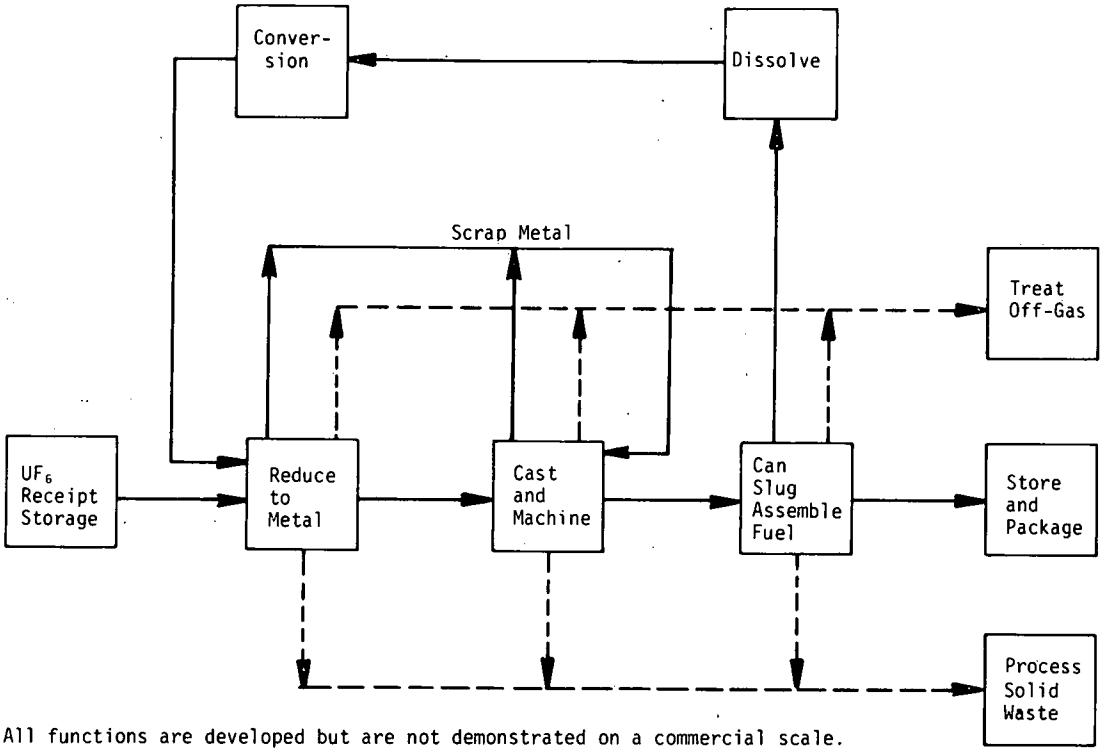


Fig. 5.20. Case 1.1.6 — Fuel Throwaway (Metal Fuel)

LEVEL 1



All functions are developed but are not demonstrated on a commercial scale.

Material is a negligible radiation hazard.

Material convertibility is Category B for all steps except "Treat Off-Gas" and "Process Solid Waste."

Fig. 5.21. Case 1.1.6 — Fuel Throwaway (Metal Fuel); Slightly Enriched (2-4% ^{235}U) Uranium Fabrication (2.0)

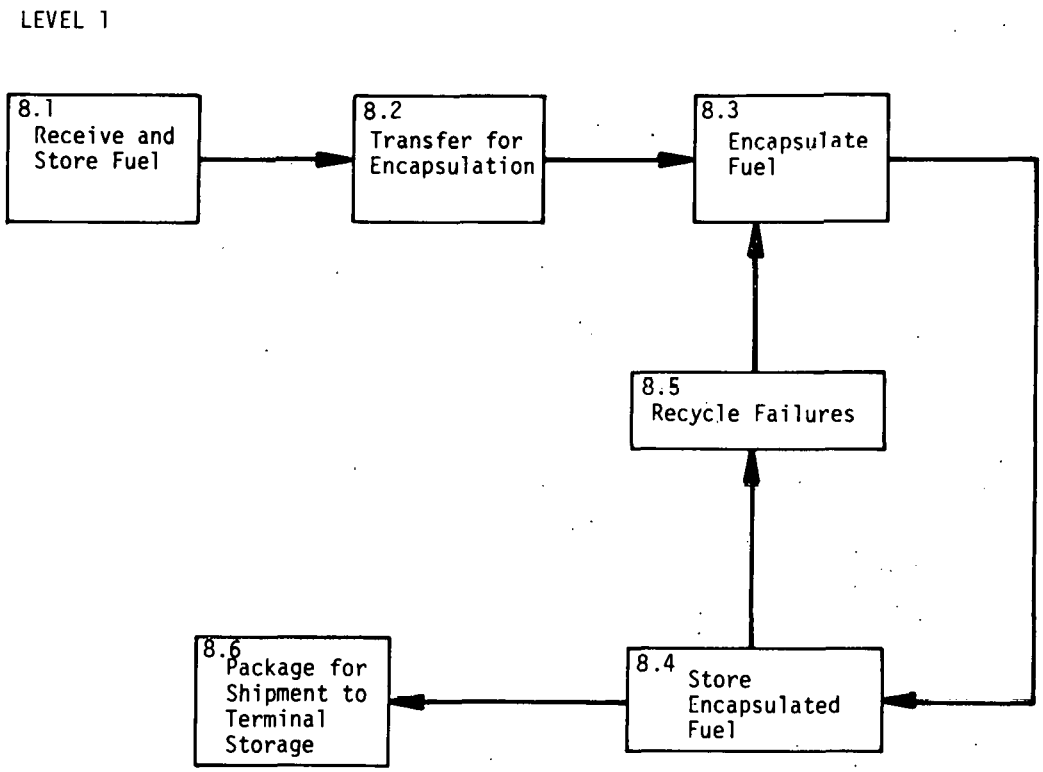


Fig. 5.22. Case 1.1.6 – Fuel Throwaway (Metal Fuel); Encapsulate Fuel (8.0)

Table 5.7. Comparison of Oxide and Metal Fuel Cycles for LWR

Cycle	Oxide	Metal
Fuel Enrichment, % ^{235}U charged	3.2	1.5
Fuel Exposure, MWD/tonne	30,000	12,000 (6000)[a]
Uranium Feed Required, Tails = 0.25% ^{235}U (throwaway) kg/kg fuel	5.87	2.71
Normalized and Corrected for Fuel Exposure	1.0	1.15
Enrichment Required, Tails = 0.25% ^{235}U (throwaway), SWU/kg fuel	4.21	1.06
Normalized and Corrected for Fuel Exposure	1.0	0.63
Pu Produced, kg fissile/MT	6.2	4.1
Relative Pu Produced, kg/reactor-year	1.0	~1.5

[a]The metal density is about twice the oxide density, so the exposure conversion is necessary for exposure comparisons.

probably raise formidable problems with respect to safety (metal-water reaction in case of a fuel failure) and exposure (fuel growth) characteristics. Nor does the use of uranium metal fuel appear to have any safeguards or non-proliferation advantages.

Operating characteristics of uranium oxide and uranium metal fuel cycles are compared in Table 5.8. The estimates indicate some enrichment savings, but more ore consumption with metal fuel. The plutonium produced in the fuel would be about 80% fissile, and considerably more plutonium would be produced annually with metal fuel. The estimates are based on metal fuel exposures that are considered difficult to attain without major penalties on LWR power as a consequence of fuel growth. If a major metallurgical program were initiated, metal fuel for use in LWR's might be available in about 20 years (10 years for development and 10 years to commercialize and license the fuel cycle). Licensing the uranium metal fuel cycle in the U.S. may be difficult because the safety reviews would address the risk of a metal-water reaction in the event of a cladding failure. The chemical energy is high for the metal-water reaction, but negligible for uranium oxide fuel.

Chemical recovery of plutonium from metal fuel diverted from the throwaway mode is no more difficult than from oxide fuel. The waste volume occupied by the irradiated metal fuel discharged from LWR would equal or exceed that of oxide fuel because of the lower exposure attainable with metal fuel.

5.2.7 Case 1.1.7 (LWR): Recycle of Uranium and Uranium-Plutonium Oxide (Coprocessed)

Level 0 segments are shown schematically in Figure 5.23; Level 1 steps, in Figures 5.24, 5.25, and 5.26. The state-of-the-art and material characteristics for the chemical reprocessing step (4.0) and the MOX fabrication step (5.0) are shown in Table 5.9.

This case is similar to Case 1.1.3 except a source of high radiation has not been deliberately added to the plutonium-containing stream as a non-proliferation measure. This case is analyzed to identify the benefit, if any, of radiation in the plutonium stream as a proliferation deterrent.

Table 5.8. Case 1.1.6 (LWR): Fuel Throwaway (Metal Fuel)

	Process Step	State-of-the-Art	Material Description	Material Location	Radiation Hazard	Material Convertibility
A.	Encapsulation (8.0)					
	8.1 Receive Fuel	D	Irradiated LWR Fuel	Basin	High[a]	C[a]
	8.2 Transfer to Encapsulation	D	Irradiated Fuel	Basin	High	C
	8.3 Encapsulate Fuel Remove Gas	S[b]	Irradiated Fuel	Canyon	High	C
	8.4 Store Encapsulated Fuel	D	Irradiated Fuel	Basin or Vault	High	C
	8.5 Recycle Failures	S	Failed Irradiated Fuel	Canyon	High	C
	8.6 Package for Shipment	CE	Irradiated Fuel	Basin or Vault	High	C

[a] Varies with age of fuel.

[b] Some Canadian development work for CANDU fuel may be applicable.

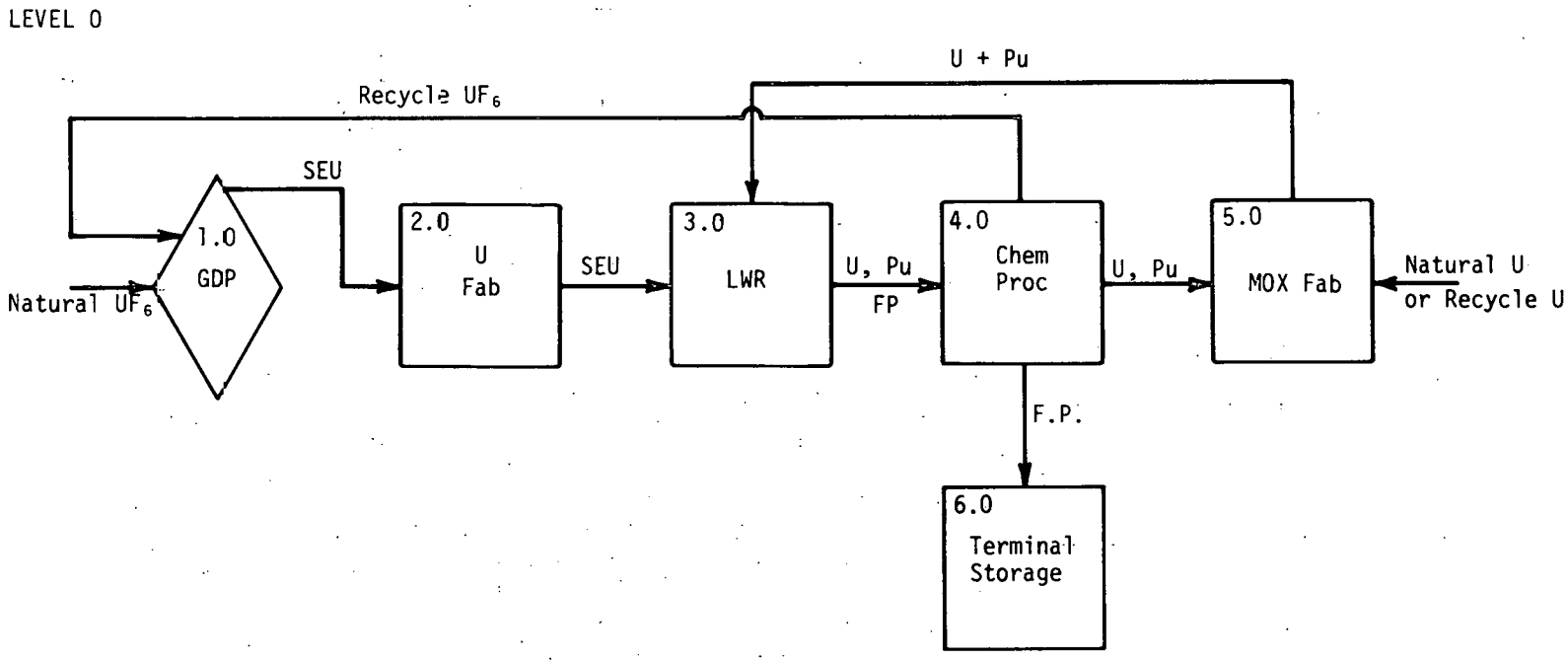
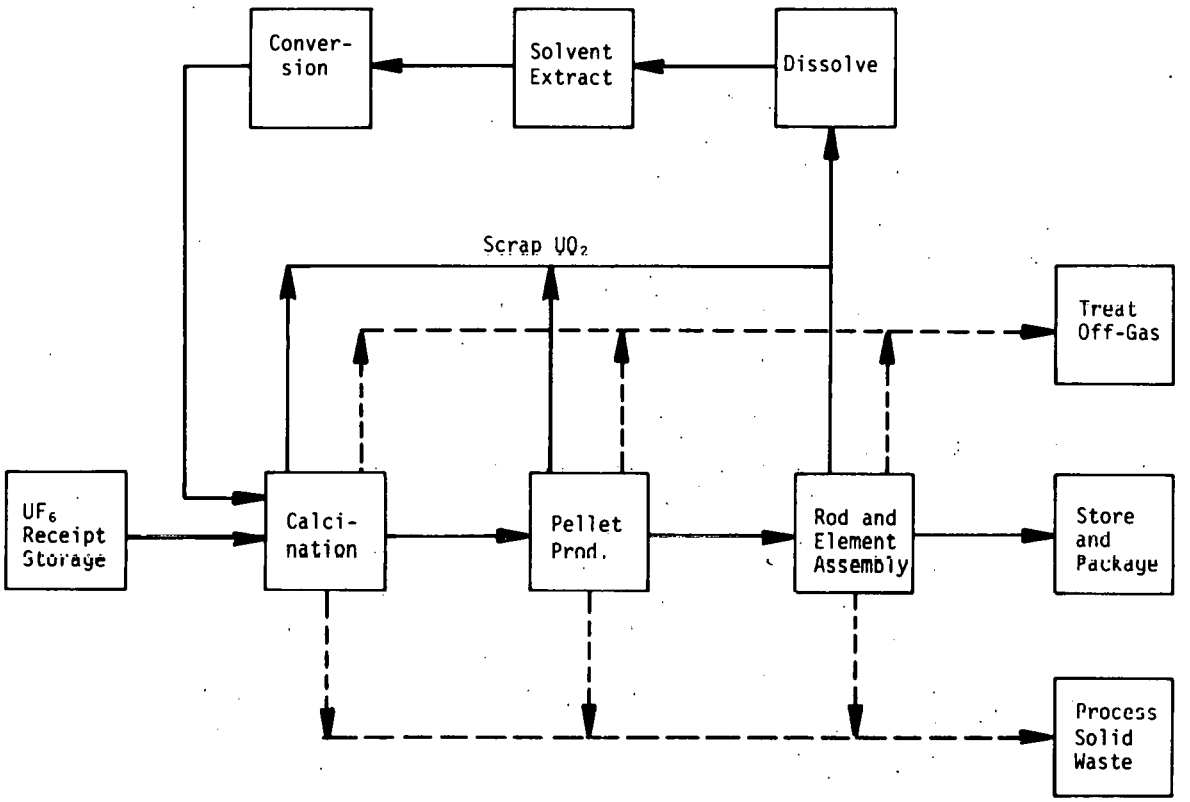


Fig. 5.23. Case 1.1.7 — Recycle of Uranium and Uranium-Plutonium Oxide (Coprocessed)

LEVEL 1



All functions are developed state of the art; material is a negligible radiation hazard.

Material convertibility is Category B for all steps except "Treat Off-Gas" and "Process Solid Waste."

Fig. 5.24. Case 1.1.7 — Recycle of Uranium and Uranium-Plutonium Oxide (Coprocessed); Slightly Enriched (2-4% ^{235}U) Uranium Fabrication (2.U)

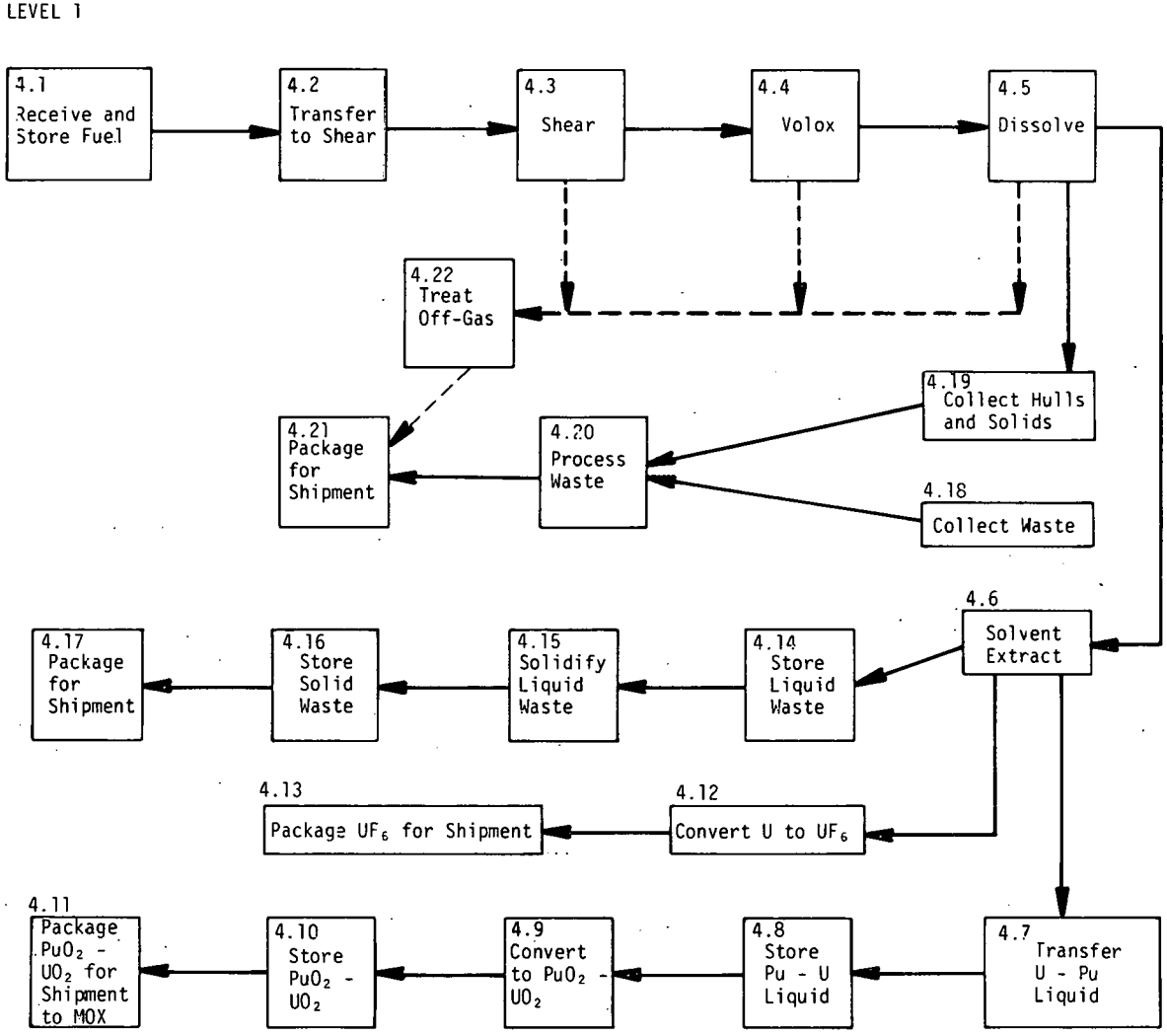


Fig. 5.25. Case 1.1.7 — Recycle of Uranium and Uramium-Plutonium Oxide (Coprocessed); Chemical Reprocessing (4.0)

