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An Evaluation of Fuel Fabrication and the Back End of the Fuel Cycle for Light-Water- and Heavy-Water-Cooled Nuclear Power Reactors

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AN EVALUATION OF FUEL FABRICATION AND THE BACK END OF THE FUEL CYCLE FOR LIGHT-WATER- AND HEAVY-WATER-COOLED NUCLEAR POWER REACTORS

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ABSTRACT

The classification of water-cooled nuclear reactors offers a number of fuel cycles that present inherently low risk of weapons proliferation while making power available to the international community. Eight fuel cycles in light water reactor (LWR), heavy water reactor (HWR), and the spectral shift controlled reactor (SSCR) systems have been proposed to promote these objectives in the International Fuel Cycle Evaluation (INFCE) program. Each was examined in an effort to provide technical and economic data to INFCE on fuel fabrication, refabrication, and reprocessing for an initial comparison of alternate cycles. The fuel cycles include three once-through cycles that require only fresh fuel fabrication, shipping, and spent fuel storage; four cycles that utilize denatured uranium-thorium and require all recycle operations; and one cycle that considers the LWR-HWR tandem operation requiring refabrication but no reprocessing. The once-through cycles include an LWR, an SSCR, and an HWR fueled with UO_2; the denatured cycles include an LWR fueled with $(^{235,238}U,Th)O_2$, an LWR fueled with $(^{233,238}U,Th)O_2$, an SSCR fueled with $(^{233,238}U,Th)O_2$, and an HWR fueled with $(^{233,238}U,Th)O_2$; the tandem cycle utilized UO_2 fuel, with the LWR discharge being used to fuel an HWR.

Three commercially feasible processes are available for fabrication and/or refabrication of these fuels: the powder-pellet process, the Sphere-Cal process, and the Sphere-Pac process. Each process functions to take heavy-metal nitrate solutions and prepare oxide fuels of the specific form and composition required for the reactor. Although there are important differences in the mechanical and nuclear designs of the fuels for these reactors, the functional operations to convert feed materials into finished fuel assemblies for any type of reactor are similar.

The four fuel cycles that require recovery and recycle of fuel values utilize a modified Thorex process by which uranium, plutonium, and thorium can be decontaminated. Since the denatured fuels produce significant amounts of plutonium, it is necessary to reprocess in a secured area and to retain the recovered plutonium in this area. If plutonium is recycled, the reactor that utilizes it must also be in the secured area. These reactor fuel cycles employ intimate fissile-fertile

fuel mixtures; hence it is unnecessary to completely partition uranium and thorium during reprocessing. Partitioning needs to be carried out only to the extent necessary to satisfy mass flow requirements of the reactor cycle.

This evaluation indicates that the basic technology for recycling thorium fuels exists through reference to demonstrated procedures for uranium-plutonium fuels; however, thorium fuel recycle has not been demonstrated. A significant research and development (R&D) effort needs to be carried out prior to commercialization of the thorium cycle. Testing and qualification of the various fuels to qualify their use in licensed power reactors are expected to require considerable time. Refabrication of these fuels is complicated by the presence of ^{232}U in recycle uranium and ^{228}Th in recycle thorium. The decay of these isotopes adds considerable radioactivity to both uranium and thorium, requiring remotely operated and maintained refabrications. Technology and processes that are amenable to remote operation must be developed. There are significant new requirements for all principal operations in reprocessing. A modified version of the Thorex process that is required to handle uranium-plutonium-thorium mixtures must be demonstrated. Although head-end operations are similar to developed procedures for metal-clad uranium-plutonium fuels, anticipated differences in treating uranium-thorium fuels will necessitate significant development. In off-gas treatment, it can be expected that future regulations governing plant effluents will require rather thorough removal of all radioactive gases; these requirements necessitate R&D to upgrade existing processes or develop new procedures, for example, radon removal. The principal new requirement imposed on waste treatment is to determine the effect of fluoride ion (required to catalyze ThO_2 dissolution) on waste treatment operations. Once-through fuel cycles require development to qualify fuels for the longer in-reactor exposure, but no reprocessing or refabrication research is needed; however, spent fuel storage will require proof of fuel element integrity and heat removal methods. The tandem LWR-HWR fuel cycle is unique in that it is neither a once-through system nor a recycle system, requiring refabrication but no reprocessing. The main R&D effort involved in this case is designing of LWR and HWR reactor systems so that movement of fuel between the two reactors requires a minimum effort and cost in refabrication or reconfiguration.

Research and development costs for the once-through systems are estimated to be in the range $40 to $70 million for improved LWRs, $80 to $120 million for SSCRs, $300 to $500 million for HWRs, and $1000 to $2300 million for tandem LWRs/HWRs. These costs include demonstration facilities for the HWR and tandem fuel cycles. Research and development costs for the recycle systems are in the range $1800 to $3300 million; these include the costs of demonstration facilities for both fresh fuel fabrication and all recycle operations. It is estimated that approximately 20 years will be required for a complete recycle demonstration.