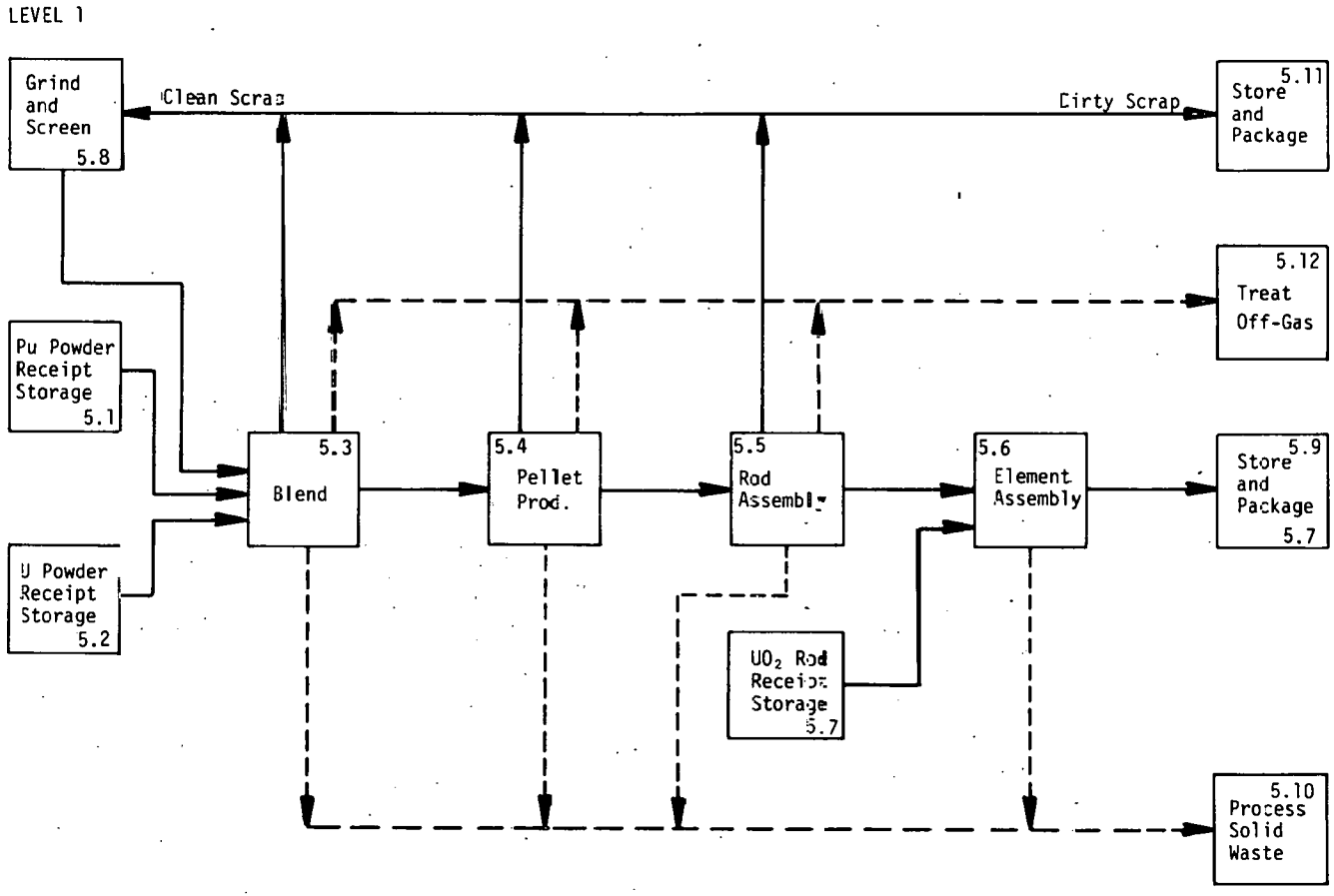


Fig. 5.26. Case 1.1.7 — Recycle of Uranium and Uranium-Plutonium Oxide (Coprocessed); MOX Fabrication (5.0)

Table 5.9. Case 1.1.7 (LWR): Recycle of Uranium and Uranium-Plutonium (Coprocessed)

	Process Step	State-of-the-Art	Material Description	Material Location	Radiation Hazard	Material Convertibility
A.	Chemical Reprocessing (4.0)					
4.1	Receive and Store Fuel	D	Irradiated Fuel	Basin	High	C
4.2	Transfer to Shear	D	Irradiated Fuel	Basin	High	C
4.3	Shear	D	Fuel Elements	Canyon	High	C
4.4	Voloxidation	HL	Fuel Element Pieces	Canyon	High	C
4.5	Dissolve	D	Fuel in Solution	Canyon	High	C
4.6	Solvent Extraction	D	Solution of U, Pu, Fission Products	Canyon	High	C
4.7	Transfer Pu-U Liquid	CP	Pu-U Nitrate	Canyon	Low	D
4.8	Store Pu-U	CP	Pu-U Nitrate	Canyon	Low	D
4.9	Convert to PuO_2-UO_2	CP	PuO_2-UO_2	Cell	Low	D
4.10	Store PuO_2-UO_2	D	PuO_2-UO_2	Cell	Low	D
4.11	Package for Shipment	D	PuO_2-UO_2	Cell	Low	D
4.12	Convert U to UF_6	D	U Nitrate	Cell	Negligible	B
4.13	Package UF_6	D	UF_6	Warehouse	Negligible	B
4.14	Store Liquid Waste	D	Radioactive Liquids	Canyon	High	
4.15	Solidify Liquid Waste	HDF	Radioactive Solids	Canyon	High	
4.16	Store Solid Waste	HDF	Radioactive Solids	Canyon	High	
4.17	Package for Shipment	CE	Radioactive Solids	Canyon	High	
4.18	Collect Waste	D	Radioactive Solids	Canyon	Variable	
4.19	Collect Hulls and Solids	D	Radioactive Solids	Canyon	High	
4.20	Process Waste	CE	Radioactive Solids	Canyon	High	
4.21	Package for Shipment	CE	Radioactive Solids	Cell	High	
4.22	Treat Off-Gas	HL	Radioactive Gases	Canyon	High	
B.	MOX Fabrication (5.0)					
5.1	Receive and Store Pu Powder	HDF	PuO_2	Cell	Low	D
5.2	Receive and Store U Powder	HDF	UO_2	Cell	Negligible	B
5.3	Blend	HDF	PuO_2/UO_2	Cell	Low	D
5.4	Produce Pellets	HDF	PuO_2/UO_2	Cell	Low	D
5.5	Assemble Rods	HDF	PuO_2/UO_2/Cladding	Cell	Low	D
5.6	Prepare Assemblies	HDF	PuO_2/UO_2/Cladding	Cell	Low	D
5.7	UO_2 Rod Storage	D	UO_2/Cladding	Warehouse	Negligible	B
5.8	Regrind and Screen Scrap	HDF	PuO_2/UO_2	Cell	Low	D
5.9	Store and Package	HDF	PuO_2/UO_2	Warehouse	Low	D
5.10	Process Solid Waste	HDF	Miscellaneous	Cell	Low	
5.11	Store and Package Scrap	HDF	Miscellaneous	Cell		
5.12	Treat Off-Gas	HDF	Pu	Cell		

The major advantages of this coprocessing process are compatibility with U.S. qualified and licensed fuel fabrication methods, some possible small economic advantage in fuel fabrication, and a reduction in the risk of successful diversion of plutonium by subnational groups. The mass of UO_2 - PuO_2 required for a given amount of plutonium is increased by about a factor of 8 over the spiked, separated plutonium (Case 1.1.2), but is less than the corresponding mass of irradiated fuel by a factor of 12. Therefore, the advantage of this fuel cycle is related to domestic safeguards or to internationally controlled reprocessing centers.

Preliminary study of the uranium-plutonium coprocessing has indicated that the Pu-U mixture obtained from solvent extraction for coprecipitation (Figure 5.25) should contain 12% Pu for LWR. The mixture will be diluted to about 5% PuO_2 (Figure 5.26) at the MOX facility for fabrication into LWR fuel. Higher concentrations may be required for LMFBR fuel processing. If further study permits increasing the plutonium concentration beyond 12%, some cost-savings would be expected.

5.2.8 Case 2.1.6 (GCR): Fuel Throwaway - Metal Fuel (Magnox)

Level 0 segments are shown schematically in Figure 5.27; Level 1 steps in Figures 5.28 and 5.29. The state-of-the-art and material characteristics for this case are shown in Table 5.10.

The processes described in this case are for a throwaway fuel cycle which is representative of the British gas-cooled reactor (Calder Hall type). This reactor concept is now considered obsolete and non-competitive by the U.S. because of the low power-density inherent with natural uranium metal fuel (Magnox-clad) cooled by carbon dioxide gas. A typical reactor power level is only 200 MW_{th} (35 MW_e). Although the case is defined as fuel throwaway, the British have reprocessed fuel from Calder Hall reactors using the basic Purex process.

5.2.9 Case 2.1.7 (GCR) Recycle of Uranium and Plutonium in Advanced Gas Reactors (Oxide Fuel)

Level 0 segments are shown schematically in Figure 5.30; Level 1 steps, in Figures 5.31, 5.32, and 5.33. The state-of-the-art and material attractiveness for the chemical processing step (4.0) and the MOX fabrication step (5.0) are shown in Table 5.11.

LEVEL 0

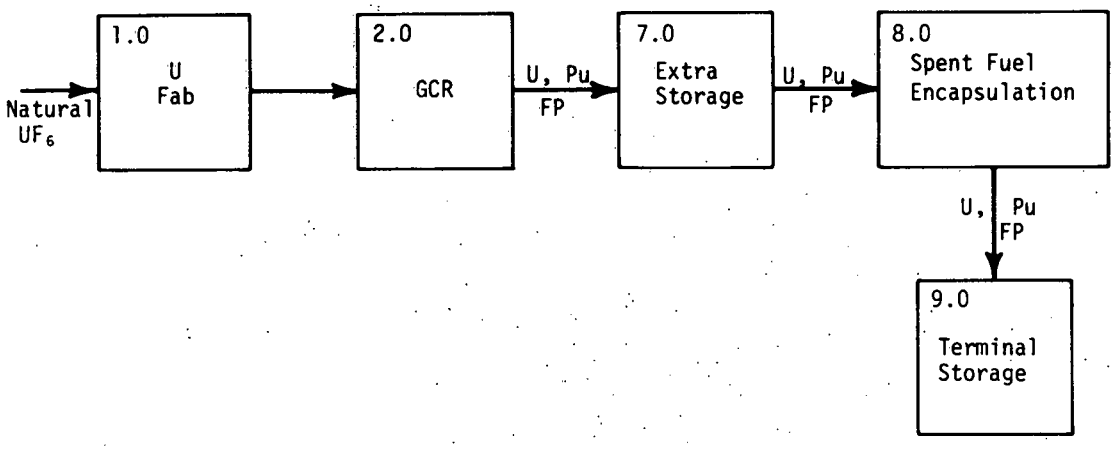
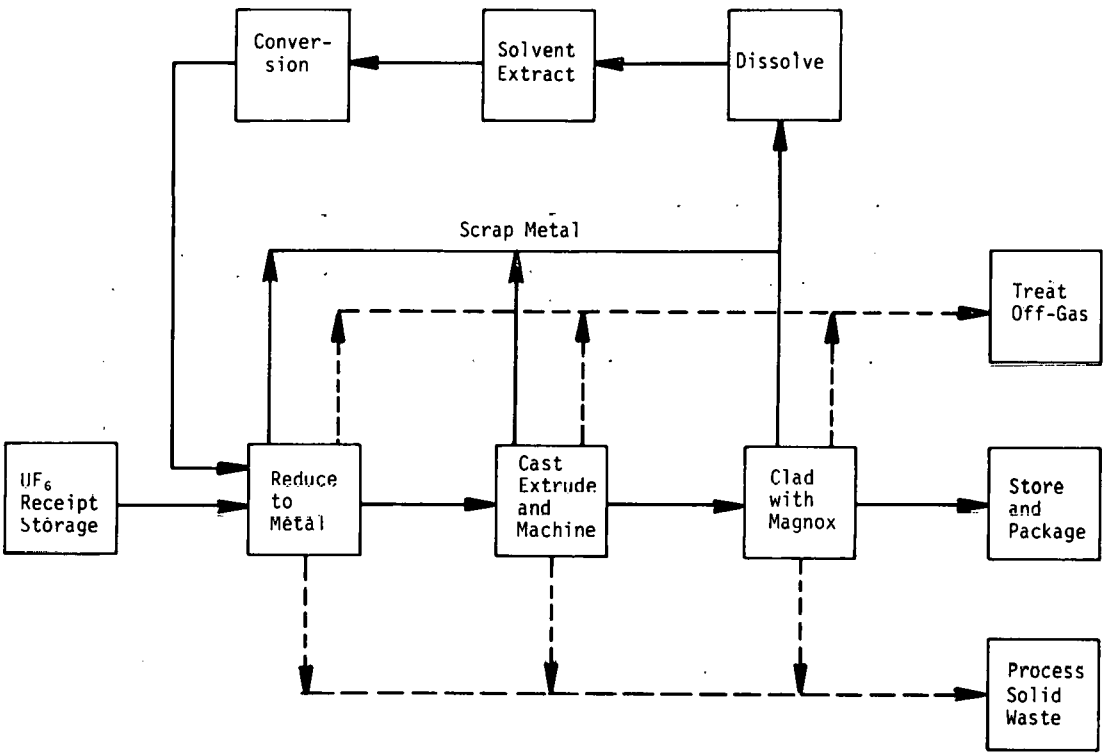


Fig. 5.27 Case 2.1.6 — Fuel Throwaway-Metal Fuel (Magnox)

LEVEL 1



All functions are developed state of the art, but are not demonstrated on a commercial scale.

Radiation hazard is negligible.

Material convertibility is Category B for all steps except "Treat Off-Gas" and "Process Solid Waste."

Fig. 5.28. Case 2.1.6 — Fuel Throwaway-Metal Fuel (Magnox); Natural Uranium Metal Fabrication (1.0)

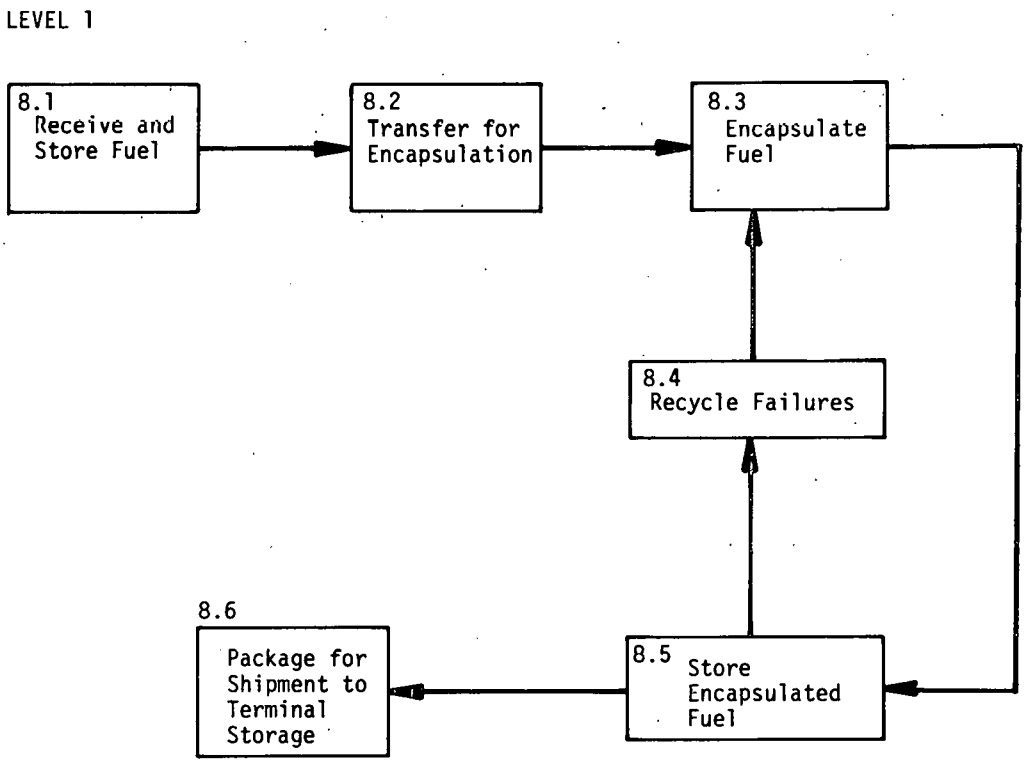


Fig. 5.29. Case 2.1.6 — Fuel Throwaway-Metal Fuel (Magnox); Encapsulated Fuel (8.0)

Table 5.10. Case 2.1.6 (GCR): Fuel Throwaway (Magnox)

Process Step	State-of-the-Art	Material Description	Material Location	Radiation Hazard	Material Convertibility
A. Encapsulation (8.0)					
8.1 Receive Fuel	D	Irradiated Magnox Fuel	Basin	High[a]	C[a]
8.2 Transfer to Encapsulation	D	Irradiated Fuel	Basin	High	C
8.3 Encapsulate Fuel Remove Gas	S[b]	Irradiated Fuel	Canyon	High	C
8.4 Store Encapsulated Fuel	D	Irradiated Fuel	Basin or Vault	High	C
8.5 Recycle Failures	S	Failed Irradiated Fuel	Canyon	High	C
8.6 Package for Shipment to Storage	CE	Irradiated Fuel	Basin or Vault	High	C

[a]Varies with age of fuel.

[b]Some Canadian development work for CANDU fuel may be applicable.

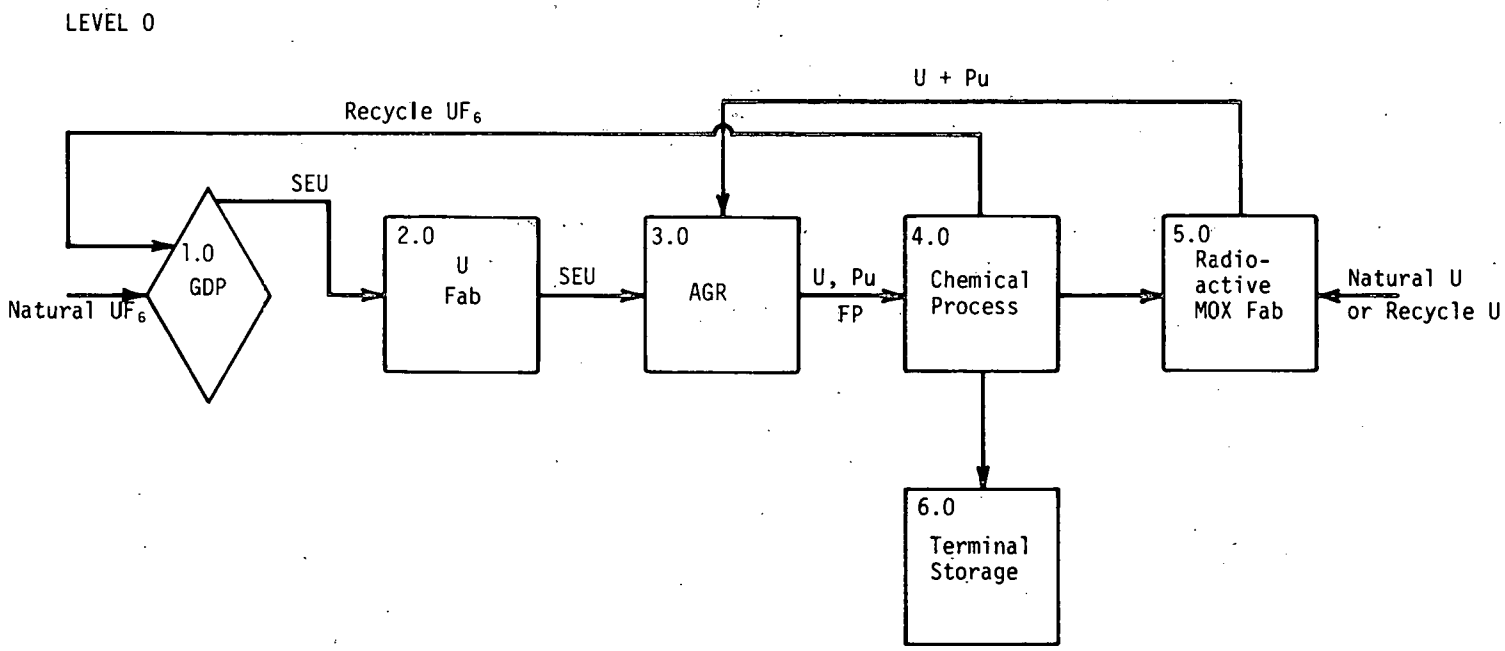
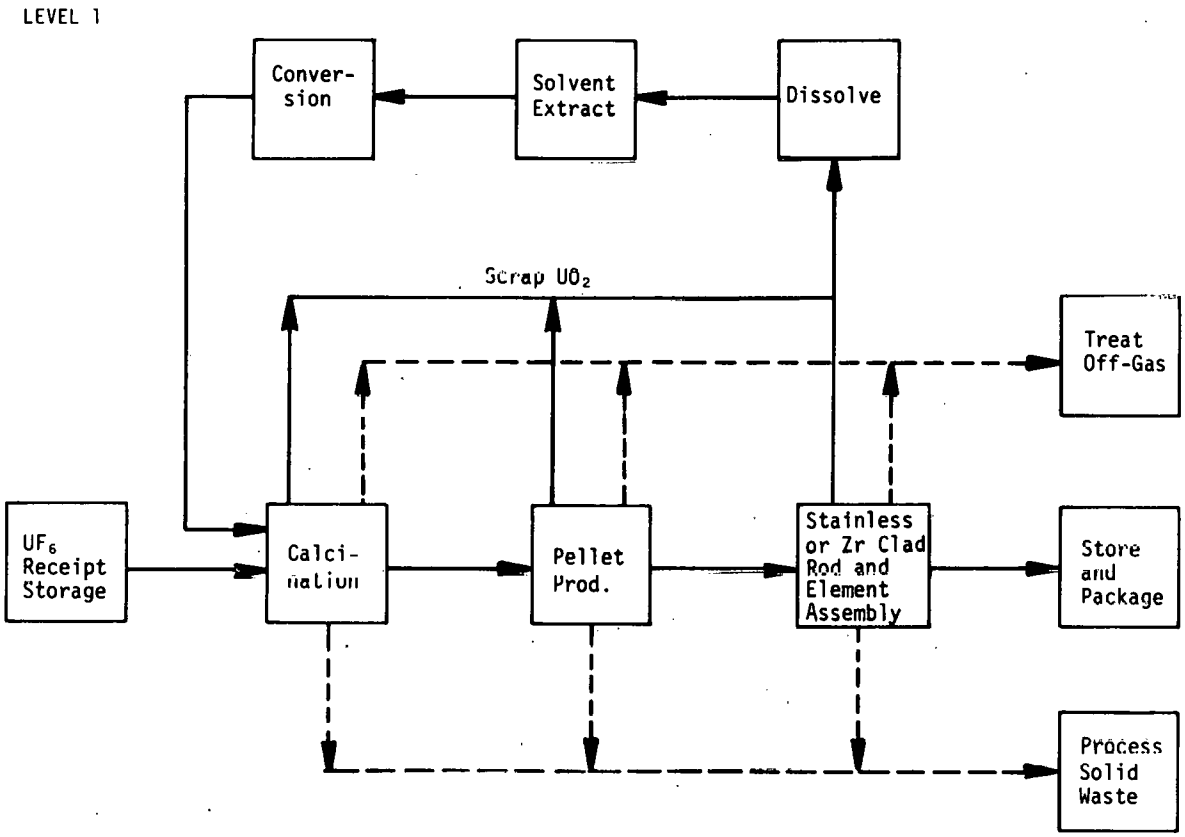


Fig. 5.30. Case 2.1.7 — Recycle of Uranium and Plutonium in AGR (Oxide Fuel)



All functions are developed state of the art; radiation hazard is negligible.

Material convertibility is Category B for all steps except "Treat Off-Gas" and "Process Solid Waste."

Fig. 5.31. Case 2.1.7 — Recycle of Uranium and Plutonium in AGR (Oxide Fuel); Slightly Enriched (2-4% ^{235}U) Uranium Fabrication (2.0)

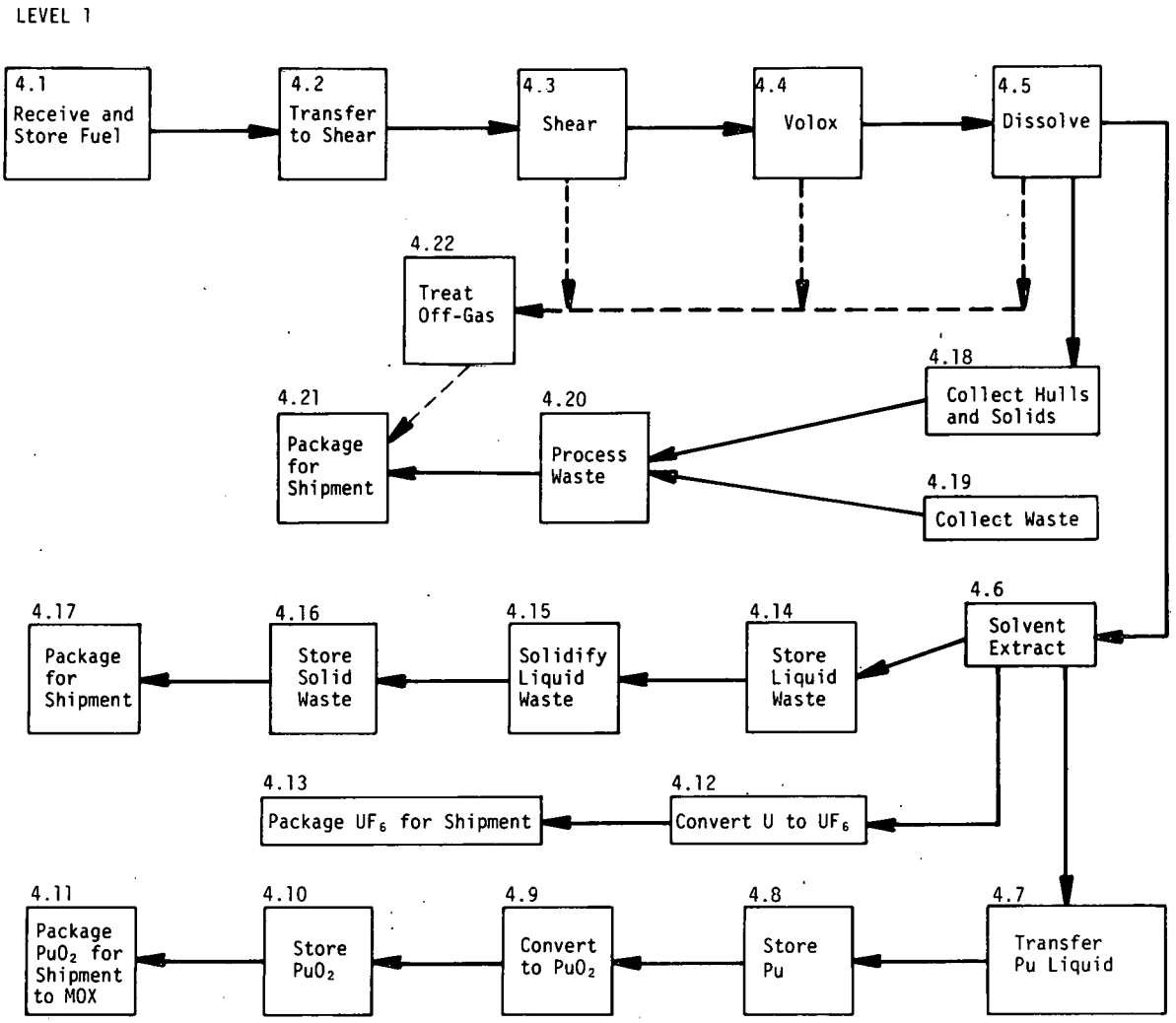


Fig. 5.32. Case 2.1.7 — Recycle of Uranium and Plutonium in AGR (Oxide Fuel); Chemical Reprocessing (4.0)

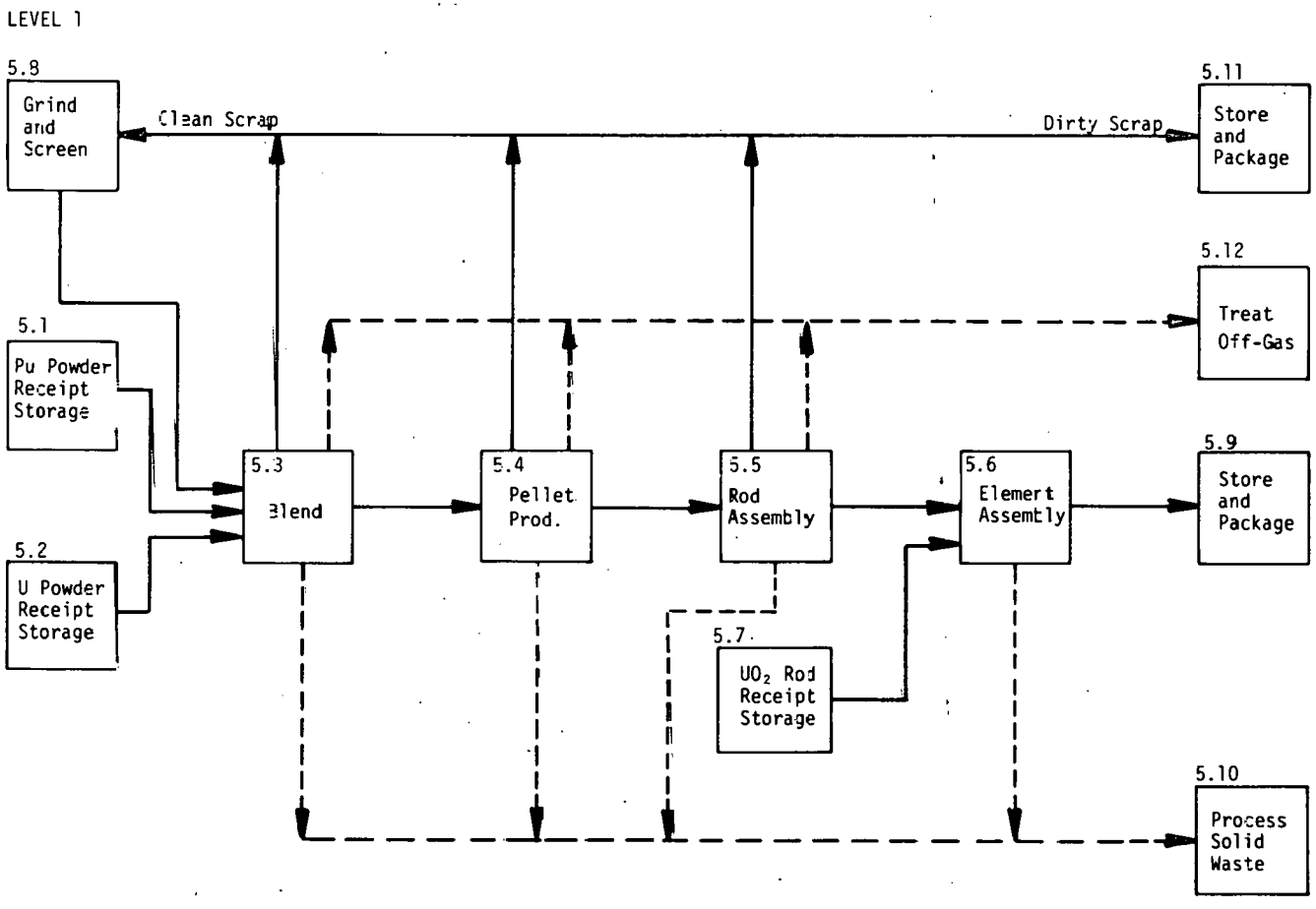


Fig. 5.33. Case 2.1.7 — Recycle of Uranium and Plutonium in AGR (Oxide Fuel); MOX Fabrication (5.0)

Table 5.11. Case 2.1.7 (GCR): Recycle of Uranium and Plutonium in AGR (Oxide Fuel)

	Process Step	State-of-the-Art	Material Description	Material Location	Radiation Hazard	Material Convertibility
A.	Chemical Reprocessing (4.0)					
4.1	Receive and Store Fuel	D	Irradiated Fuel	Basin	High	C
4.2	Transfer to Shear	D	Irradiated Fuel	Basin	High	C
4.3	Shear	D	Fuel Elements	Canyon	High	C
4.4	Voloxidation	HL	Fuel Element Pieces	Canyon	High	C
4.5	Dissolve	D	Fuel in Solution	Canyon	High	C
4.6	Solvent Extraction	D	Solution of U, Pu, Fission Products	Canyon	High	C
4.7	Transfer Pu Liquid	D	Pu Nitrate	Canyon	Low	D
4.8	Store Pu	D	Pu Nitrate	Canyon	Low	D
4.9	Convert to PuO_2	HE	PuO_2	Cell	Low	D
4.10	Store PuO_2	D	PuO_2	Cell	Low	D
4.11	Package for Shipment	D	PuO_2	Cell	Low	D
4.12	Convert U to UF_6	D	U Nitrate	Cell	Negligible	B
4.13	Package UF_6	D	UF_6	Warehouse	Negligible	B
4.14	Store Liquid Waste	D	Radioactive Liquids	Canyon	High	
4.15	Solidify Liquid Waste	HDF	Radioactive Solids	Canyon	High	
4.16	Store Solid Waste	HDF	Radioactive Solids	Canyon	High	
4.17	Package for Shipment	CE	Radioactive Solids	Canyon	High	
4.18	Collect Waste	D	Radioactive Solids	Canyon	Variable	
4.19	Collect Hulls and Solids	D	Radioactive Solids	Canyon	High	
4.20	Process Waste	CE	Radioactive Solids	Canyon	High	
4.21	Package for Shipment	CE	Radioactive Solids	Cell	High	
4.22	Treat Off-Gas	HL	Radioactive Gases	Canyon	High	
B.	MOX Fabrication (5.0)					
5.1	Receive and Store Pu Powder	HDF	PuO_2	Cell	Low	D
5.2	Receive and Store U Powder	HDF	UO_2	Cell	Negligible	B
5.3	Blend	HDF	PuO_2/UO_2	Cell	Low	D
5.4	Produce Pellets	HDF	PuO_2/UO_2	Cell	Low	D
5.5	Assemble Rods	HDF	PuO_2/UO_2/Cladding	Cell	Low	D
5.6	Prepare Assemblies	HDF	PuO_2/UO_2/Cladding	Cell	Low	D
5.7	UO_2 Rod Storage	D	UO_2/Cladding	Warehouse	Negligible	B
5.8	Regrind and Screen Scrap	HDF	PuO_2/UO_2	Cell	Low	
5.9	Store and Package	HDF	PuO_2/UO_2	Warehouse	Low	
5.10	Process Solid Waste	HDF	Miscellaneous	Cell	Low	
5.11	Store and Package Scrap	HDF	Miscellaneous	Cell		
5.12	Treat Off-Gas	HDF	Pu	Cell		

The processes described in this case include recycle of uranium and plutonium discharged from an advanced gas-cooled reactor (AGR). A typical example of an AGR, the Hinkley B Station, is rated at 1500 MW_{th} (621 MW_e). The fuel is slightly-enriched uranium oxide pellets (~2% ^{235}U) clad in stainless steel; typical fuel exposure is 18,000 MWD/Tonne. Fuel coolant is carbon dioxide gas and the reactor core structure is graphite. The fuel cycle is similar to the LWR cycle except for the lower enrichment and enhanced plutonium conversion ratio with graphite (less ore consumed) with some savings offset by the lower fuel exposure. Reprocessing systems are very similar to the base recycle case for LWR. Even with the similarities in fuel cycle, introduction of a gas-cooled reactor into the U.S. power system would probably require many years for commercialization and licensing.

5.2.10 Case 3.1.1 (HWR): Recycle of Plutonium

Level 0 segments are shown schematically in Figure 5.34; Level 1 steps in Figures 5.35, 5.36, and 5.37. The state-of-the-art and material attractiveness of the chemical reprocessing step (4.0) and the MOX fabrication step (5.0) are shown in Table 5.12.

The CANDU reactor was developed in Canada for domestic use and for export. Current power rating of CANDU reactors is about 2200 MW_{th} (630 MW_e) and some advanced designs are projected to operate at 750 MW_e. The reactor can be fueled with natural uranium, but some economic benefit is calculated with slight enrichment (~1% ^{235}U) or with plutonium recycle. Higher capital costs are estimated for HWRs than for LWRs because of the cost of heavy water and engineering refinements to minimize heavy water leakage. The incentives and impediments for introduction of HWRs in the U.S. have been studied. Commercialization and licensing of the HWR design in the U.S. would probably require many years.

Under current economic conditions in the U.S., reprocessing and recycle of plutonium from HWRs is probably not attractive. Other countries lacking uranium ore and enrichment facilities (e.g., Brazil) plan on plutonium recovery. The processes for plutonium recovery are very similar to those of the base case for recycle of LWR fuel.

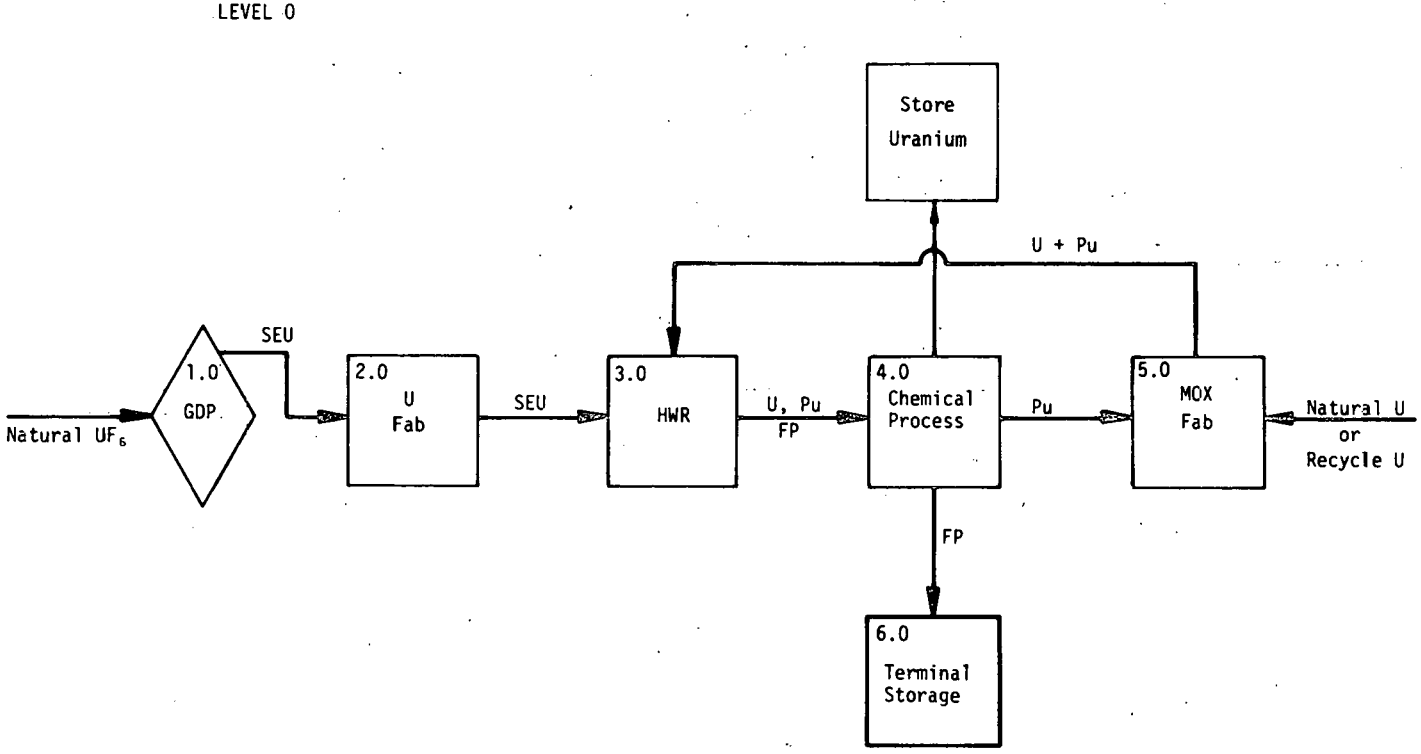
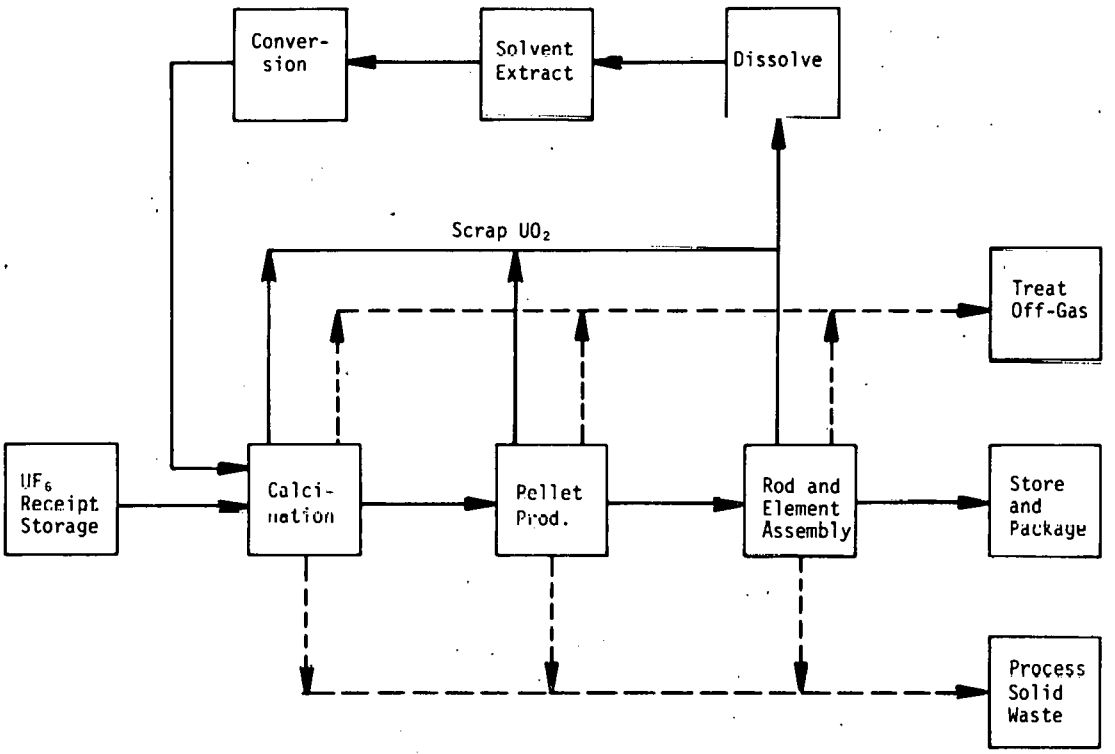


Fig. 5.34. Case 3.1.1 — Recycle of Plutonium

LEVEL 1



All functions are developed state of the art in Canada; radiation level is negligible.