Established industry plant design, construction, and operating cost estimates were made for fabrication plants with an annual capacity of 520

metric tons of heavy metal (MTHM), for refabrication plants with an annual capacity of 480 MTHM, and for reprocessing plants with an annual capacity of 1500 MTHM. The capital and operating costs were used in a discounted cash flow analysis to obtain a levelized product price. Three sets of economic assumptions were used to exhibit unit costs for low-risk financing, typical industrial financing, and high-risk financing. The unit costs, which are cycle specific, were computed separately for fabrication, refabrication, and reprocessing.

Preliminary environmental, safety and licensability assessments indicate that recycle plants and other facilities for these water reactor fuel cycles can be designed, constructed, and operated within current regulations.

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1. INTRODUCTION

The classification of water-cooled nuclear power reactors includes those that are cooled by light water (LWRs), those that are cooled by heavy water (HWRs), and those that are cooled by a mixture of light and heavy water (SSCRs). Most operating and planned nuclear power installations employ LWRs of either the pressurized water (PWR) or boiling water (BWR) type; however, HWRs have been extensively developed in Canada and are in use there as well as in other parts of the world. Although the SSCR concept has been proven in lattice experiments, no power reactors of this type have been built. Such reactors would make use of both LWR and HWR technology to give improved fuel utilization; therefore, they are prominent candidates for achieving the benefits of nuclear energy within the direction of the INFCE program.

The aim of INFCE is to identify fuel cycles that present inherently low risk of nuclear weapons proliferation while supplying nuclear power to the international community. In order to limit the availability of nuclear materials from which weapons can be made, it is desirable to fuel reactors with nuclear materials that are inherently unsuitable for weapons. In addition, it is recognized that a limitation is imposed on expansion of the nuclear power industry by the size of the uranium resource base and the efficiency with which available resources can be used. Alternate fuel cycles have been proposed to promote these objectives in LWR, HWR, and SSCR systems. This study examines fuel fabrication

and the back end of the proposed fuel cycles for these reactors to provide technical and economic data to INFCE for the initial comparison of alternate cycles.

The fuel cycles of this study include three once-through cycles that require only fresh fuel fabrication, shipping, and spent fuel storage, four cycles that utilize denatured uranium-thorium and require all recycle operations, and one cycle that considers the LWR-HWR tandem operation requiring refabrication but no reprocessing. They are designated as follows:

1. LWR — UO_2 fuel; improved, once-through fuel cycle,[1]
2. LWR — denatured $(^{235}U,Th)O_2$ fuel with recycle,[2]
3. LWR — denatured $(^{233}U,Th)O_2$ fuel with recycle,[2]
4. SSCR — UO_2 fuel; optimized, once-through fuel cycle,[3]
5. SSCR — denatured $(^{233}U,Th)O_2$ fuel with recycle,[4]
6. HWR — UO_2 fuel; once-through fuel cycle,[5]
7. HWR — denatured $(^{233}U,Th)O_2$ fuel with recycle,[6] and
8. LWR/HWR — UO_2 fuel; tandem fuel cycle.[7]

2. DESCRIPTION OF THE FUEL CYCLES

Each of the eight fuel cycles of this study utilizes fissile material concentrations that are isotopically unsuitable for weapons fabrication. When ^{235}U is the fissile component, 20 wt % ^{235}U is the allowable upper limit; the corresponding limit for ^{233}U is 12 wt %. Under these restrictions, the LWR, HWR, or SSCR need not be located in a secured center. However, since irradiated fuel from all cycles contains appreciable plutonium, the reprocessing plant must be placed in a secured area. Plutonium is not recycled to its parent system and thus may be retained in monitored storage or recycled to a reactor located within the same secured area as the reprocessing plant. The following descriptions of these alternate fuel cycles identify the major components and fuel requirements for the cycle; in the case of recycle fuels, the required reprocessing option(s) and refabrication requirements are pointed out.

Detailed discussions of fabrication, reprocessing, and refabrication may be found in Sects. 3, 4, and 5 respectively.

2.1 LWR — UO_2 Fuel; Improved, Once-Through Cycle[1]

Programs are under way in the United States to develop improved, once-through fuel cycles for LWRs. Emphasis is being placed on near-term improvements that can be made without the need for extensive reactor development and that can be retrofitted into LWRs of current design. Improvement categories under consideration are increased burnup to perhaps 50,000 to 55,000 MWd per MTHM, lattice changes to increase the water/fuel ratio, spectrum shift without the use of heavy water, enrichment zoning, and modified fuel management and handling.

The fuel cycle for this LWR is relatively simple since only fabrication and spent fuel storage are required (Fig. 2.1). The initial fuel is low-enriched uranium with a ^{235}U content of about 2.21 wt %. Fuel is discharged at the rate of 20% of the core inventory per year, and the replacement fuel has a higher enrichment than that of the original charge. At equilibrium, the enrichment of the replacement elements is about 4.15 wt %. Initial fuel is irradiated until the fissile uranium content is reduced to about 0.43 wt %; however, sufficient bred plutonium is present to increase the total fissile concentration to about 1.04 wt %. The corresponding values for irradiated makeup fuel are about 0.71 wt % for fissile uranium and 1.51 wt % for total fissile material.