Material convertibility is Category B for all steps except "Treat Off-Gas" and "Process Solid Waste."

Fig. 5.35. Case 3.1.1 — Recycle of Plutonium; Slightly Enriched (0.7-1.5% ^{235}U) Uranium Fabrication (2.0)

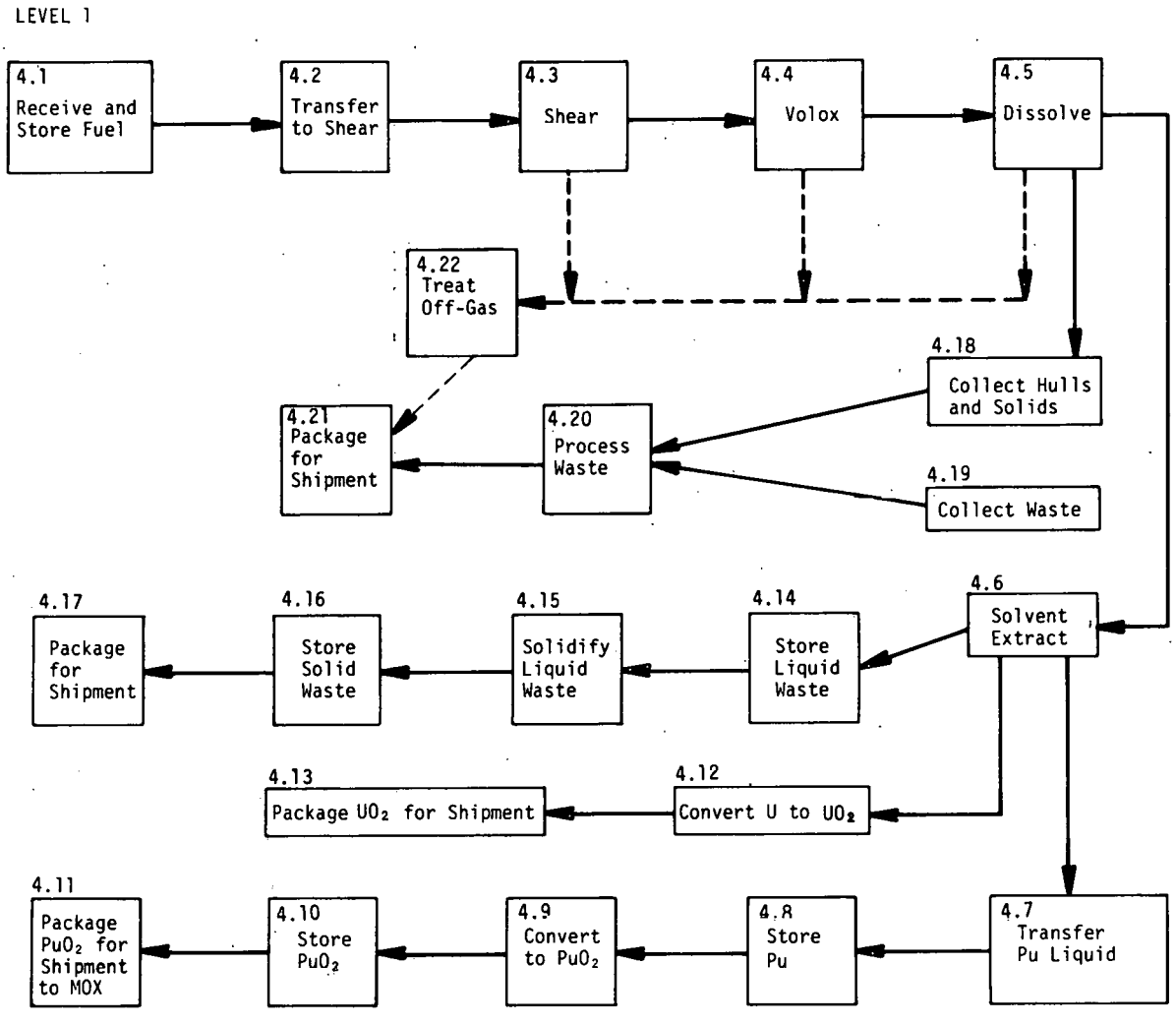


Fig. 5.36. Case 3.1.1 — Recycle of Plutonium; Chemical Reprocessing (4.0)

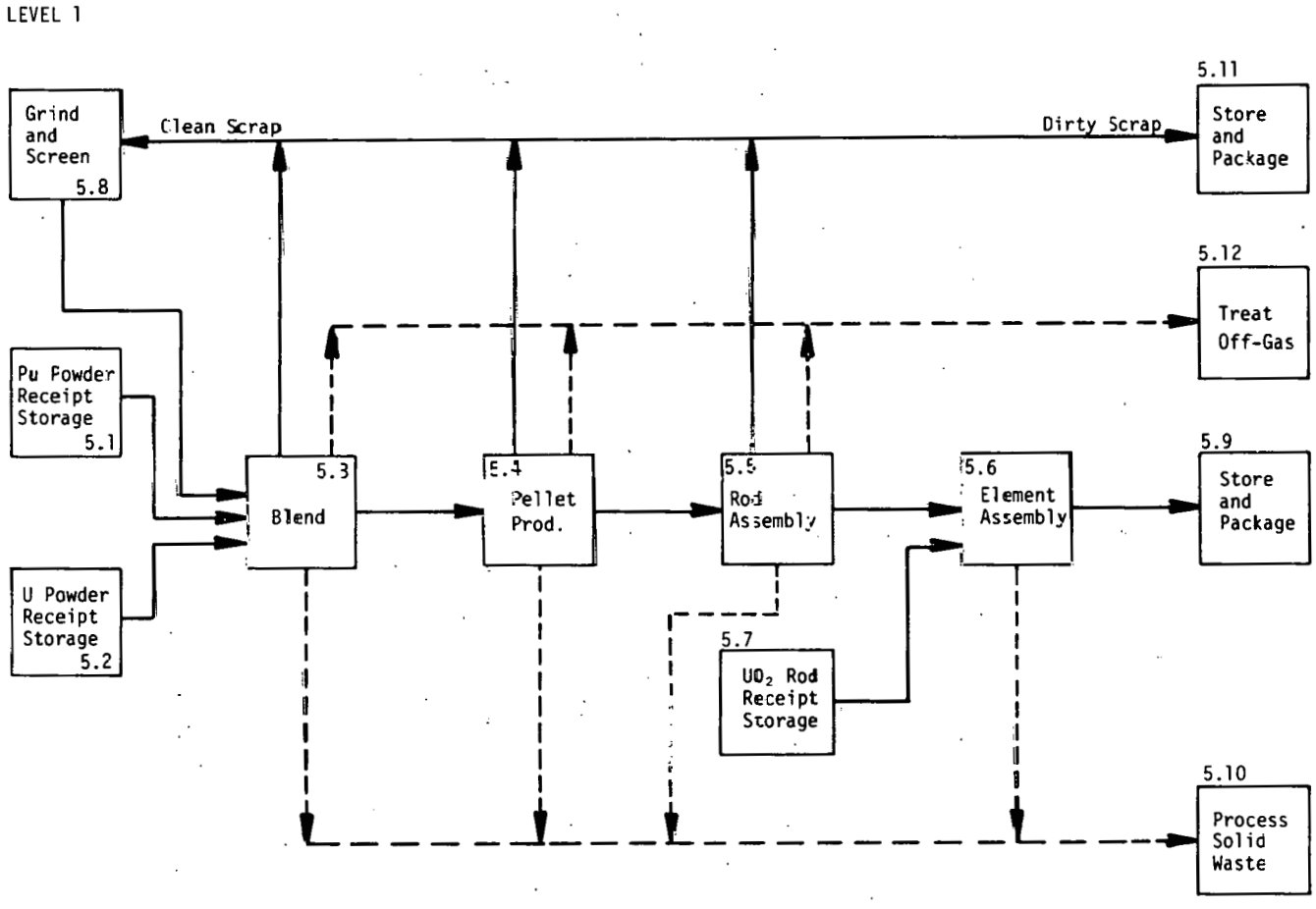


Fig. 5.37. Case 3.1.1 – Recycle of Plutonium; MOX Fabrication (5.0)

Table 5.12. Case 3.1.1 (HWR): Recycle of Plutonium

	Process Step	State-of-the-Art	Material Description	Material Location	Radiation Hazard	Material Convertibility
A.	Chemical Reprocessing (4.0)					
4.1	Receive and Store Fuel	D	Irradiated Fuel	Basin	High	C
4.2	Transfer to Shear	D	Irradiated Fuel	Basin	High	C
4.3	Shear	D	Fuel Elements	Canyon	High	C
4.4	Voloxidation	HL	Fuel Element Pieces	Canyon	High	C
4.5	Dissolve	D	Fuel in Solution	Canyon	High	C
4.6	Solvent Extraction	D	Solution of U, Pu, Fission Products	Canyon	High	C
4.7	Transfer Pu Liquid	D	Pu Nitrate	Canyon	Low	D
4.8	Store Pu	D	Pu Nitrate	Canyon	Low	D
4.9	Convert to PuO_2	HE	PuO_2	Cell	Low	D
4.10	Store PuO_2	D	PuO_2	Cell	Low	D
4.11	Package for Shipment	D	PuO_2	Cell	Low	D
4.12	Convert U to UO_2	D	UO_2	Cell	Negligible	B
4.13	Package UO_2	D	UO_2	Warehouse	Negligible	B
4.14	Store Liquid Waste	D	Radioactive Liquids	Canyon	High	
4.15	Solidify Liquid Waste	HDF	Radioactive Solids	Canyon	High	
4.16	Store Solid Waste	HDF	Radioactive Solids	Canyon	High	
4.17	Package for Shipment	CE	Radioactive Solids	Canyon	High	
4.18	Collect Waste	D	Radioactive Solids	Canyon	Variable	
4.19	Collect Hulls and Solids	D	Radioactive Solids	Canyon	High	
4.20	Process Waste	CE	Radioactive Solids	Canyon	High	
4.21	Package for Shipment	CE	Radioactive Solids	Cell	High	
4.22	Treat Off-Gas	HL	Radioactive Gases	Canyon	High	
B.	MOX Fabrication (5.0)					
5.1	Receive and Store Pu Powder	HDF	PuO_2	Cell	Low	D
5.2	Receive and Store U Powder	HDF	UO_2	Cell	Negligible	B
5.3	Blend	HDF	PuO_2/UO_2	Cell	Low	D
5.4	Produce Pellets	HDF	PuO_2/UO_2	Cell	Low	D
5.5	Assemble Rods	HDF	PuO_2/UO_2/Cladding	Cell	Low	D
5.6	Prepare Assemblies	HDF	PuO_2/UO_2/Cladding	Cell	Low	D
5.7	UO_2 Rod Storage	D	UO_2/Cladding	Warehouse	Negligible	B
5.8	Regrind and Screen Scrap	HDF	PuO_2/UO_2	Cell	Low	D
5.9	Store and Package	HDF	PuO_2/UO_2	Warehouse	Low	D
5.10	Process Solid Waste	HDF	Miscellaneous	Cell	Low	
5.11	Store and Package Scrap	HDF	Miscellaneous	Cell		
5.12	Treat Off-Gas	HDF	Pu	Cell		

The uranium recovered from HWR fuel is about 0.3% ^{235}U (somewhat higher with enriched fuel loading) and, hence, not economical for recycle. Storage of uranium for eventual use in breeder reactors is assumed.

5.2.11 Case 3.1.2 (HWR): Recycle of Spiked Plutonium

Level 0 segments are shown schematically in Figure 5.38; Level 1 steps, in Figures 5.39, 5.40, and 5.41. The state-of-the-art and material attractiveness for the chemical processing step (4.0) and the MOX fabrication step (5.0) are shown in Table 5.13.

Difficulties in providing an effective spiking method for plutonium fuel are discussed separately under Section 5.4, "Spiking Plutonium."

5.2.12 Case 3.1.3 (HWR): Fuel Throwaway — Oxide Fuel

Level 0 segments are shown in Figure 5.42; Level 1 steps, in Figures 5.43 and 5.44. The state-of-the-art and material attractiveness are shown in Table 5.14.

The CANDU reactors are currently operated in the fuel throwaway mode. Annual fuel throughput for HWRs is two or three times that of LWRs because the fuel exposure is only about 7,000 MWD/ton metal compared to about 30,000 MWD/ton for LWRs. This comparison is based on the same thermal output from both types of reactors. The conversion ratio of plutonium is higher in HWRs than for LWRs; thus, the stored HWR fuel would become a future resource should a decision be made to phase in breeder reactors as a consequence of high uranium cost or short supply.

5.2.13 Case 3.3.1 (HWR): Tandem Fuel — Reconstituted LWR Fuel

Level 0 segments are shown in Figure 5.45; Level 1 steps, in Figures 5.46, 5.47, and 5.48. The state-of-the-art and material characteristics for Operation 5.0 (Reconstitute Fuel for HWR) and Operation 8.0 (Spent Fuel Encapsulation) are shown in Table 5.15.

In the tandem fuel cycle, irradiated LWR fuel is reirradiated in HWRs without intermediate chemical reprocessing. The power recovered from a given amount of uranium is extended about 30%, and the fuel discharged from the HWR contains <0.5% ^{235}U and about 20% less plutonium

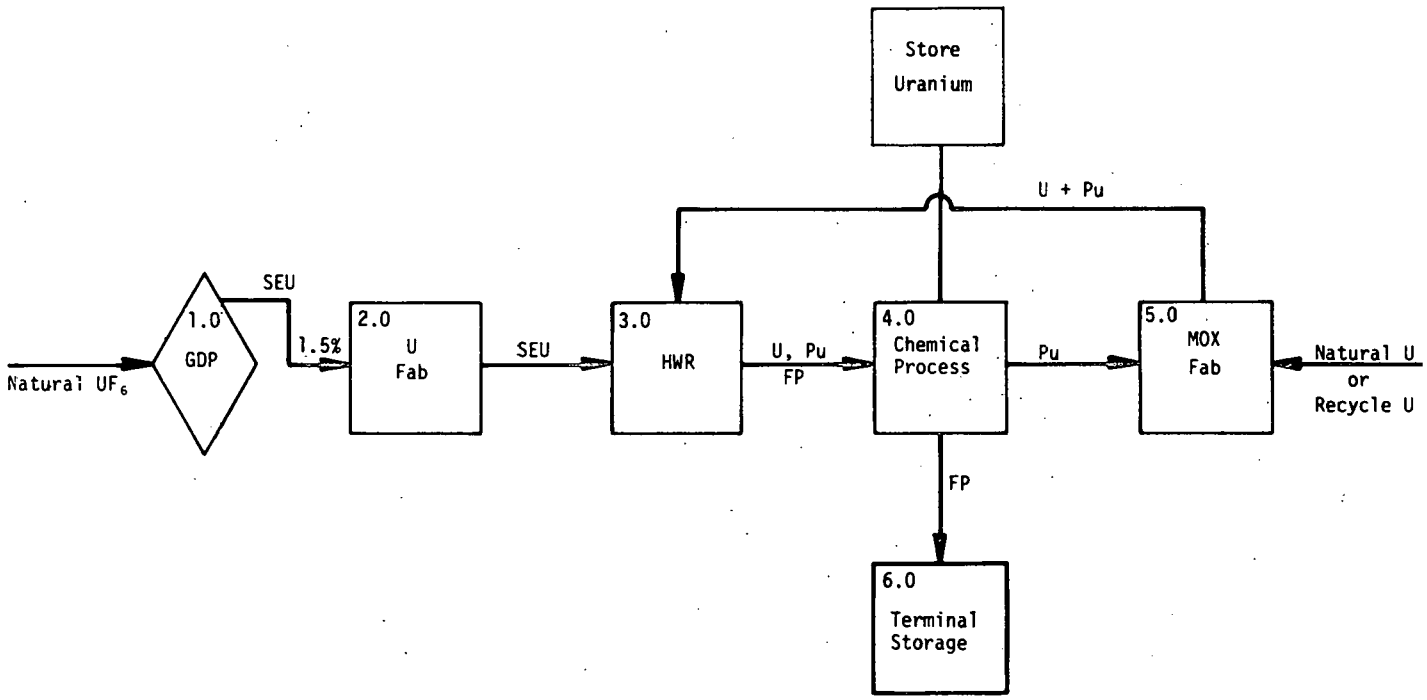
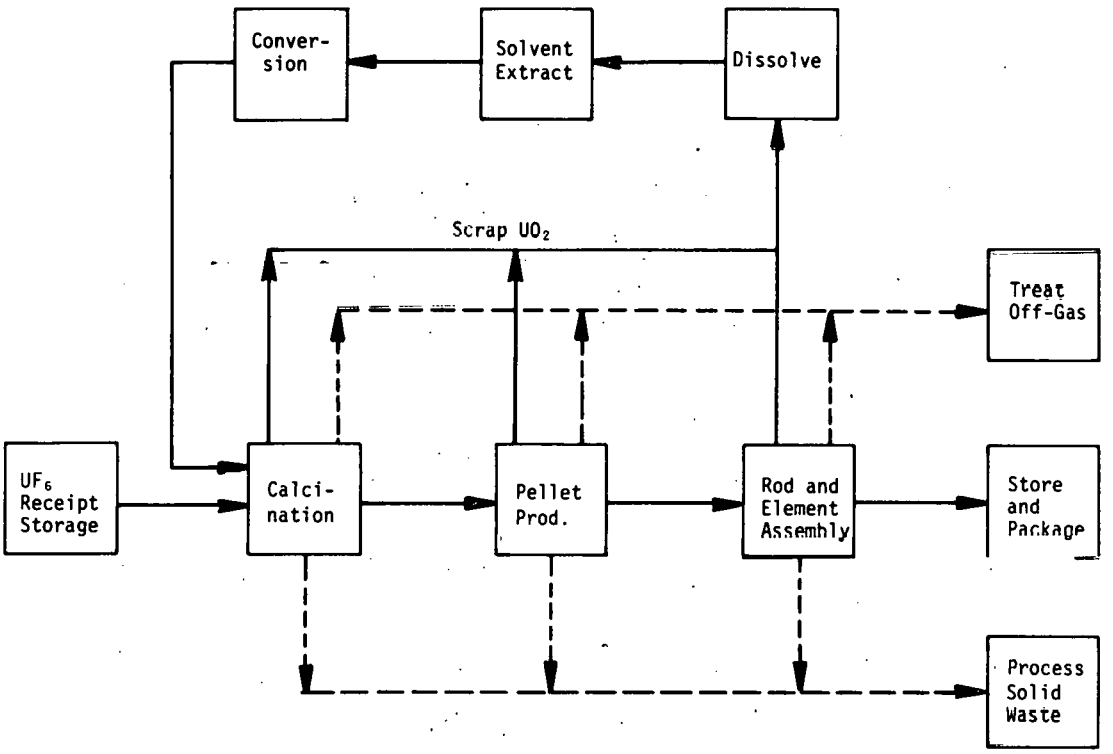


Fig. 5.38. Case 3.1.2 — Recycle of Spiked Plutonium

LEVEL 1



All functions are developed state of the art in Canada; radiation hazard is negligible.

Material convertibility is Category B for all steps except "Treat Off-Gas" and "Process Solid Waste."

Fig. 5.39. Case 3.1.2 — Recycle of Spiked Plutonium; Slightly Enriched (0.7-1.5% ^{235}U) Uranium Fabrication (2.0)

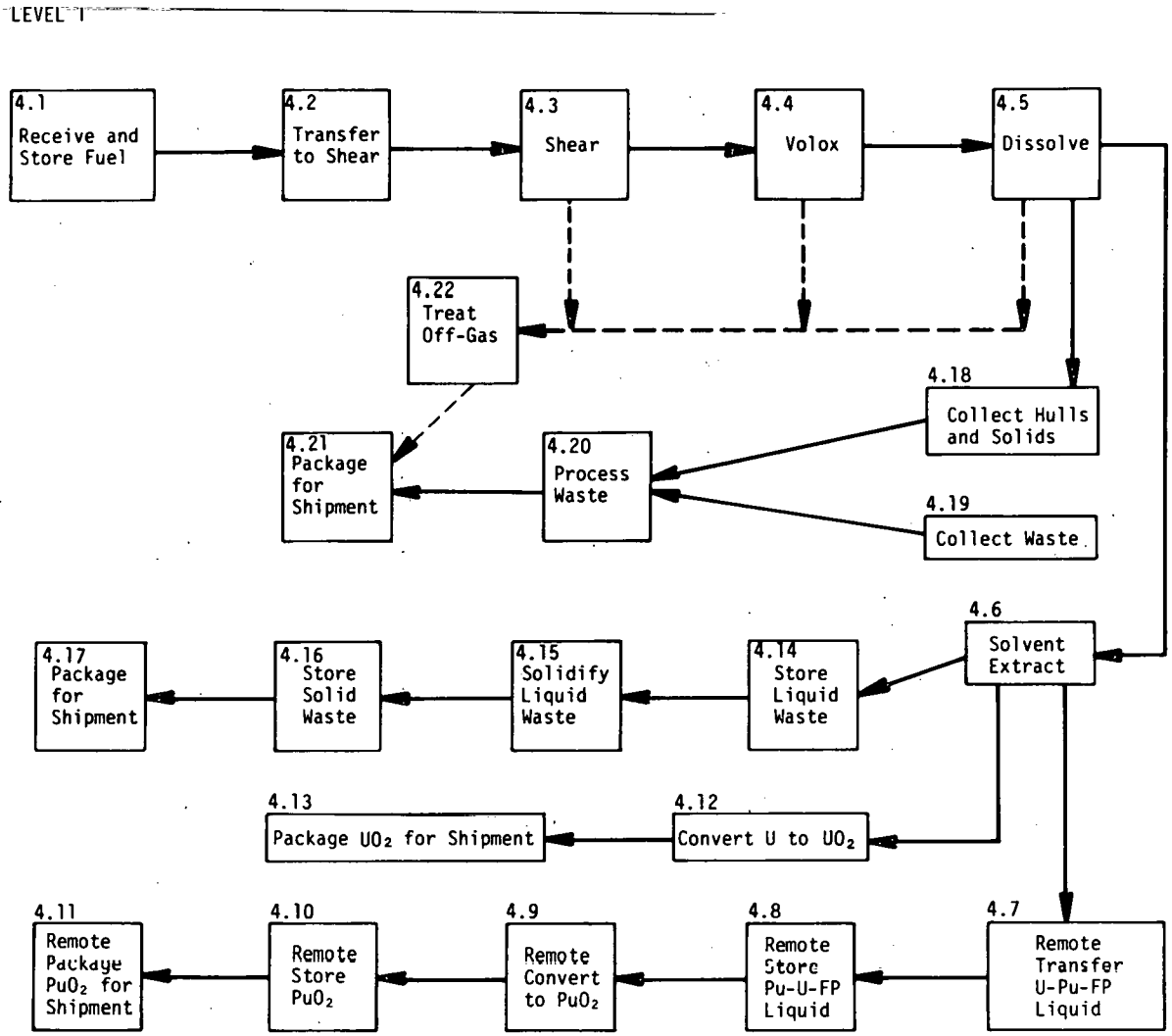


Fig. 5.40. Case 3.1.2 — Recycle of Spiked Plutonium; Chemical Reprocessing (4.0)

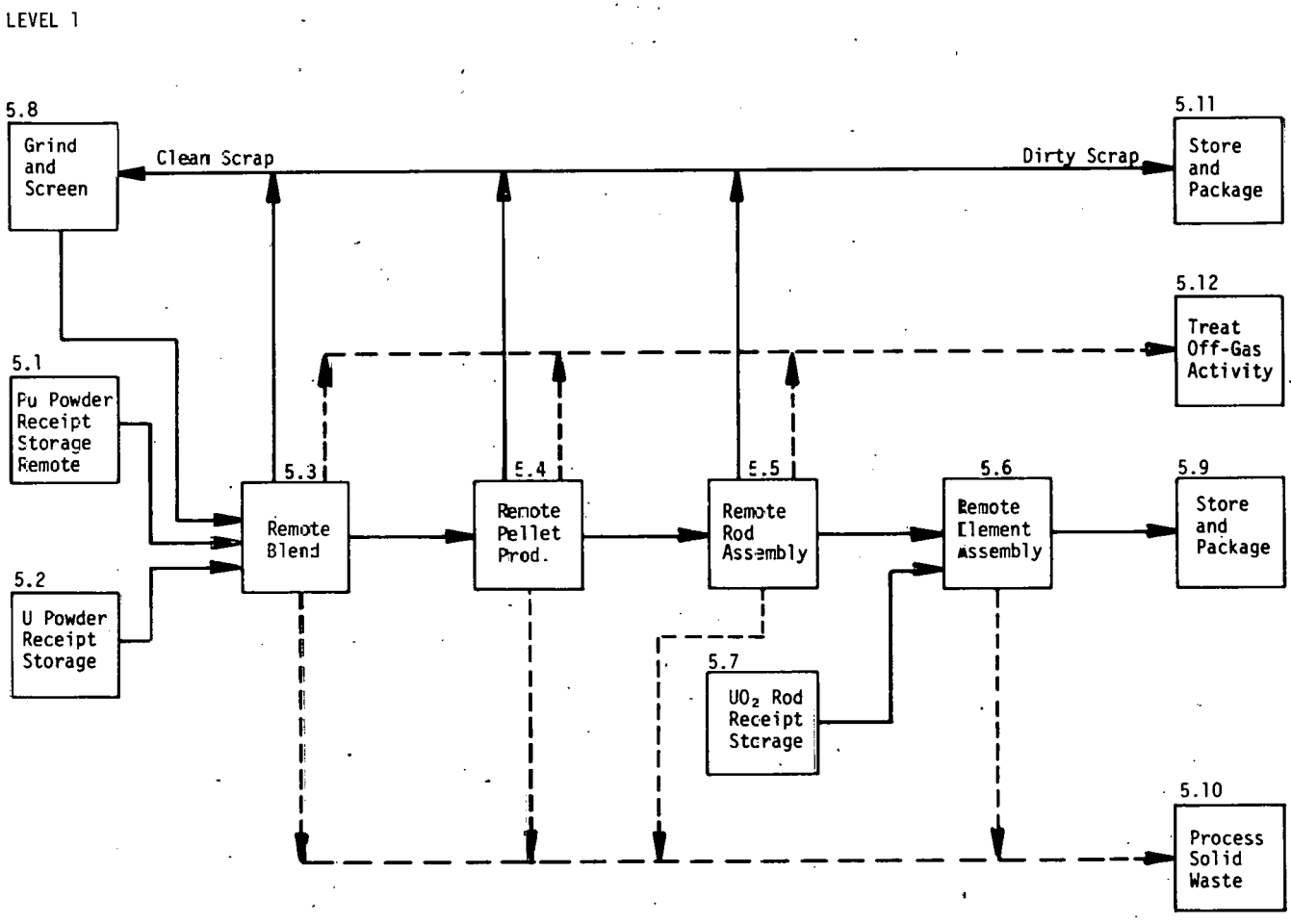


Fig. 5.41. Case 3.1.2 – Recycle of Spiked Plutonium; MOX Fabrication (5.0)

Table 5.13. Case 3.1.2 (HWR): Recycle of Spiked Plutonium

	Process Step	State-of-the-Art	Material Description	Material Location	Radiation Hazard	Material Convertibility
A.	Chemical Reprocessing (4.0)					
4.1	Receive and Store Fuel	D	Irradiated Fuel	Basin	High	C
4.2	Transfer to Shear	D	Irradiated Fuel	Basin	High	C
4.3	Shear	D	Fuel Elements	Canyon	High	C
4.4	Voloxidation	HL	Fuel Element Pieces	Canyon	High	C
4.5	Dissolve	D	Fuel in Solution	Canyon	High	C
4.6	Solvent Extraction	D	Solution of U, Pu, Fission Products	Canyon	High	C
4.7	Remote Transfer Pu Liquid/F.P.	D	Pu Nitrate/F.P.	Canyon	Medium	C
4.8	Remote Store Pu/F.P.	D	Pu Nitrate/F.P.	Canyon	Medium	C
4.9	Convert to PuO_2/F.P.	CP	PuO_2/F.P.	Cell	Medium	C
4.10	Store PuO_2/F.P.	CP	PuO_2/F.P.	Cell	Medium	C
4.11	Package for Shipment	CP	PuO_2/F.P.	Cell	Medium	C
4.12	Convert U to UF_6	D	U Nitrate	Cell	Negligible	B
4.13	Package UF_6	D	UF_6	Warehouse	Negligible	B
4.14	Store Liquid Waste	D	Radioactive Liquids	Canyon	High	
4.15	Solidify Liquid Waste	HDF	Radioactive Solids	Canyon	High	
4.16	Store Solid Waste	HDF	Radioactive Solids	Canyon	High	
4.17	Package for Shipment	CE	Radioactive Solids	Canyon	High	
4.18	Collect Waste	D	Radioactive Solids	Canyon	Variable	
4.19	Collect Hulls and Solids	D	Radioactive Solids	Canyon	High	
4.20	Process Waste	CE	Radioactive Solids	Canyon	High	
4.21	Package for Shipment	CE	Radioactive Solids	Cell	High	
4.22	Treat Off-Gas	HL	Radioactive Gases	Canyon	High	
B.	MOX Fabrication (5.0)					
5.1	Receive and Store Pu Powder	HDF	PuO_2/F.P.	Cell	Medium	D
5.2	Receive and Store U Powder	HDF	UO_2	Cell	Negligible	B
5.3	Blend	HDF	PuO_2/UO_2/F.P.	Cell	Medium	D
5.4	Produce Pellets	HDF	PuO_2/UO_2/F.P.	Cell	Medium	D
5.5	Assemble Rods	HDF	PuO_2/UO_2/F.P.	Cell	Medium	D
5.6	Prepare Assemblies	HDF	PuO_2/UO_2/F.P.	Cell	Medium	D
5.7	UO_2 Rod Storage	D	UO_2/Cladding	Warehouse	Negligible	B
5.8	Regrind and Screen Scrap	HDF	PuO_2/UO_2/F.P.	Cell	Medium	D
5.9	Store and Package	HDF	PuO_2/UO_2/F.P.	Warehouse	Medium	D
5.10	Process Solid Waste	HDF	Miscellaneous	Cell	Negligible	
5.11	Store and Package Scrap	HDF	Miscellaneous	Cell		
5.12	Treat Off-Gas	HDF	Pu	Cell		

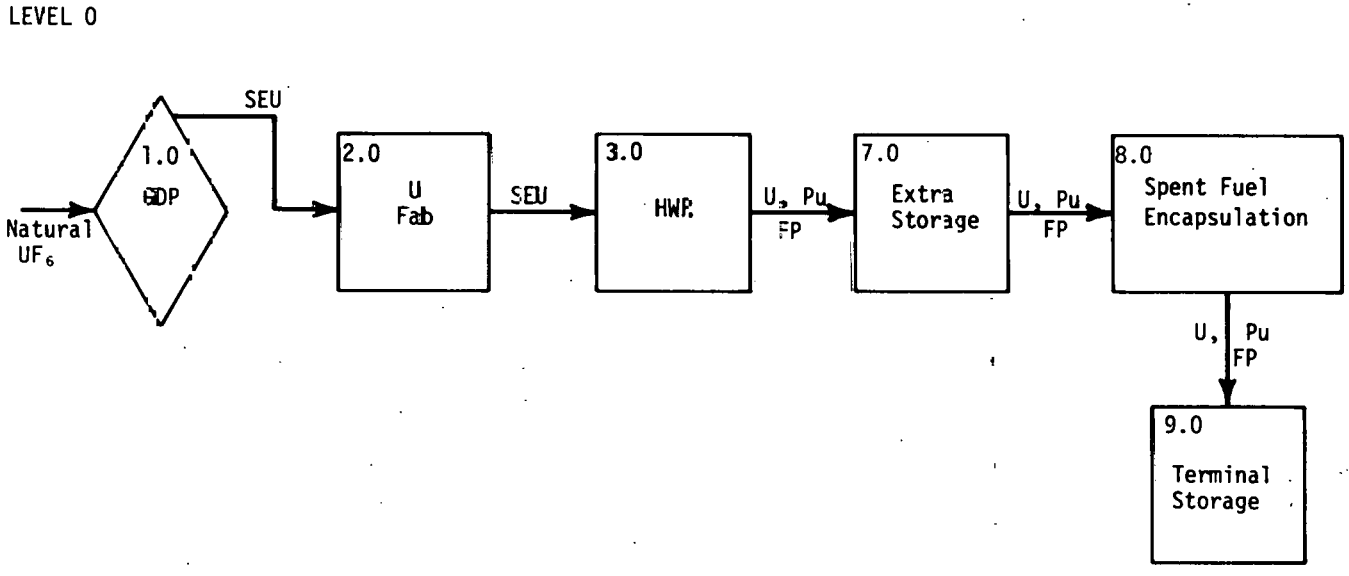
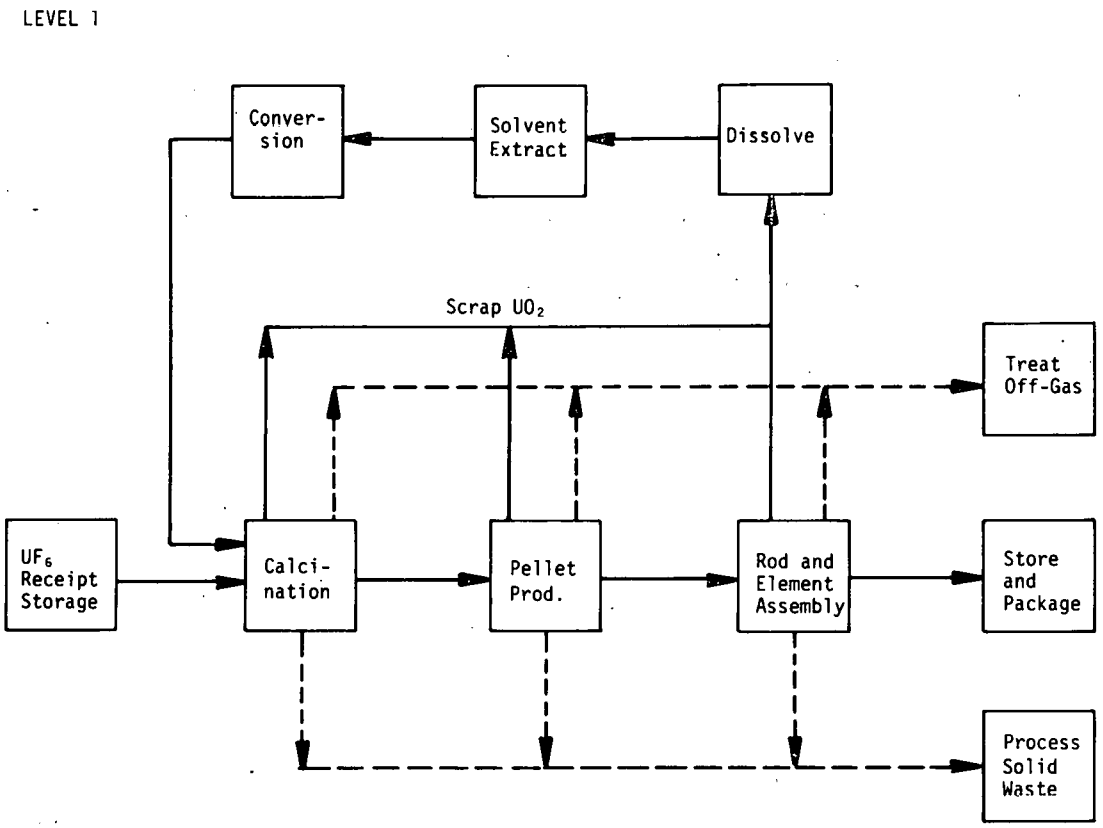


Fig. 5.42. Case 3.1.3 — Fuel Throwaway (Oxide Fuel)



All functions are developed state-of-the-art in Canada; radiation hazard is negligible.

Material convertibility is Category B for all steps except "Treat Off-Gas" and "Process Solid Waste."

Fig. 5.43. Case 3.1.3 — Fuel Throwaway (Oxide Fuel); Slightly Enriched (0.7-1.5% ^{235}U) Uranium Fabrication (2.0)

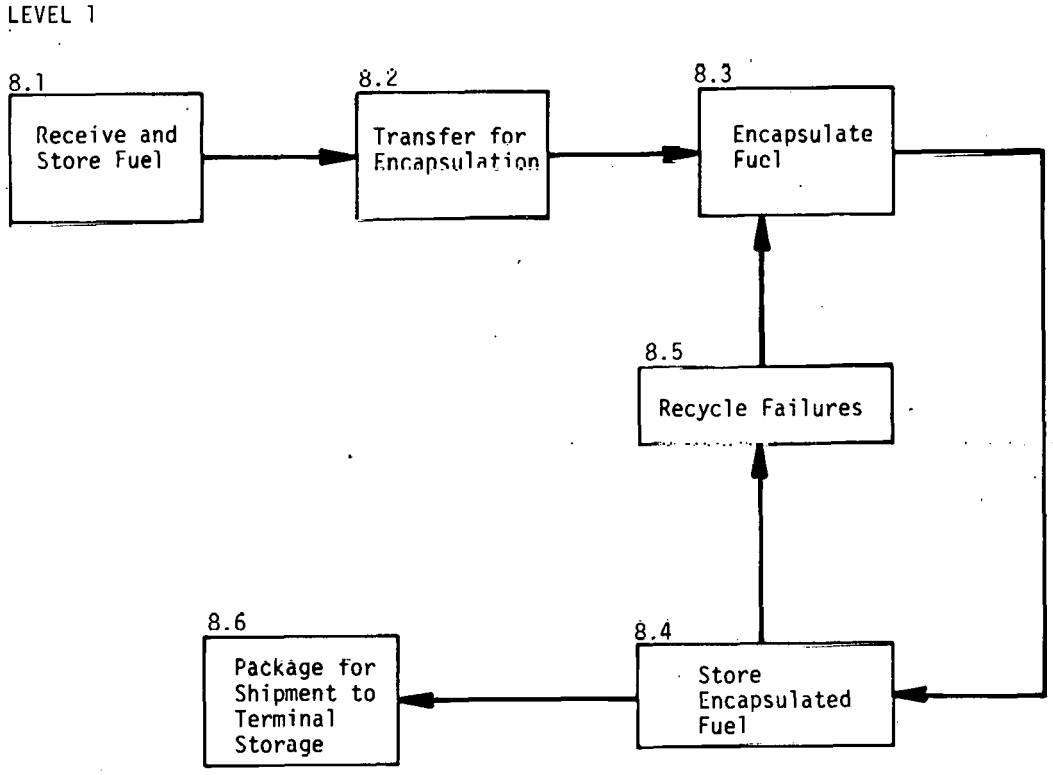


Fig. 5.44. Case 3.1.3 — Fuel Throwaway (Oxide Fuel); Spent Fuel Encapsulation (8.0)

Table 5.14. Case 3.1.3 (HWR): Fuel Throwaway (Oxide Fuel)

	Process Step	State-of-the-Art	Material Description	Material Location	Radiation Hazard	Material Convertibility
A.	Encapsulation (8.0)					
	8.1 Receive Fuel	D	Irradiated HWR Fuel	Basin	High[a]	C[a]
	8.2 Transfer to Encapsulation	D	Irradiated Fuel	Basin	High	C
	8.3 Encapsulate Fuel Remove Gas	S[b]	Irradiated Fuel	Canyon	High	C
	8.4 Store Encapsulated Fuel	D	Irradiated Fuel	Basin or Vault	High	C
	8.5 Recycle Failures	S	Failed Irradiated Fuel	Canyon	High	C
	8.6 Package for Shipment to Storage	CE	Irradiated Fuel	Basin or Vault	High	C

[a]Varies with age of fuel.

[b]Some Canadian development work for CANDU fuel may be applicable.