The storage facility retains the spent fuel until its future disposition or treatment is decided. Spent fuel storage facilities in this cycle need to be located in a secured area.

2.2 LWR — Denatured (^{235}U,Th)O_2 Fuel with Recycle[2]

One means of reducing the demand on uranium resources for LWRs is to employ a denatured uranium-thorium fuel cycle in which a significant quantity of fissile material might be bred. Both fissile plutonium

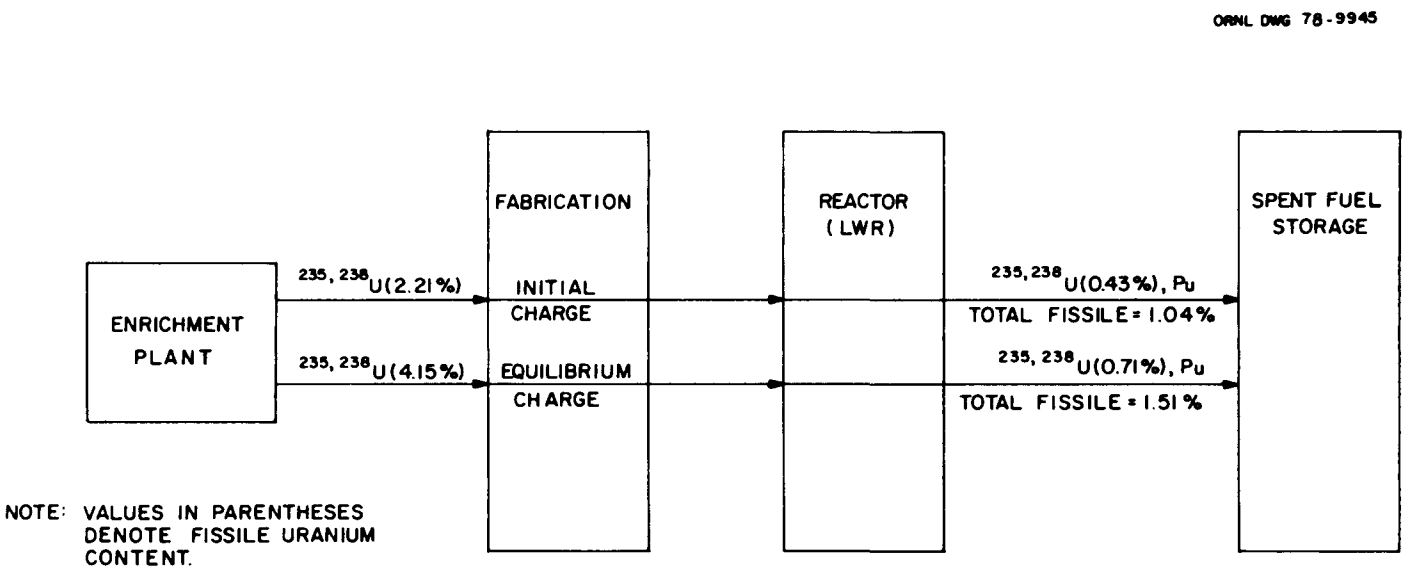


Fig. 2.1. Diagram showing heavy-metal flow for an LWR once-through fuel cycle.

(^{239}Pu and ^{241}Pu) and fissile uranium (^{233}U) are generated in the cycle. Fissile values in the spent fuel may be recovered for recycle or other disposition consistent with the mandates of the fuel cycle. The reactor need not be operated in an energy park or secure center because of the limitation imposed on the fissile content ($\leq$20% ^{235}U) of the fuel.

The fuel cycle for denatured (^{235}U,Th)O_2 fuel in LWRs is diagramed in Fig. 2.2. The initial fuel, which is fabricated by direct methods, contains no more than 20 wt % fissile isotope and is not immediately useful for weapons fabrication. One-third of the reactor inventory is discharged annually and reprocessed after a suitable decay period. In the early part of the cycle before recycle fuel is available, the discharged fuel is replaced with fresh makeup fuel. The requirement for fresh makeup is gradually reduced as recycle fuel becomes available.

Thorium-base fuels are treated by the Thorex process to separate fissile and fertile materials from fission products. This fuel cycle requires a modification to the reference Thorex process since there is a need for separating plutonium that has been bred from the ^{238}U denaturant. For high diversion resistance, it is desirable to recycle all plutonium to reactors in a secure area where the reprocessing plant is located rather than transport it to dispersed reactors. Both the operating conditions for and the chemistry of the Thorex process are adjusted to partition plutonium, uranium, and thorium so that the plutonium appears in a separate product stream. It is also necessary to partition (or partially partition) thorium from uranium since a portion of the thorium must be removed from the cycle in order to maintain the appropriate fuel balance. The remaining thorium is recycled. Plutonium is converted to the oxide (PuO_2) and then held in secure storage or recycled to other reactors such as a plutonium-thorium converter in an energy park.

Recovered uranium and thorium are blended with makeup enriched uranium for refabrication into recycle fuel elements. The enrichment of the makeup fuel may be any value greater than 20 wt % ^{235}U; however, using fuel that is near the 20 wt % enrichment level burdens the reactor with ^{238}U, resulting in the production of more plutonium in subsequent

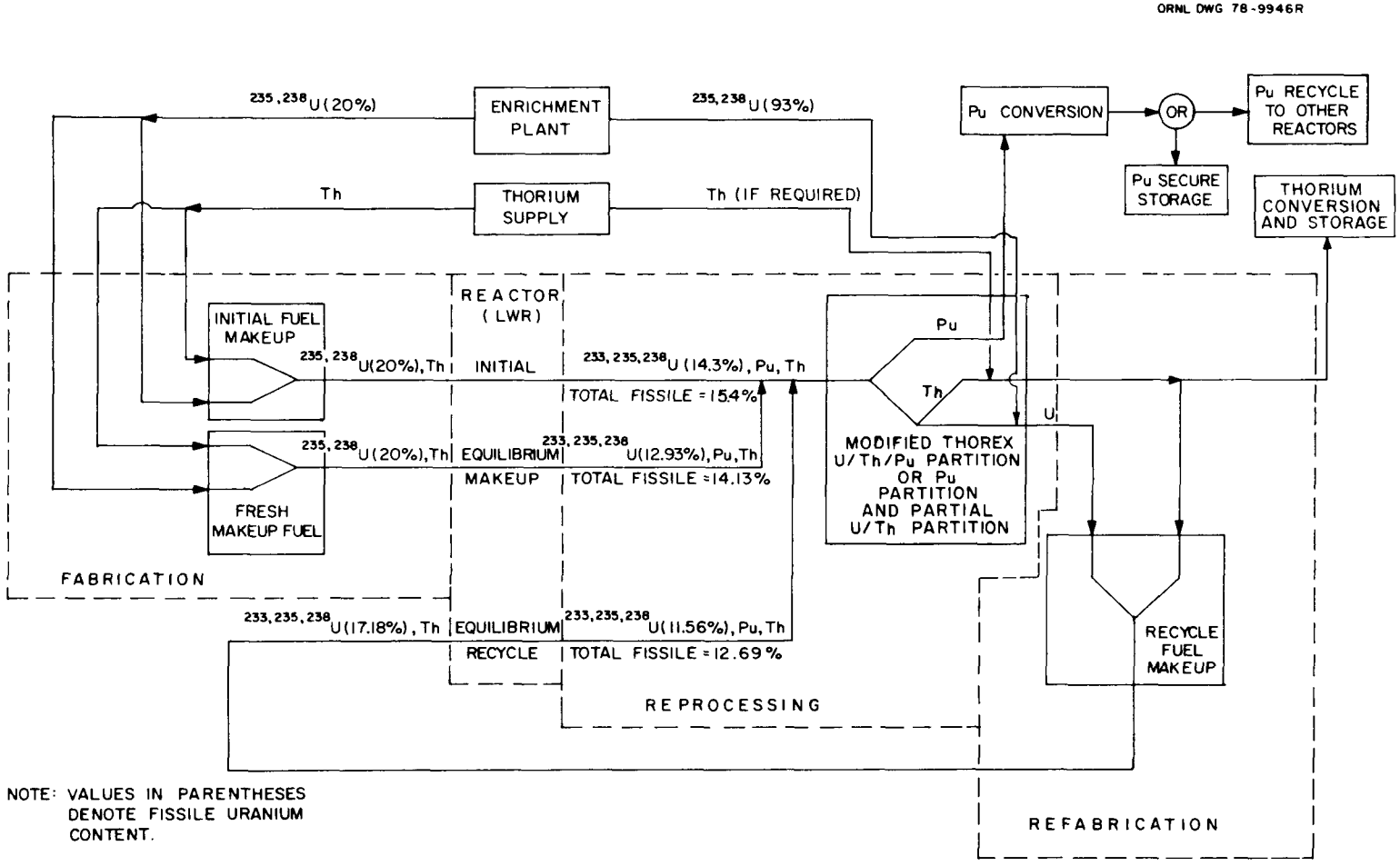


Fig. 2.2. Diagram showing heavy-metal flow in an LWR — denatured ^{235}U-thorium fuel cycle.

cycles. The higher ^{238}U inventory also necessitates a lower thorium inventory, thus decreasing ^{233}U production. A more viable option is to employ highly enriched uranium as makeup to fortify the recycle fuel; the remaining replacement fuel is prepared in the fresh fuel fabrication plant from 20 wt % ^{235}U and thorium. Alternatively, makeup uranium could be supplied from 20% enriched material in separate fuel rods.

Nuclear reactions during irradiation of thorium fuels produce small quantities of ^{232}U that is chemically inseparable from ^{233}U. The ^{232}U isotope decays into extremely radioactive daughter products that require remote refabrication of the recycle fuel. The use of highly enriched uranium for makeup has the disadvantage of requiring the transport of weapons-grade material from an enrichment facility; alternatively, the enrichment plant could be located in the secure energy center.

2.3 LWR — Denatured (^{233}U,Th)O_2 Fuel with Recycle[2]

This fuel cycle (shown in Fig. 2.3) is similar to the one described immediately above; however, there are important differences.