LEVEL 0

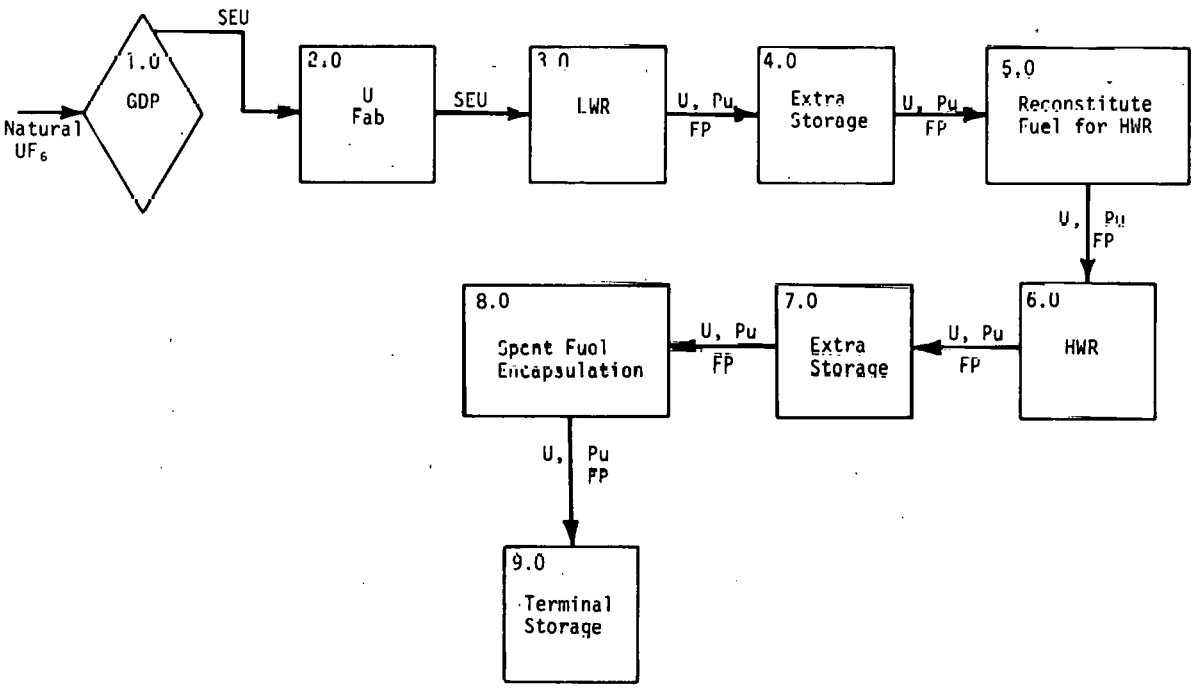
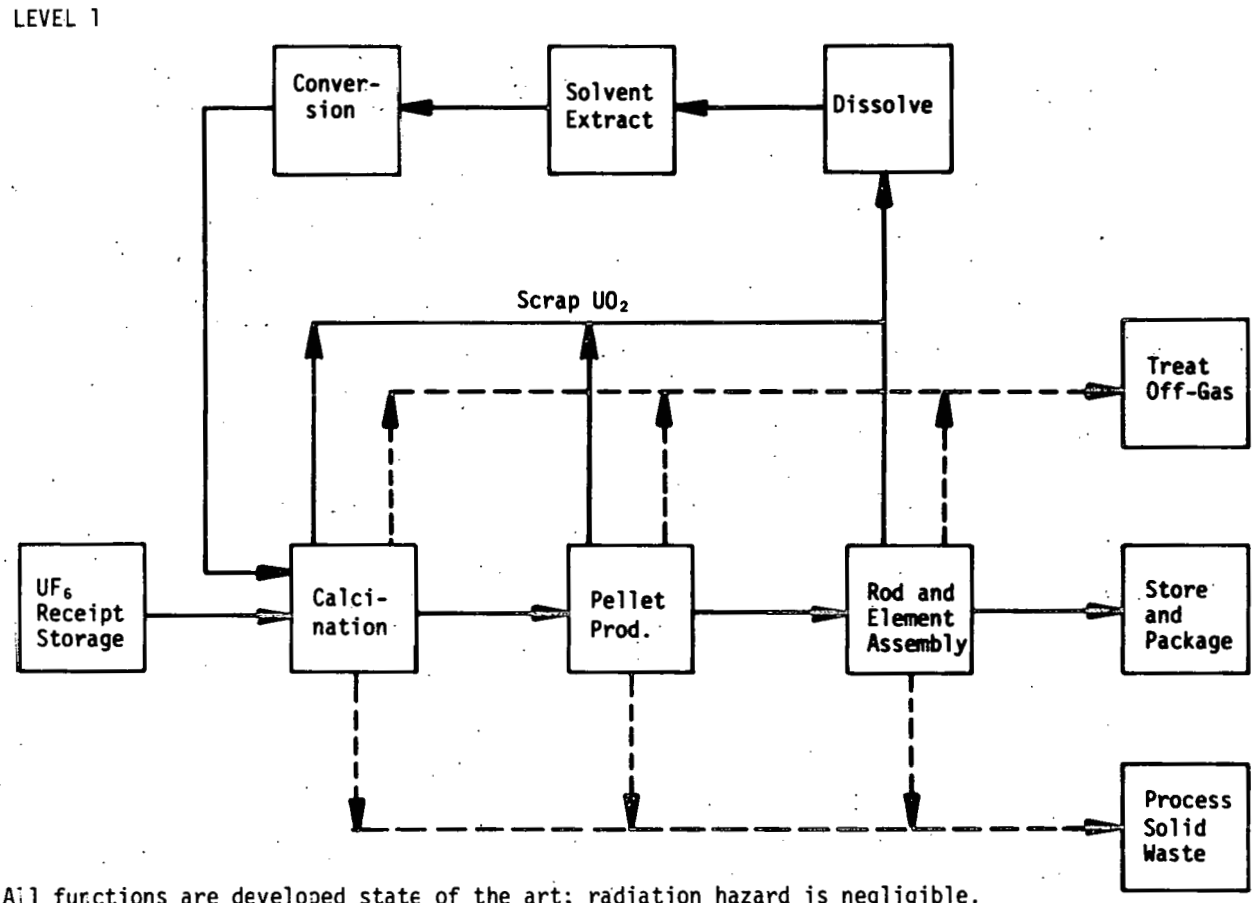


Fig. 5.45. Case 3.3.1 – Tandem Cycle - Reconstituted LWR Fuel



All functions are developed state of the art; radiation hazard is negligible.

Material convertibility is Category B for all steps except "Treat Off-Gas" and "Process Solid Waste."

Fig. 5.46. Case 3.3.1 — Tandem Cycle with Reconstituted LWR Fuel; Slightly Enriched (2-4% ^{235}U) Uranium Fabrication (2.0)

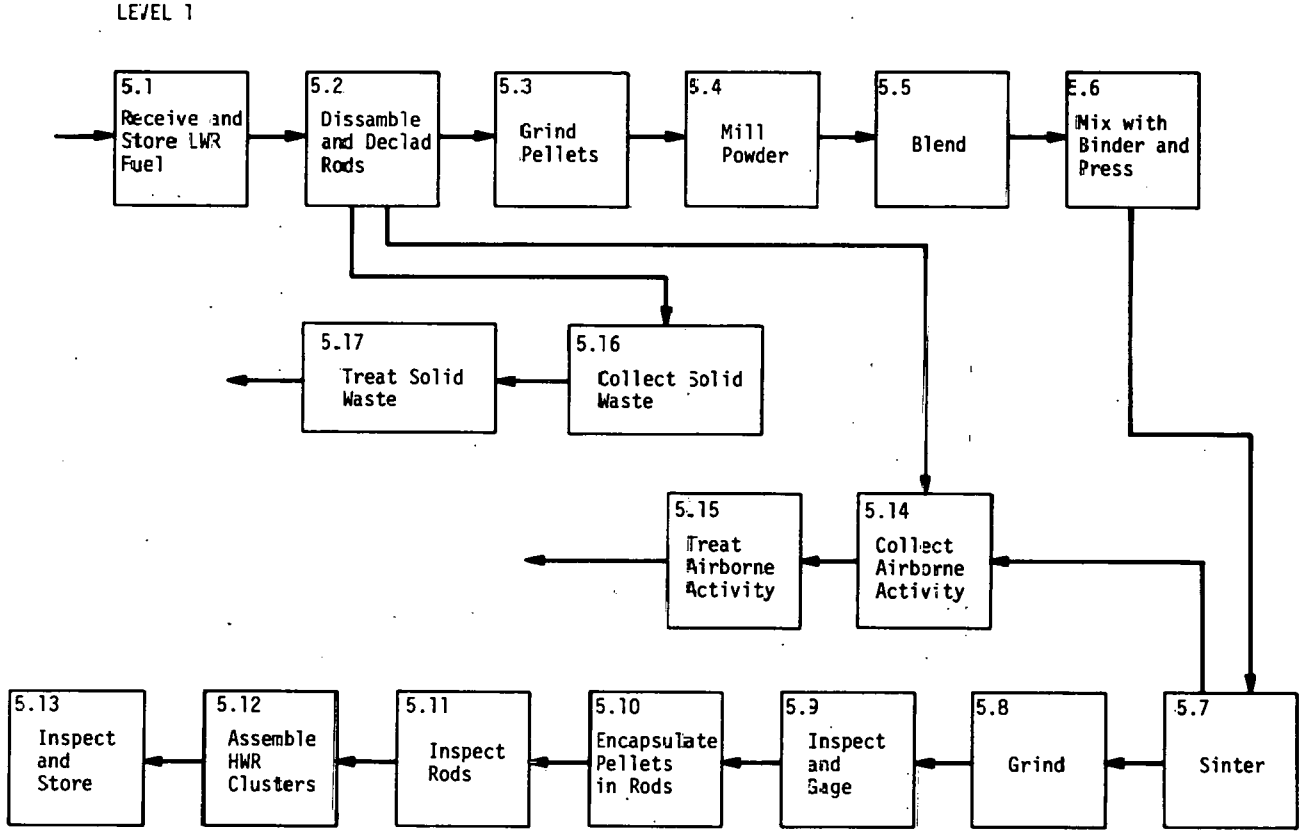


Fig. 5.47. Case 3.3.1 — Tandem Cycle with Reconstituted LWR Fuel; Reconstitute Fuel (5.6)

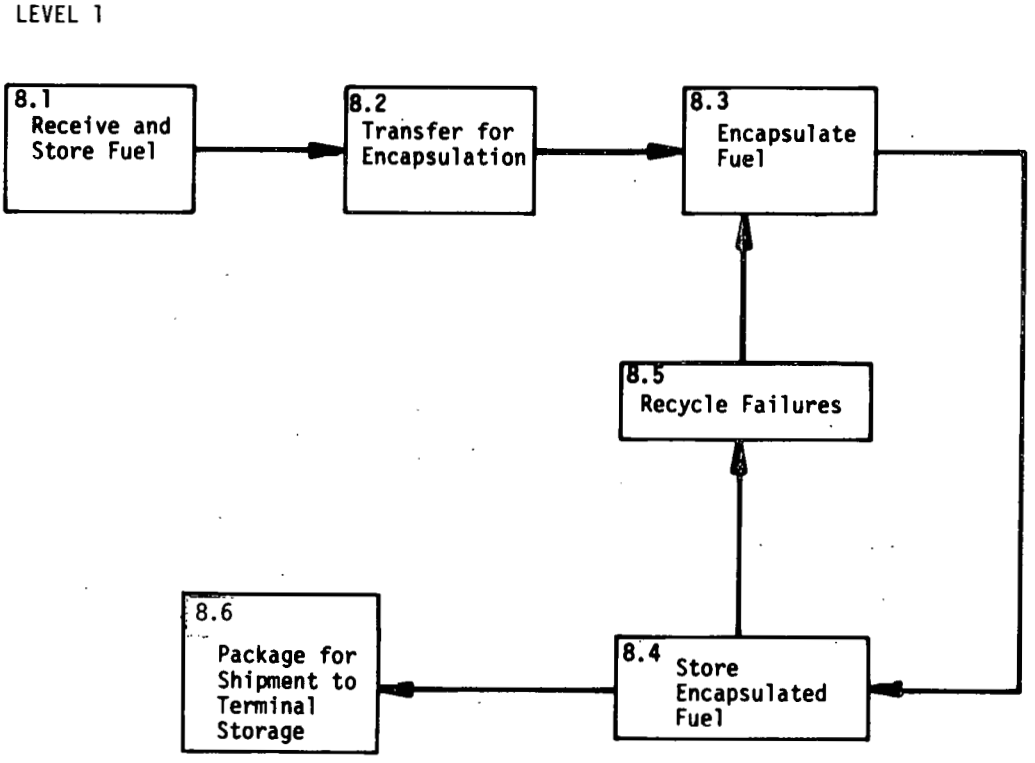


Fig. 5.48. Case 3.3.1 — Tandem Cycle with Reconstituted LWR Fuel; Spent Fuel Encapsulation (8.0)

Table 5.15. Case 3.3.1: Tandem Cycle - Reconstituted LWR Fuel

	Process Step	State-of-the-Art	Material Description	Material Location	Radiation Hazard	Material Convertibility
A.	Reconstitute Fuel (5.0)					
5.1	Receive and Store LWR Fuel	D	Irradiated Fuel	Basin	High	C
5.2	Dissassemble and Declad Rods	HL	Irradiated Fuel	Hot Cell	High	C
5.3	Grind Pellets	CL	Irradiated Fuel Pieces	Hot Cell	High	C
5.4	Mill Powder	CL	$UO_2/F.P./PuO_2$	Hot Cell	High	C
5.5	Blend Powder	CL	$UO_2/F.P./PuO_2$	Hot Cell	High	C
5.6	Mix with Binder and Press	CL	$UO_2/F.P./PuO_2$	Hot Cell	High	C
5.7	Sinter	CL	$UO_2/F.P./PuO_2$	Hot Cell	High	C
5.8	Grind	CL	$UO_2/F.P./PuO_2$	Hot Cell	High	C
5.9	Inspect and Gage	CE	$UO_2/F.P./PuO_2$	Hot Cell	High	C
5.10	Encapsulate Pellets in Rods	CE	$UO_2/F.P./PuO_2$	Hot Cell	High	C
5.11	Inspect Rods	CE	$UO_2/F.P./PuO_2$	Hot Cell	High	C
5.12	Assemble HWR Clusters	CE	Clad $UO_2/F.P./PuO_2$	Hot Cell	High	C
5.13	Inspect and Store	CE	Clad $UO_2/F.P./PuO_2$	Hot Cell	High	C
5.14	Collect Airborne Activity	CE	Air, Xe, Kr, ^3T	Hot Cells	Low, Variable	
5.15	Treat Airborne Activity	HL	Air, Xe, Kr, ^3T	Hot Cell	Low; Variable	
5.16	Collect Solid Waste	CE	Miscellaneous	Hot Cells	Low; Variable	
5.17	Treat Solid Waste	CE	Miscellaneous	Hot Cell	Medium; Variable	
B.	Encapsulation (8.0)					
8.1	Receive Fuel	D	Irradiated LWR Fuel	Basin	High[a]	C[a]
8.2	Transfer to Encapsulation	D	Irradiated Fuel	Basin	High	C
8.3	Encapsulate Fuel Remove Gas	S[b]	Irradiated Fuel	Canyon	High	C
8.4	Store Encapsulated Fuel	D	Irradiated Fuel	Basin or Vault	High	C
8.5	Recycle Failures	S	Failed Irradiated Fuel	Canyon	High	C
8.6	Package for Shipment to Storage	CE	Irradiated Fuel	Basin or Vault	High	C

[a]Varies with age of fuel.

[b]Some Canadian development work for CANDU fuel may be applicable.

than the fuel discharged from the LWR. In this case, the LWR fuel is refabricated into a fuel form that corresponds to the normal CANDU HWR configuration. Fuel pieces for HWRs are larger than for LWRs. The refabrication step may be needed to improve the uniformity of fuel power generation during HWR exposure and to match HWR mechanical design.

A preliminary review of the tandem fuel cycle was made in Reference 1. The refabrication step is technically uncertain, and the fuel performance during extended exposure is undemonstrated. Many years of technical development would probably be required to implement the tandem fuel cycle. The incentive to recover plutonium from stored fuel would depend on the demands of a breeder reactor program. LWR fuel, HWR fuel, or tandem fuel might all contain enough plutonium to warrant processing.

5.2.14 Case 3.3.2 (HWR): Tandem Fuel — Fuel Not Reconstituted

Level 0 segments are shown in Figure 5.49; Level 1 steps, in Figures 5.50 and 5.51. The state-of-the-art and material characteristics for spent fuel encapsulation (7.0) are shown on Table 5.16.

The fuel refabrication step would not be required if a fuel design could be developed that could be used directly in HWRs after discharge from LWRs. Currently there are three PWR vendors and one BWR vendor. About 75 fuel designs have been developed, and there is very little interchangeability (only for some BWR fuel designs). Fuel enrichments are also a variable. It seems unlikely that a dual-purpose fuel design could be developed without a major compromise in fuel economy and without causing a reduction in reactor power.

5.2.15 Case 3.4.1 (HWR): Spectral Shift Reactor — Throwaway of Oxide Fuel (LWR-HWR Hybrid)

Level 0 segments are shown in Figure 5.52; Level 1, is shown in Figure 5.53. The state-of-the-art and material characteristics for this case are shown on Table 5.17.

The spectral shift reactor concept is based on increasing the moderating characteristics of the coolant during fuel exposure to compensate for reactivity changes. This shift in moderating characteristics is achieved by reducing the fraction of heavy water (D_2O) in

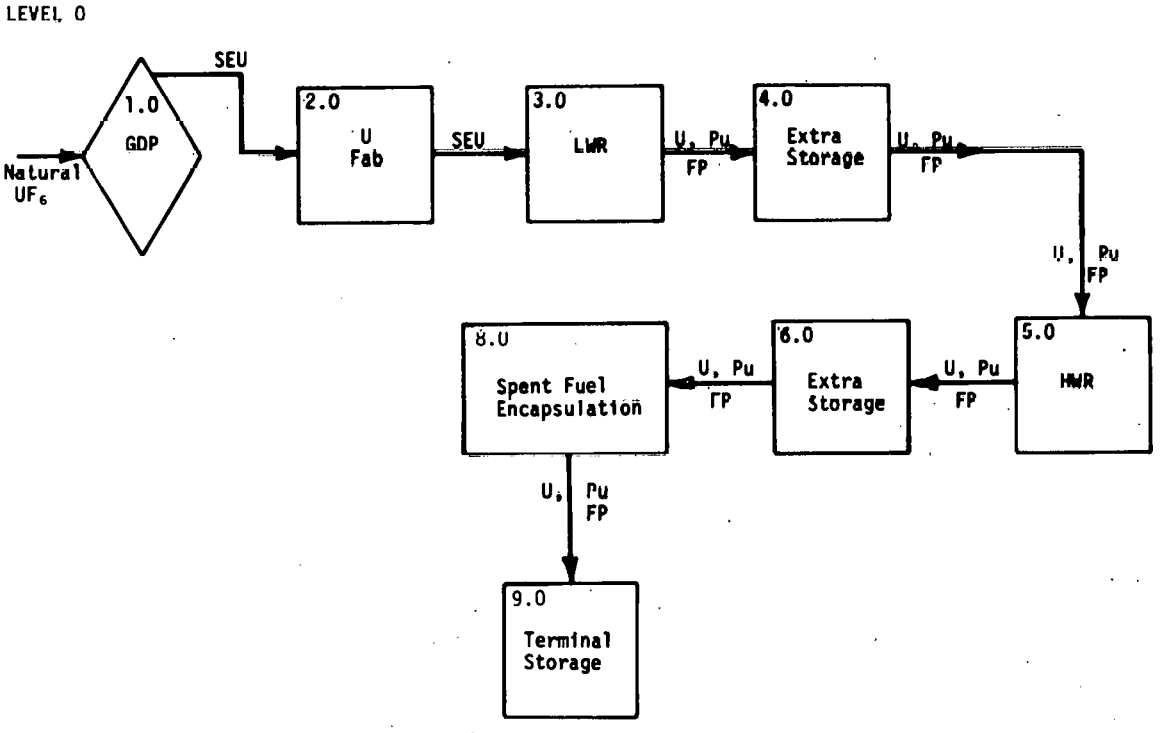
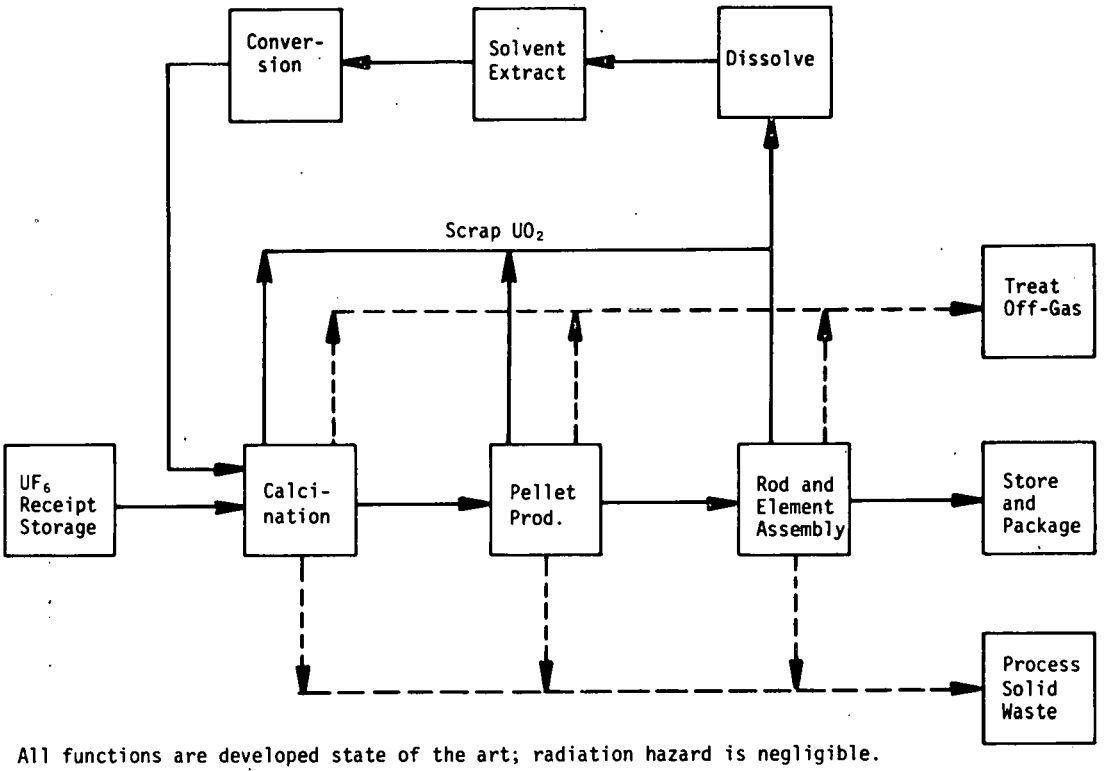


Fig. 5.49. Case 3.3.2 — Tandem Cycle - Fuel Not Reconstituted

LEVEL 1



All functions are developed state of the art; radiation hazard is negligible.