The ^{233}U-thorium cycle is designed to conserve uranium resources by utilizing ^{233}U instead of ^{235}U to provide initial and makeup fissile inventories. It is presumed that ^{233}U is available from an external source such as a thorium-blanketed LMFBR or from converter reactors (LWRs, HWRs, SSCRs) that utilize plutonium-thorium fuel; all supplying reactors use highly enriched makeup.

The initial fuel charge is prepared in a secure center by blending appropriate quantities of ^{233}U, ^{238}U denaturant, and thorium. The presence of small quantities of ^{232}U in the ^{233}U necessitates remote fabrication to protect personnel from energetic gamma emissions from the ^{232}U decay chain. The fuel blend is adjusted to a fissile content of about 12 wt % to make it unsuitable for weapons fabrication.

One-third of the reactor inventory is removed annually for reprocessing. After a cooling period of about 180 days, the spent fuel is treated in a modified Thorex process to remove fission products and partition

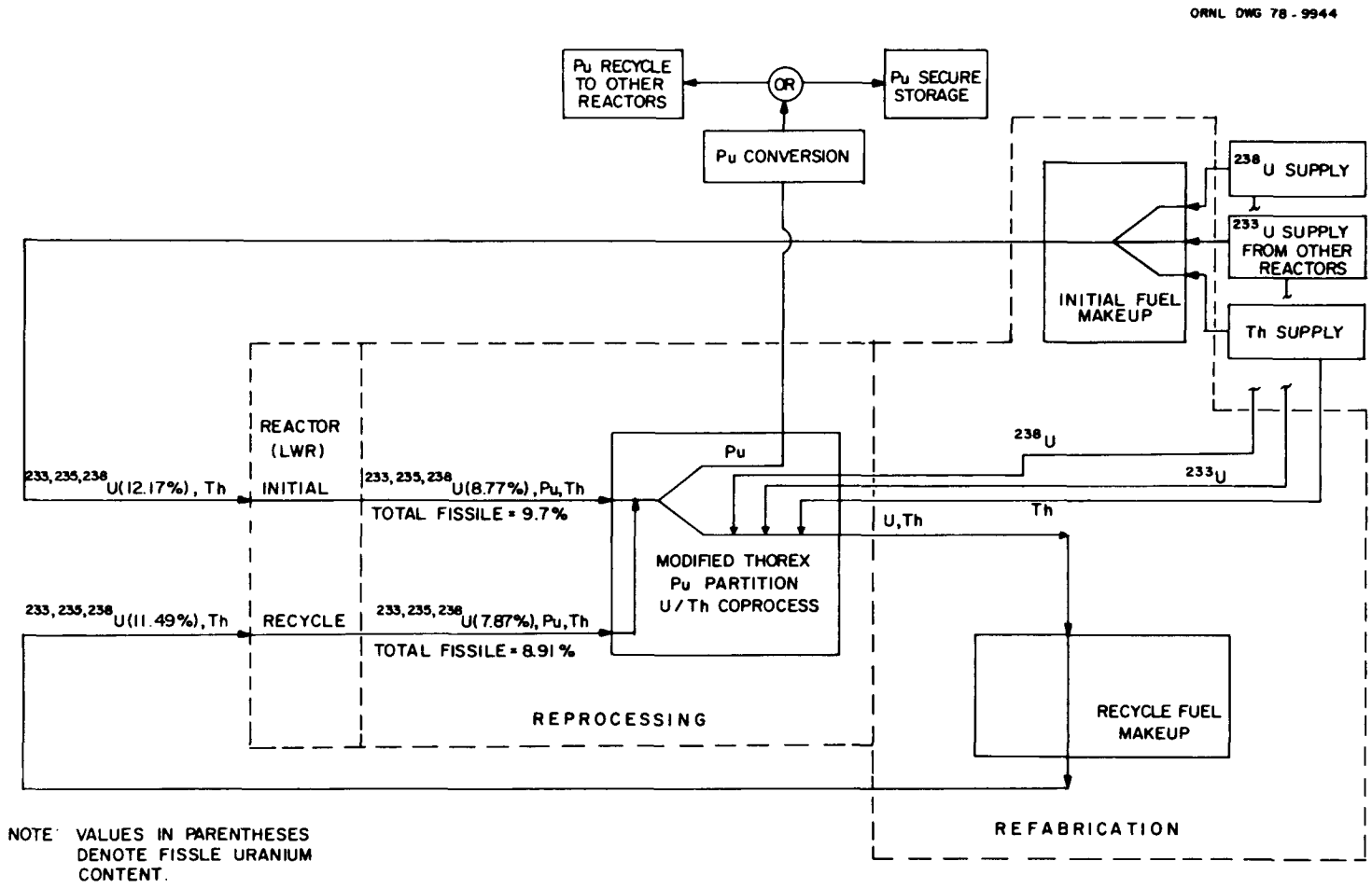


Fig. 2.3. Diagram showing heavy-metal flow for an LWR — denatured ^{233}U-thorium fuel cycle.