Material convertibility is Category B for all steps except "Treat Off-Gas" and "Process Solid Waste."

Fig. 5.50. Case 3.3.2 — Tandem Cycle - Fuel Not Reconstituted; Slightly Enriched (2-4% ^{235}U) Uranium Fabrication (2.0)

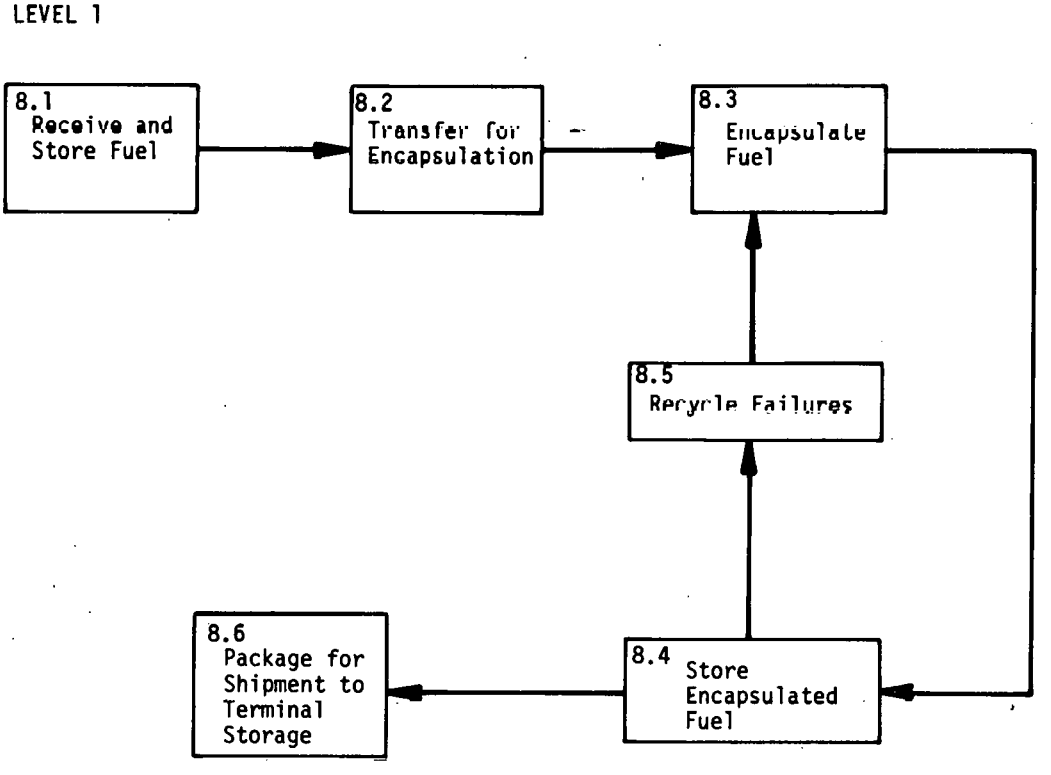


Fig. 5.51. Case 3.3.2 — Tandem Cycle - Fuel Not Reconstituted; Spent Fuel Encapsulation (8.0)

Table 5.16. Case 3.3.2 (HWR): Tandem Cycle - Fuel Not Reconstituted; Encapsulation (8.0)

	Process Step	State-of-the-Art	Material Description	Material Location	Radiation Hazard	Material Convertibility
A.	Encapsulation (8.0)					
	8.1 Receive Fuel	D	Irradiated LWR Fuel	Basin	High[a]	C[a]
	8.2 Transfer to Encapsulation	D	Irradiated Fuel	Basin	High	C
	8.3 Encapsulate Fuel Remove Gas	S[b]	Irradiated Fuel	Canyon	High	C
	8.4 Store Encapsulated Fuel	D	Irradiated Fuel	Basin or Vault	High	C
	8.5 Recycle Failures	S	Failed Irradiated Fuel	Canyon	High	C
	8.6 Package for Shipment to Storage	CE	Irradiated Fuel	Basin or Vault	High	C

[a]Varies with age of fuel.

[b]Some Canadian development work for CANDU fuel may be applicable.

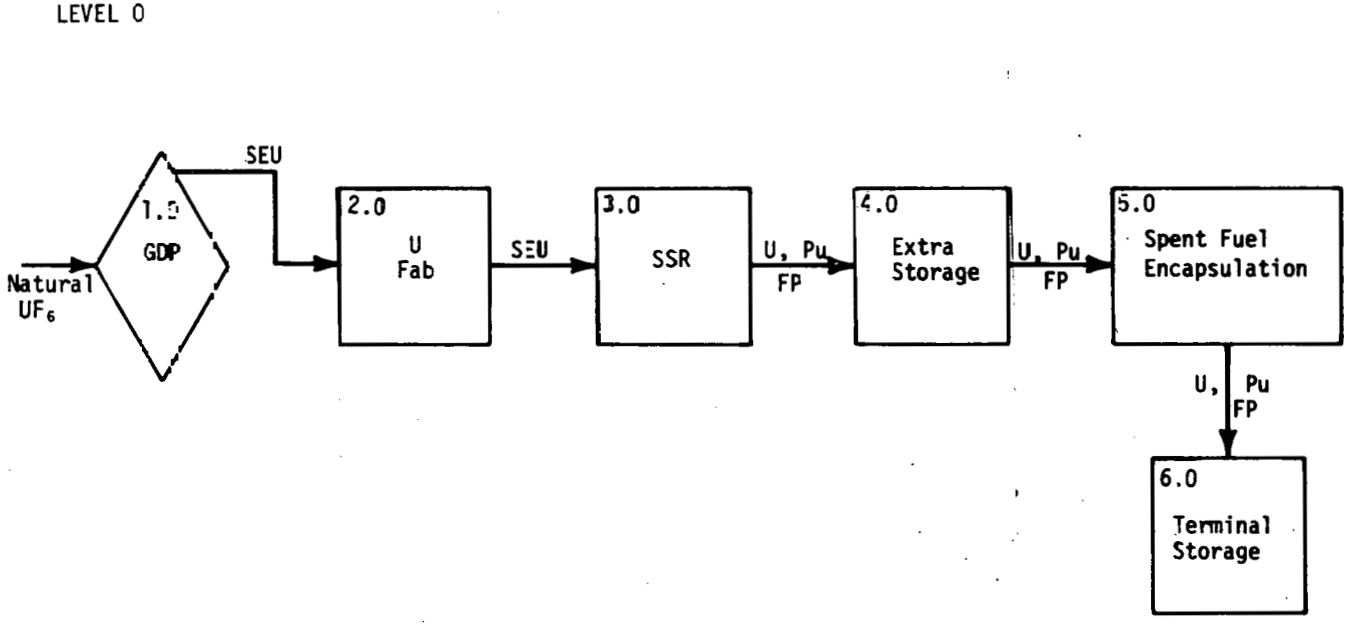
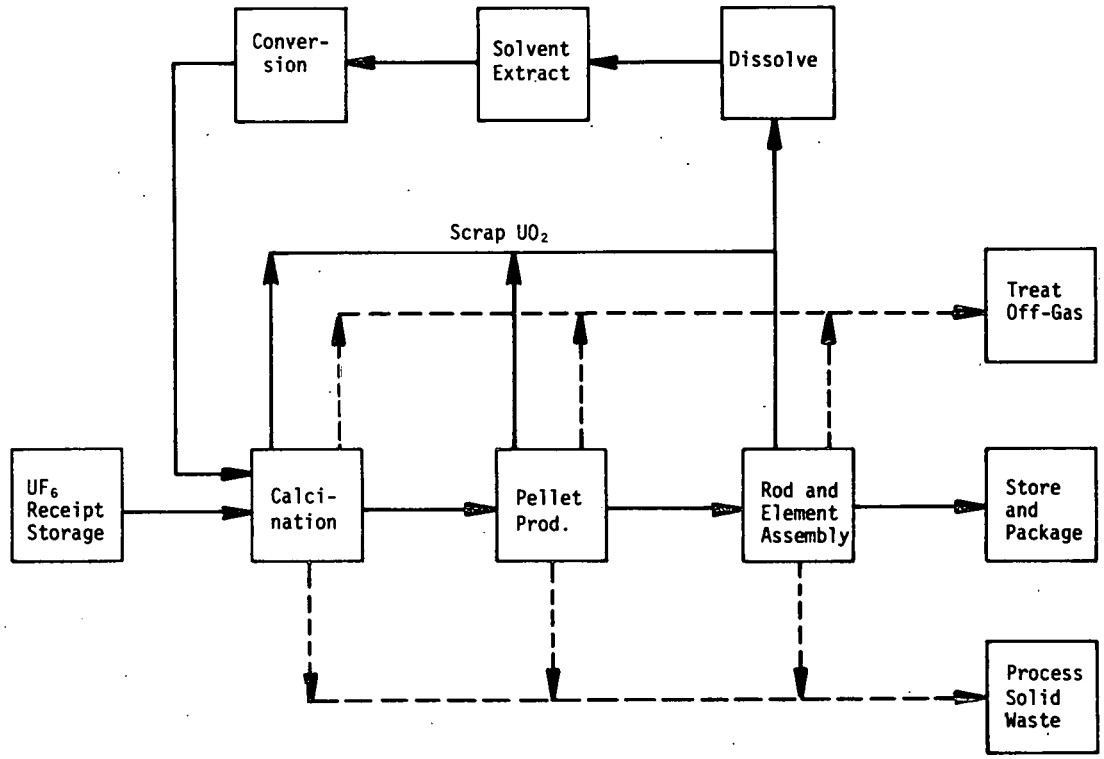


Fig. 5.52. Case 3.4.1 — Spectral Shift Reactor (Throwaway of Oxide Fuel)

LEVEL 1



All functions are developed state of the art; radiation hazard is negligible.

Material convertibility is Category B for all steps except "Treat Off-Gas" and "Process Solid Waste."

Fig. 5.53. Case 3.4.1 — Spectral Shift Reactor (Throwaway of Oxide Fuel); Slightly Enriched (2.4% ^{235}U) Uranium Fabrication (2.0)

Table 5.17. Case 3.4.1 (LWR-HWR): Spectral Shift Reactor (Throwaway - Oxide Fuel)

	Process Step	State-of-the-Art	Material Description	Material Location		Material Convertibility
A.	Encapsulation (8.0)					
	8.1 Receive Fuel	D	Irradiated LWR Fuel	Basin	High[a]	C[a]
	8.2 Transfer to Encapsulation	D	Irradiated Fuel	Basin	High	C
	8.3 Encapsulate Fuel Remove Gas	S[b]	Irradiated Fuel	Canyon	High	C
	8.4 Store Encapsulated Fuel	D	Irradiated Fuel	Basin or Vault	High	C
	8.5 Recycle Failures	S	Failed Irradiated Fuel	Canyon	High	C
	8.6 Package for Shipment to Storage	CE	Irradiated Fuel	Basin or Vault	High	C

[a]Varies with age of fuel.

[b]Some Canadian development work for CANDU fuel may be applicable.

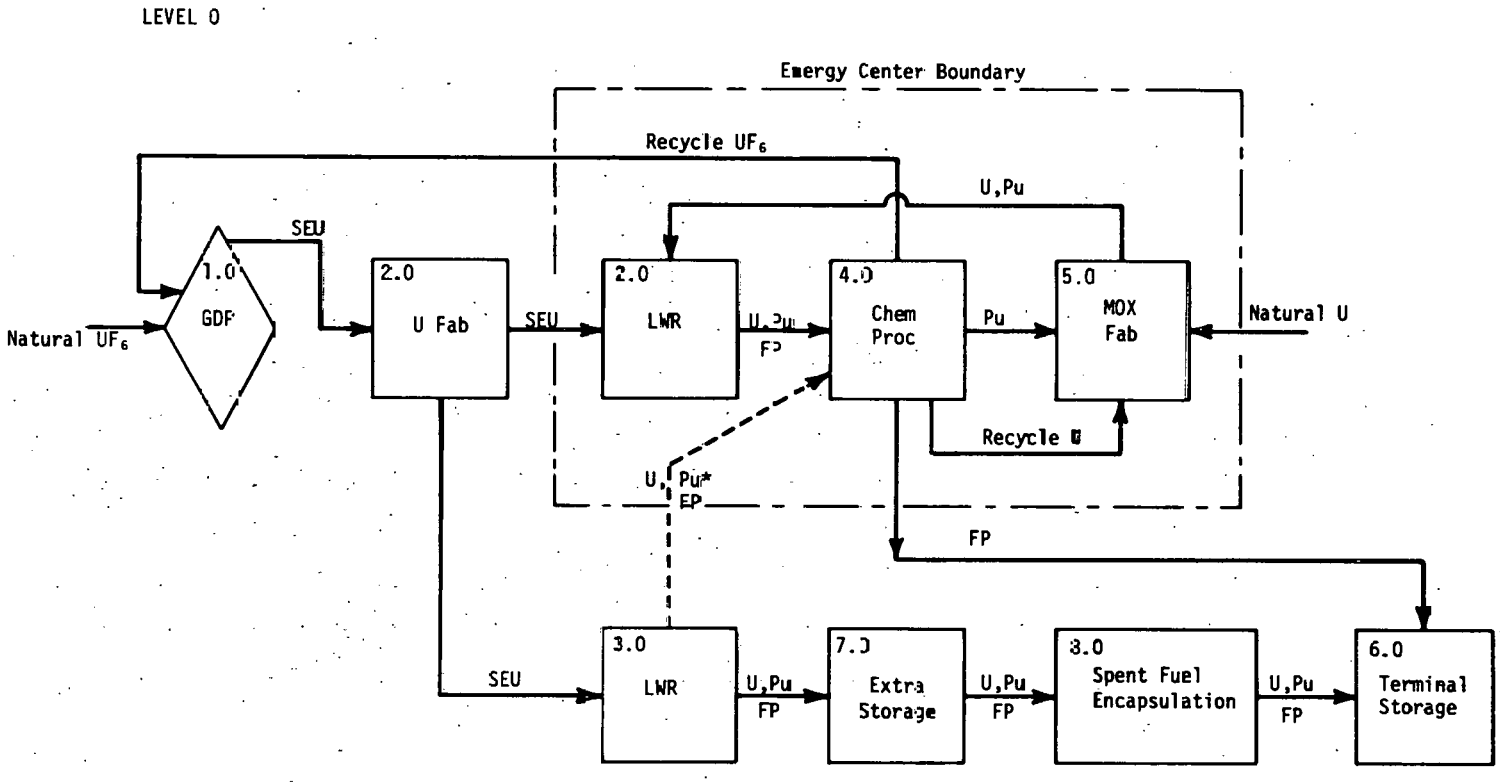
the coolant from 60 to 10% during a fuel cycle, so that the spectrum approaches that of the LWR-PWR. This mode of operation has the advantage of reducing the amount of enrichment needed from 3.2 to 2.6 wt % ^{235}U in the oxide fuel and eliminates the necessity of using boron as a chemical shim for reactivity control in a PWR. B&W once attempted to market the spectral shift reactor.

If the same fuel exposure is attained (by producing more ^{239}Pu fuel by resonance absorption with the spectral shift reactor concept), a resource savings of about 20% is attainable without reprocessing. It is not clear that annual fuel reloadings are compatible with the spectral shift mode. The details of adjusting the coolant concentration should be carefully reviewed. The conversion from 10% to 60% D_2O would probably require a significant inventory of D_2O which, in addition to facilities for upgrading the discharged coolant, would increase the capital cost of the spectral shift concept.

5.2.16 Case 7.1.1 (Energy Center - LWR): Combination of Fuel Cycles with Internal Control of Plutonium

Level 0 segments are shown in Figure 5.54; Level 1 steps, in figures 5.55, 5.56, 5.57 and 5.58. The state-of-the-art and material characteristics for chemical processing (4.0), MOX fabrication (5.0) and spent fuel encapsulation (8.0) are shown in Table 5.18.

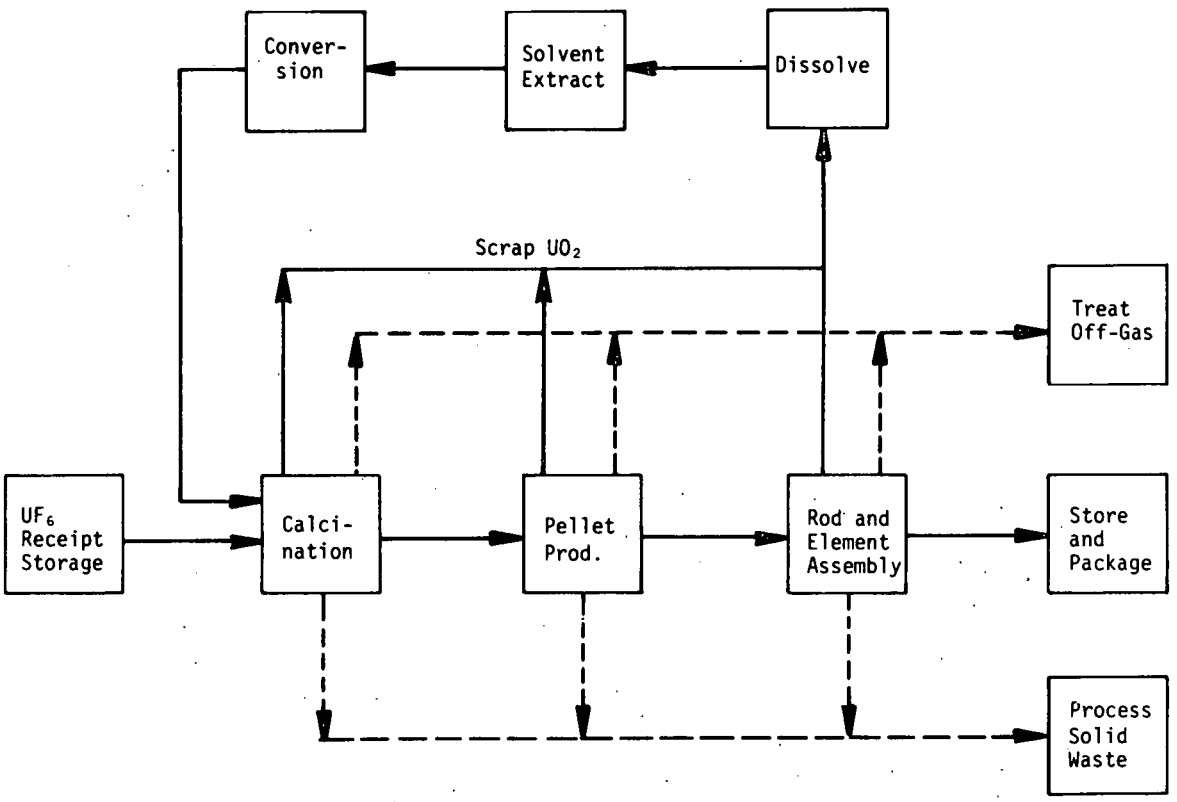
Light water reactors fueled with uranium-plutonium would be operated inside the energy center for internal control of the plutonium. Uranium-fueled LWRs would be operated outside the energy center. All irradiated fuel from outside the energy center could be shipped into the energy center for internal processing and eventual use if the reactor capacity in the center were large enough. About 30% of the total LWR power would have to be located within the energy center. The flexibility shown in the diagram implies some shipment of fuel to terminal storage as if the energy center does not have sufficient capacity.



* Utilization of outside fuel depends on relative number of reactors inside and outside of parks.

Fig. 5.54. Case 7.1.1 – Energy Center; Internal Control of Plutonium

LEVEL 1



All functions are developed state of the art; radiation hazard is negligible.