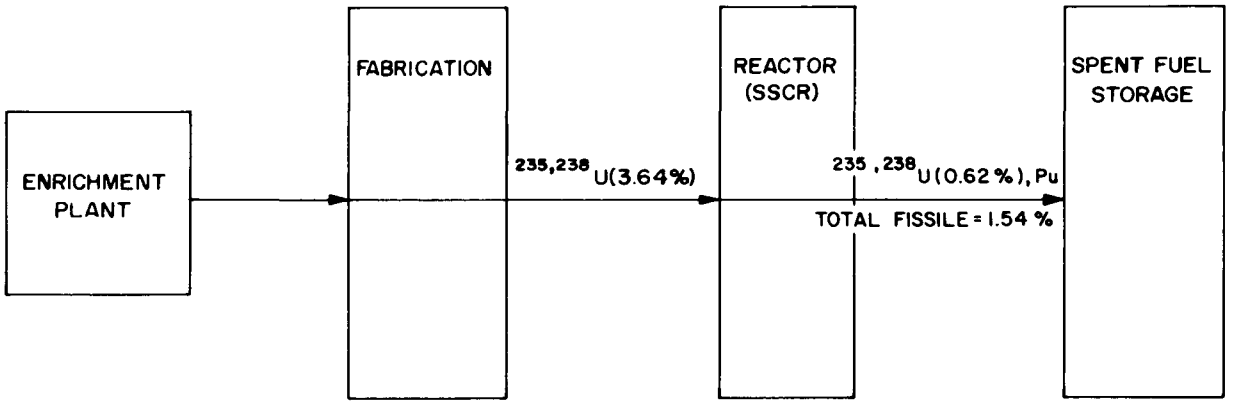
plutonium from uranium and thorium. Uranium and thorium are coprocessed and fortified with makeup ^{233}U, thorium, and ^{238}U denaturant; this mixture is then converted to oxides and sent to refabrication to be made into new fuel assemblies. Fissile uranium in the refabricated fuel is about 12 wt % ^{233}U, based on total uranium. The reprocessing plant must be located in a secure center in order to protect the plutonium product. After separation by solvent extraction, this product is converted to oxide (PuO_2) for secure storage or for use in other reactors in the secure center.

The denatured ^{233}U-thorium fuel cycle offers diversion and proliferation resistance over the denatured ^{235}U-thorium cycle in that the ^{233}U has high activity associated with it. In each case, the recycle fuels are comparably unattractive for diversion due to their ^{232}U content, which increases the radiation hazard, and due to the denaturing effect of ^{238}U. The ^{233}U-thorium fuel cycle, however, requires an appropriate symbiotic balance between reactors that produce ^{233}U by burning plutonium and the LWR that burns this ^{233}U and produces plutonium.

2.4 SSCR — UO_2 Fuel; Optimized, Once-Through Fuel Cycle[3]

The SSCR makes use of LWR reactor technology and components that are already highly developed. In this concept, the excess neutrons generated during operation of the reactor are directed preferentially to capture in fertile material, rather than in control poisons, by alteration of the heavy water--light water ratio in the coolant. Thus, more fissile material (plutonium in this cycle) is produced and burned than in conventional LWRs, having the overall effect of decreasing the demand on uranium resources. The similarity of the SSCR and the well-developed LWR increases the probability of near-term acceptance and deployment of the SSCR concept.

The once-through fuel cycle for the SSCR is relatively simple since only fresh fuel fabrication and spent fuel transportation plus storage are required (Fig. 2.4). Uranium, enriched to about 3.6 wt % ^{235}U, is received at the fabrication plant and fabricated into fuel elements. After exposure in an SSCR to about 50,000 MWd/MTHM, the irradiated fuel



NOTE: VALUE IN PARENTHESES DENOTES FISSILE URANIUM CONTENT.

Fig. 2.4. Diagram showing heavy-metal flow for an SSCR once-through fuel cycle.

is discharged and stored until its future disposition or treatment is decided. The spent fuel storage facility must be located in a secure center; however, the reactor does not have this restriction.

The optimized, once-through cycle of the SSCR attains about 50% higher fuel burnup than the conventional LWR by employing five in-core batches rather than three; however, the burnup is no higher than that of the improved LWR. Fuel is exposed for five years, with 20% of the fuel being removed annually. At equilibrium, the discharged fuel contains about 0.62 wt % fissile uranium and sufficient additional plutonium to increase the total fissile content to about 1.5 wt %.

2.5 SSCR — Denatured (^{233}U,Th)O_2 Fuel with Recycle[4]

This fuel cycle (Fig. 2.5) is proposed in order to incorporate the conservation advantages for uranium resources that come from recycling fuel and utilizing thorium. It differs from that of the LWR ^{233}U-thorium cycle shown in Fig. 2.3 only in the concentrations of irradiated and makeup fuel.

The initial and makeup ^{233}U for this reactor is furnished from an exogenous supply (e.g., an LMFBR blanketed with thorium) located in a secure energy center with the reprocessing facility. The radioactivity of the ^{232}U and its daughter products in ^{233}U fuel necessitates remote fabrication of both initial and makeup fuels. After irradiation to about 33,000 MWd/MTHM, the fuel is removed from the reactor, cooled about 180 days, and reprocessed. Since the fuel charge contained ^{238}U denaturant, a significant quantity of plutonium is present in the irradiated element. Reprocessing utilizes a modified Thorex process to separate plutonium from uranium and thorium, and from fission products. Plutonium is recovered in purified form and converted to an oxide (PuO_2); it may then be recycled to the same reactor that supplies the ^{233}U or may be held in secure storage. Coprocessed uranium and thorium are fortified with makeup ^{233}U, ^{238}U denaturant, and thorium and subsequently converted to mixed oxides [(U,Th)O_2] for refabrication into recycle fuel. The recycle material has a fissile content of about 12 wt %.

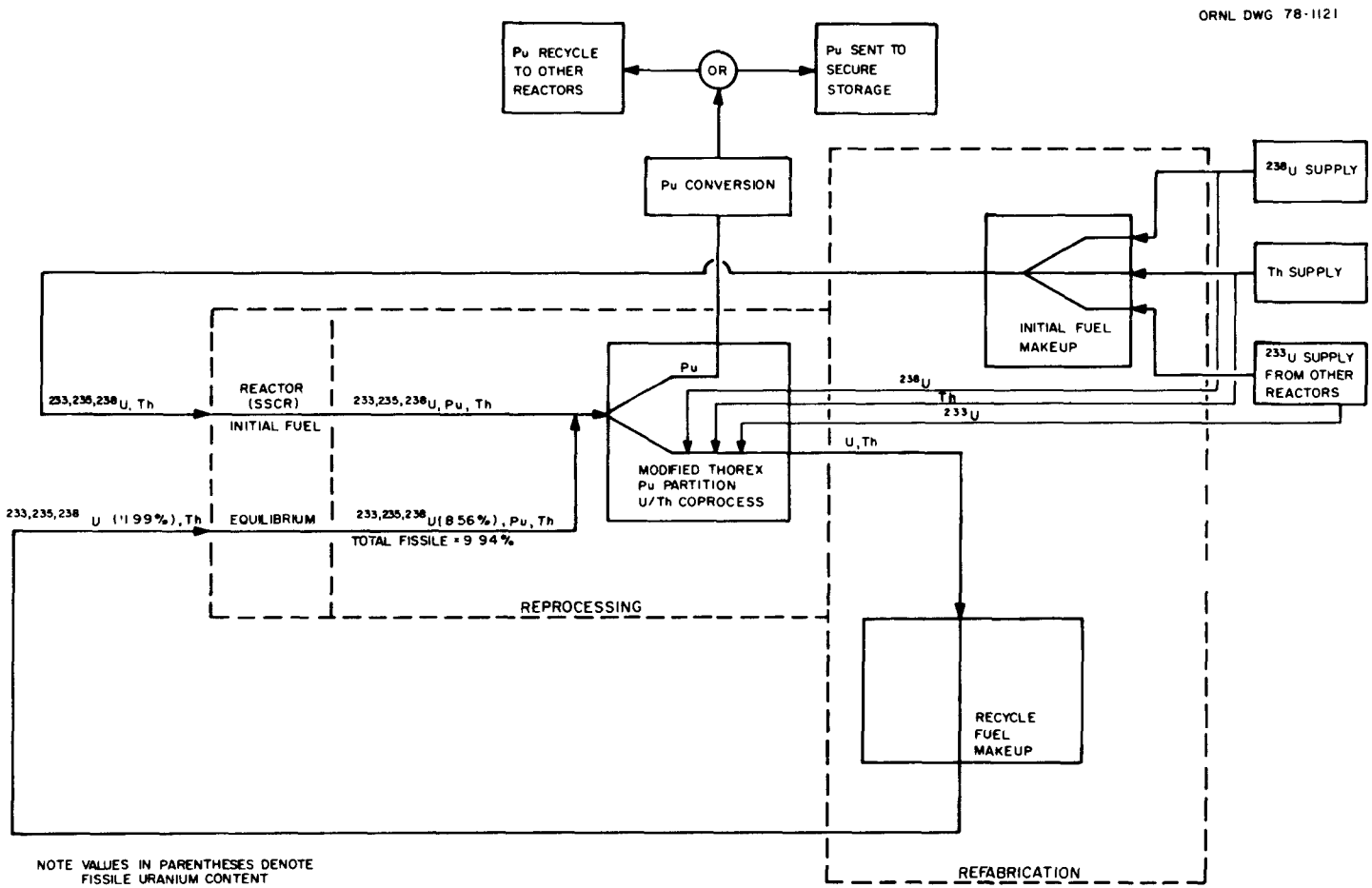


Fig. 2.5. Diagram showing heavy-metal flow for an SSCR — denatured ^{233}U-thorium fuel cycle.

One-third of the core inventory is removed annually for reprocessing. At equilibrium, the spent fuel contains about 8.6 wt % fissile uranium, based on total uranium, and sufficient plutonium to increase the total fissile material to about 9.9 wt %, based on total uranium and plutonium content. Diversion and proliferation resistance of this fuel cycle is enhanced by the highly radioactive nature of recycle ^{233}U fuel and by isotopic dilution of the fissile material below the concentration that is practical for a weapon.