Material convertibility is Category B for all steps except "Treat Off-Gas" and "Process Solid Waste."

Fig. 5.55. Case 7.1.1 — Energy Center; Slightly Enriched (2-4% ^{235}U) Uranium Fabrication (2.0)

LEVEL 1

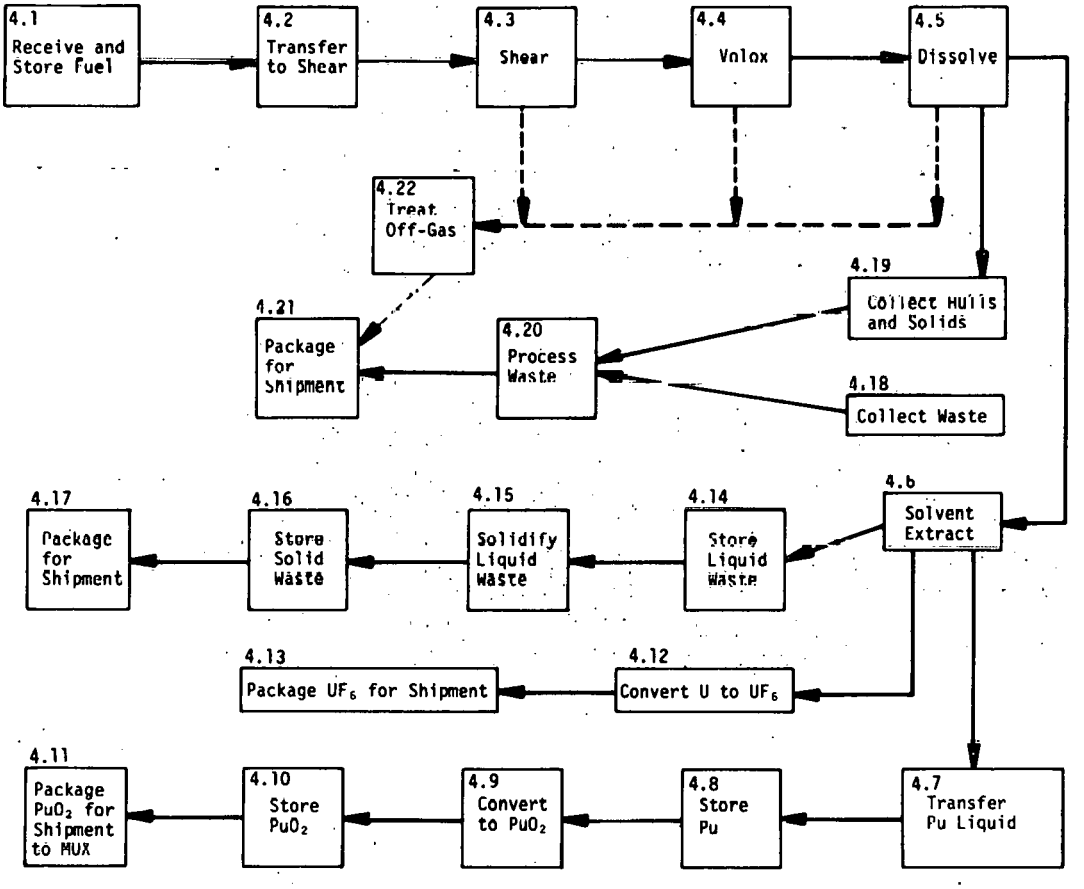


Fig. 5.56. Case 7.1.1 — Energy Center; Chemical Reprocessing (4.0)

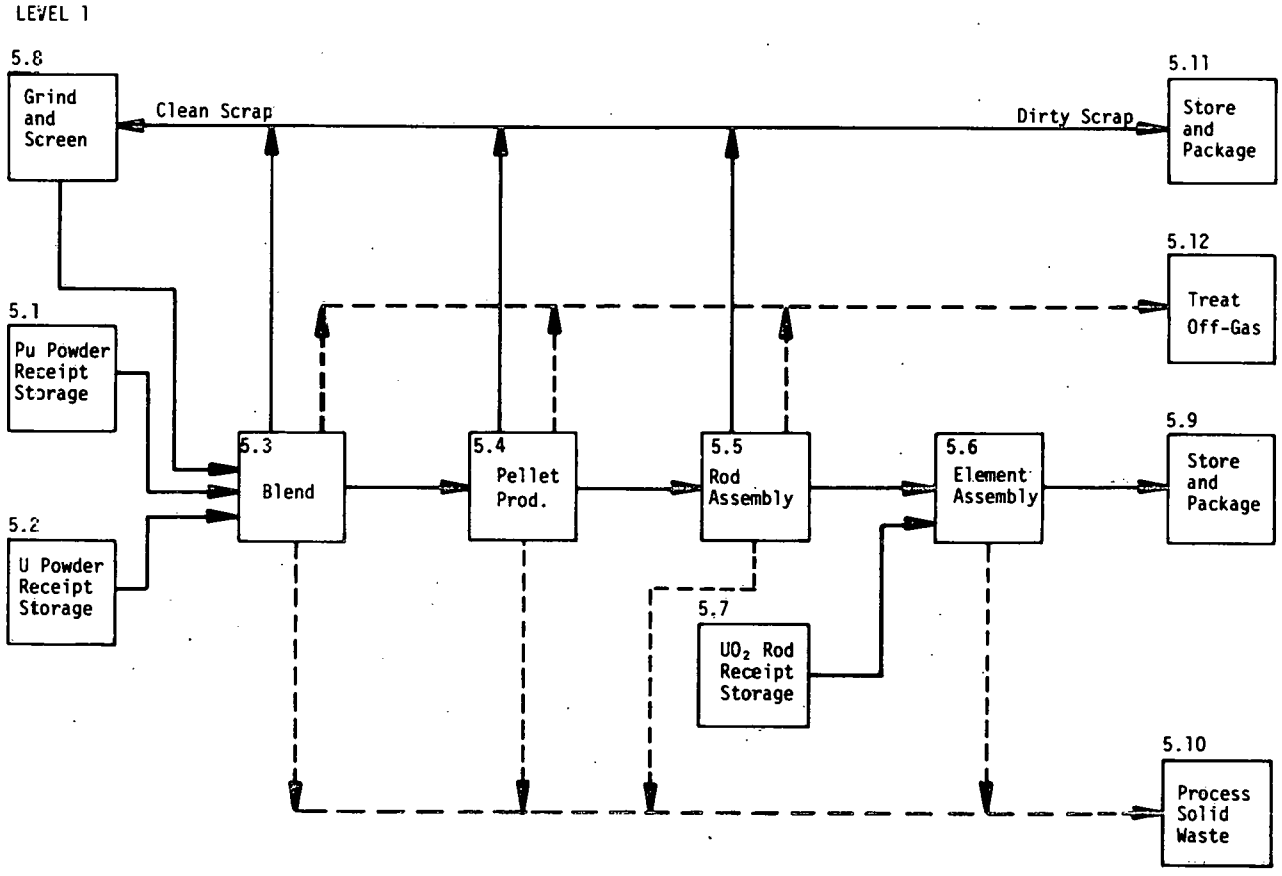


Fig. 5.57. Case 7.1.1 – Energy Center; MOX Fabrication (5.0)

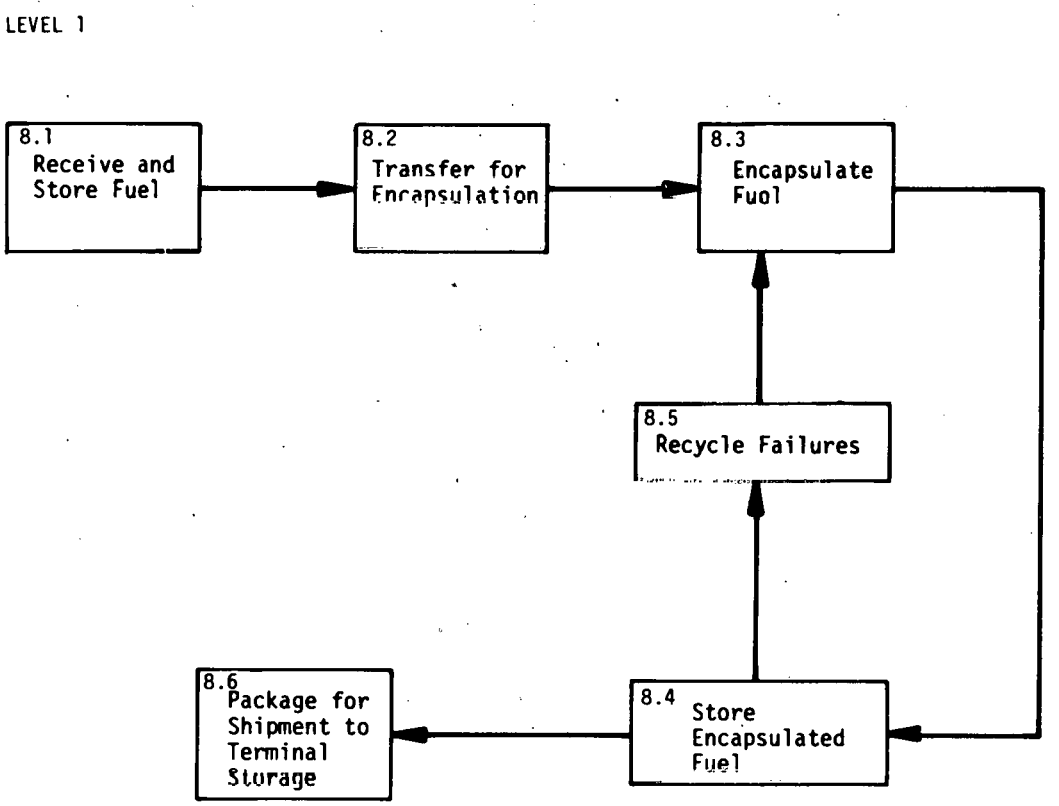


Fig. 5.58. Case 7.1.1 — Energy Center; Spent Fuel Encapsulation (8.0)

Table 5.18. Case 7.1.1 (LWR Energy Center): Internal Recycle of Plutonium

	Process Step	State-of-the-Art	Material Description	Material Location	Radiation Hazard	Material Convertibility
A.	Chemical Reprocessing (4.0)					
4.1	Receive and Store Fuel	D	Irradiated Fuel	Basin	High	C
4.2	Transfer to Shear	D	Irradiated Fuel	Basin	High	C
4.3	Shear	D	Fuel Elements	Canyon	High	C
4.4	Voloxidation	HL	Fuel Element Pieces	Canyon	High	C
4.5	Dissolve	D	Fuel in Solution	Canyon	High	C
4.6	Solvent Extraction	D	Solution of U, Pu, Fission Products	Canyon	High	C
4.7	Transfer Pu Liquid	D	Pu Nitrate	Canyon	Low	D
4.8	Store Pu	D	Pu Nitrate	Canyon	Low	D
4.9	Convert to PuO_2	HE	PuO_2	Cell	Low	D
4.10	Store PuO_2	D	PuO_2	Cell	Low	D
4.11	Package for Shipment	D	PuO_2	Cell	Low	D
4.12	Convert U to UF_6	D	U Nitrate	Cell	Negligible	B
4.13	Package UF_6	D	UF_6	Warehouse	Negligible	B
4.14	Store Liquid Waste	D	Radioactive Liquids	Canyon	High	
4.15	Solidify Liquid Waste	HDF	Radioactive Solids	Canyon	High	
4.16	Store Solid Waste	HDF	Radioactive Solids	Canyon	High	
4.17	Package for Shipment	CE	Radioactive Solids	Canyon	High	
4.18	Collect Waste	D	Radioactive Solids	Canyon	Variable	
4.19	Collect Hulls and Solids	D	Radioactive Solids	Canyon	High	
4.20	Process Waste	CE	Radioactive Solids	Canyon	High	
4.21	Package for Shipment	CE	Radioactive Solids	Cell	High	
4.22	Treat Off-Gas	HL	Radioactive Gases	Canyon	High	
B.	MOX Fabrication (5.0)					
5.1	Receive and Store Pu Powder	HDF	PuO_2	Cell	Low	D
5.2	Receive and Store U Powder	HDF	UO_2	Cell	Negligible	B
5.3	Blend	HDF	PuO_2/UO_2	Cell	Low	D
5.4	Produce Pellets	HDF	PuO_2/UO_2	Cell	Low	D
5.5	Assemble Rods	HDF	PuO_2/UO_2/Cladding	Cell	Low	D
5.6	Prepare Assemblies	HDF	PuO_2/UO_2/Cladding	Cell	Low	D
5.7	UO_2 Rod Storage	D	UO_2/Cladding	Warehouse	Negligible	B
5.8	Regrind and Screen Scrap	HDF	PuO_2/UO_2	Cell	Low	D
5.9	Store and Package	HDF	PuO_2/UO_2	Warehouse	Low	D
5.10	Process Solid Waste	HDF	Miscellaneous	Cell	Negligible	
5.11	Store and Package Scrap	HDF	Miscellaneous	Cell		
5.12	Treat Off-Gas	HDF	Pu	Cell		
C.	Encapsulation (8.0)					
8.1	Receive Fuel	D	Irradiated LWR Fuel	Basin	High[a]	C[a]
8.2	Transfer to Encapsulation	D	Irradiated Fuel	Basin	High	C
8.3	Encapsulate Fuel Remove Gas	S[b]	Irradiated Fuel	Canyon	High	C
8.4	Store Encapsulated Fuel	D	Irradiated Fuel	Basin or Vault	High	C
8.5	Recycle Failures	S	Failed Irradiated Fuel	Canyon	High	C
8.6	Package for Shipment to Storage	CE	Irradiated Fuel	Basin or Vault	High	C

[a] Varies with age of fuel.

[b] Some Canadian development work for CANDU fuel may be applicable.

5.3 PROCESSING SIDESTREAMS

Normal processing sequences for the 16 alternative fuel cycles analyzed by SRL are shown in the previous figures and tables. However, the ease of diverting fissionable material into a sidestream for special processing and conversion into weapons is also of interest in this study. The ease of modifying a process to recover the fissionable material in a more convenient form will also be studied. A preliminary description of such sidestream processing to recover plutonium is given in Table 5.19 for the SRL cases. Sidestream processing for some cases will be studied in more detail in Phase 2. The cases are listed in order of increasing difficulty; however, the degree of difficulty is judged to be minor.

Recovery of plutonium from a solution can be accomplished by precipitation or ion exchange as well as the large-scale industrial separation methods (mixer-settlers, pulse columns, etc.). Radiation provides some interference with processing, but is only a minor complication in a nationally planned program of recovery.

5.4 SPIKING PLUTONIUM

Cases 1.1.2, 1.1.3, and 3.1.2 require that plutonium streams contain a source of high radiation to serve as a deterrent to diversion of plutonium for unauthorized purposes. Some relationship between the radiation level and resistance to sidestream diversion (to proliferation) may be developed in future work to aid in determining a suitable processing method. Preliminary consideration of spiking plutonium assumes plutonium containing the residual fission products after one cycle of solvent extraction will be adequately spiked. Calculations by the SRL Reactor Physics Division indicate the radiation level in the spiked plutonium is about ten-fold higher than in purified plutonium, because an effective DF of 5000 is calculated for the first cycle of solvent extraction. Thus, the spiked plutonium would emit more than several R/hr/kg. It would be necessary to decrease the effectiveness of the Purex process if a further increase in radiation is warranted.

Table 5.19. Diversion of Sidestreams

Case[a]	Sidestream Composition	Sidestream Radiation	Steps to Obtain Pu Metal in Plant After Decision to Proliferate	Facility for Pu Recovery: Industrial Plant	Facility for Pu Recovery: Clandestine
U, Pu Recovery Cases					
1.1.1 Recycle of U and Pu to LWR (Base) or 2.1.7 Recycle of U and Pu to AGR or 3.1.1 Recycle of U and Pu to HWR	PuO_2	1-2 R/hr	Reduce PuO_2.	Yes	Possible
1.1.7 Coprocessed Pu Recycle to LWR	PuO_2-UO_2	~1 R/hr	Modify operation to separate U and Pu, reduce Pu.	Yes	Possible
1.1.2 Spiked Pu Recycle to LWR or 3.1.2 Spiked Pu Recycle to HWR	PuO_2	Under study but assumed to be over 10 R/hr	Modify operation to separate Pu and fission products, recover and reduce Pu (shielded operation).	Yes	Possible
1.1.3 Coprocessed Spiked Pu Recycle to LWR	PuO_2-UO_2	Over 10 R/hr	Modify operation to separate Pu from U and fission products, recover and reduce Pu (shielded operation).	Yes	Possible
U Recovery Case					
1.1.4 Recycle Uranium-Plutonium and Fission Products to Waste (LWR)	PuO_2 and fission products	Variable but very high >10^4 R/hr	Modify operation to separate, recover, and reduce Pu (shielded operation).	Yes	Possible
Throwaway Cases					
Reactor Type / Fuel					
1.1.5 LWR / Oxide; 1.1.6 LWR / Metal; 2.1.6 GCR / Metal; 3.1.3 HWR / Oxide; 3.4.1 PWR/HWR / Oxide	Irradiated Fuel Assemblies	Variable but very high	Divert fuel assemblies, dissolve fuel, separate Pu, reduce Pu.	No	Required
Tandem Cases					
3.3.1 3.3.2	Irradiated Fuel Assemblies	Variable but very high	Same as for throwaway.	No	Required
Energy Center					
7.1.1 LWR with Pu recycle inside LWR with U fuel only outside Irradiated fuel shipped back to center	Irradiated Fuel Assemblies	Variable but very high	Same as for throwaway outside energy center. Energy center is not sited in a non-weapons state.	No	Required

[a]Cases listed in order of increasing technical difficulty.

Several techniques of spiking plutonium are listed in Table 5.20. All methods provide some deterrence to diversion of small amounts of fissile material, but none seem significantly proliferation-resistant. All would complicate the handling of the fuel and increase the risk of a severe occupational dose to members of the nuclear work force. Most methods impair the fuel fabrication process. However, all can be relatively easily defeated when a national determination is made to obtain purified plutonium. The ease of chemical separation of plutonium, fission products, and uranium (ion exchange, precipitation, or solvent extraction with valence adjustments) is not significantly complicated by the need for shielding to reduce dose rates. A national plan to assemble equipment for the task would not be delayed by radiation; cruder equipment and poorer yields with radiation would not seriously deter the task. For this reason, irradiated fuel assemblies (throwaway cases) or the mixture of plutonium and fission products (Case 1.1.4) would also provide appropriate feed materials for a covert crude plutonium recovery plan.

Fission products considered for spiking plutonium are characterized in Table 5.21. No isotope has ideal qualifications of high dose, long half-life, non-volatility, and chemical affinity for the plutonium stream.

Zirconium and ruthenium isotopes were suggested[2] as the best candidates for deterring diversion (safeguards) in plutonium for up to 200 days after separation from fuel that had decayed 160 days. The fuel cost was estimated to increase 10 to 50% as a consequence of the fuel containing fission products. No consideration of resistance to proliferation was included in Reference 2.

5.5 REFERENCES

1. A. J. Frankel and N. L. Shapiro, *Appraisal of PWR-HWR Tandem Fuel Cycles*, Combustion Engineering Corp., Report NPSD-45, 1977.
2. Bruce Hutchins, *Denatured Plutonium: A Study of Deterrent Action*, USERDA Report EPRI 310, 1975.

Table 5.20. Spiking Plutonium

	Method	Effect on Fuel Process	Means to Defeat	Radiation Level
A.	Fission Products			
	Incomplete Removal	Yes	Yes; recycle	High; but will decay
	Selective Partition and Add-Back	Yes	Yes; avoid add-back	High; but will decay
	Irradiate Fuel after Fabrication	No	Yes; bypass irradiation	Yes; but will decay
B.	Cobalt Sources			
	Mixed with Pu Fuel	Yes	Yes; separate chemically	High; less decay
	Added to Fuel Assembly or to Pu Fabrication Plant	No	Yes	High; less decay
C.	^{238}Pu	No	No	Minor (~few R/hr)

Table 5.21. Candidates for Fission Product Spiking of Plutonium

Isotope	Radiation Level, R/hr @ 1 yr/kg Pu Gamma	Beta	Half Life, yr	Cross Section, barns	Volatiles in MOX Pellet Fab.	Simple Contaminate Pu
Y^{91}	1	5	0.2	1.4	-	-
Zr^{95}	2×10^3	2	0.2	-	-	Yes
Ru^{103}	40	<0.1	0.1	-	Yes	Yes
Ru^{106}	2×10^3	200	1	0.15	Yes	Yes
Sb^{125}	150	5	2.7	-	-	-
Cs^{137}	2×10^3	7	30	0.1	Yes	-
Ce^{141}	1	<0.1	0.1	29	-	No[a]
Ce^{144}	7×10^2	200	0.8	1	-	No[a]

[a]Pu contaminated with a mixture of rare earths is achievable, but selective contamination with cerium and not gadolinium or samarium is very difficult.

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