2.6 HWR — UO_2 Fuel; Once-Through Fuel Cycle[5]

The HWR once-through fuel cycle has been extensively developed in Canada and is in commercial use there as well as in other parts of the world. Hence a large technological base exists for further development and rapid deployment of this family of reactors. The inherent neutron economy of a heavy-water-moderated and -cooled system augurs well for increased energy production per unit uranium resource.

The fuel management scheme for this system is shown in Fig. 2.6. The facilities include only an enrichment plant, a fresh fuel fabrication plant, the reactor, and a spent fuel storage facility. The initial reactor charge is fabricated from natural uranium; however, subsequent loadings are prepared from uranium that is about 1.2 wt % enriched. The use of enriched uranium allows fuel burnups of about 20,000 MWd/MTHM, almost three times the burnup of current designs. The ^{235}U content of the discharged fuel is equivalent to the enrichment of tailings from an isotope enrichment plant.

On-power refueling is used to remove irradiated fuel elements from the reactor and reload makeup fuel, giving a semicontinuous flow of spent fuel. This fuel, which contains about 0.5 wt % fissile material, is transported to a spent fuel storage facility where it is held pending a decision as to final disposition. The entire core is changed on a 2.7-year cycle.

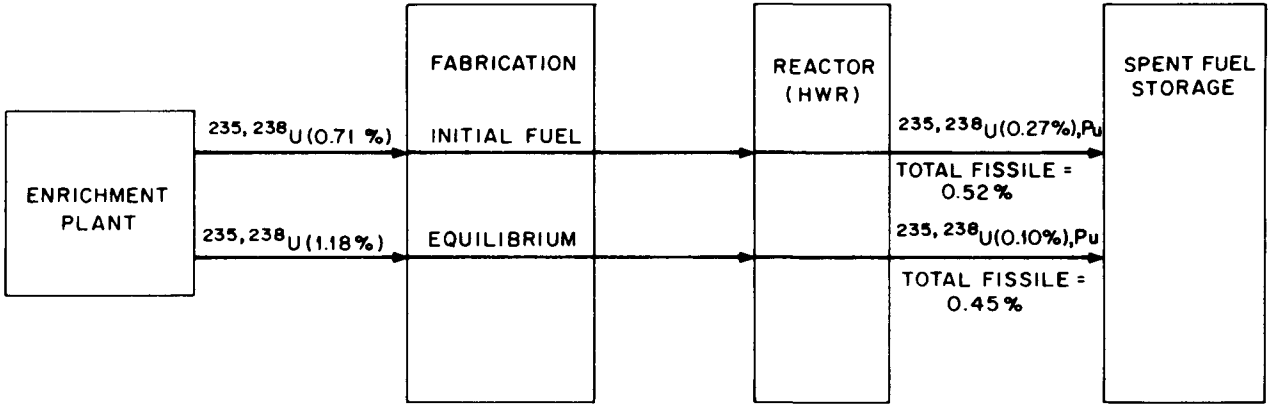


Fig. 2.6. Diagram showing heavy-metal flow for an HWR — low-enriched, once-through fuel cycle.

2.7 HWR — Denatured (^{233}U,Th)O_2 Fuel with Recycle[6]

A second way to use HWRs to extend uranium resources is through the denatured ^{233}U-thorium fuel cycle. This cycle requires nominal modifications to the HWR while making use of the broad technological base that has already been developed for these reactors. An exogenous source of ^{233}U is needed; and, as noted above, this material could be produced in thorium-blanketed LMFBRs* or from converter reactors (LWR, SSCR, HWR) that utilize plutonium-thorium fuel. The time for such LMFBR introduction is much further in the future than the introduction of these HWRs.

Fuel cycle facilities needed for the denatured ^{233}U-thorium fuel cycle are shown in Fig. 2.7. These consist of the reactor, a reprocessing plant, and a fabrication/refabrication plant. Radioactivity from the ^{232}U and its daughters in the makeup fuel requires that the initial loading be fabricated in a remotely operated refabrication plant. Denaturant (^{238}U) and fissile material (^{233}U) are blended to a fissile concentration of about 12 wt % and then mixed with thorium to the required fuel composition of 88% thorium and 12% uranium.

The continuous refueling feature of HWRs allows more freedom in choosing initial enrichments and fuel exposures than is available to the designer of reactors that are fueled at discreet intervals. The HWR on the denatured uranium-thorium cycle may operate economically at a lower fuel exposure than, for example, an LWR and achieve a higher conversion ratio. Makeup fuel costs are thereby lower, but recycle costs are higher. Higher-burnup fuel cycles incur lower reprocessing and refabrication costs but higher makeup fuel costs. Thus, there is an economic trade-off between recycle costs and makeup fuel costs. This fuel cycle, which has not been optimized, is characterized by a fuel exposure of 15,000 MWd/MTHM.

Irradiated fuel is treated by a modified Thorex process to remove fission products and to partition plutonium from uranium and thorium. Plutonium is converted to oxide (PuO_2) for secure storage or recycle to a transmutation reactor for breeding additional ^{233}U. Uranium and thorium

*LMFBRs are liquid-metal fast breeder reactors.

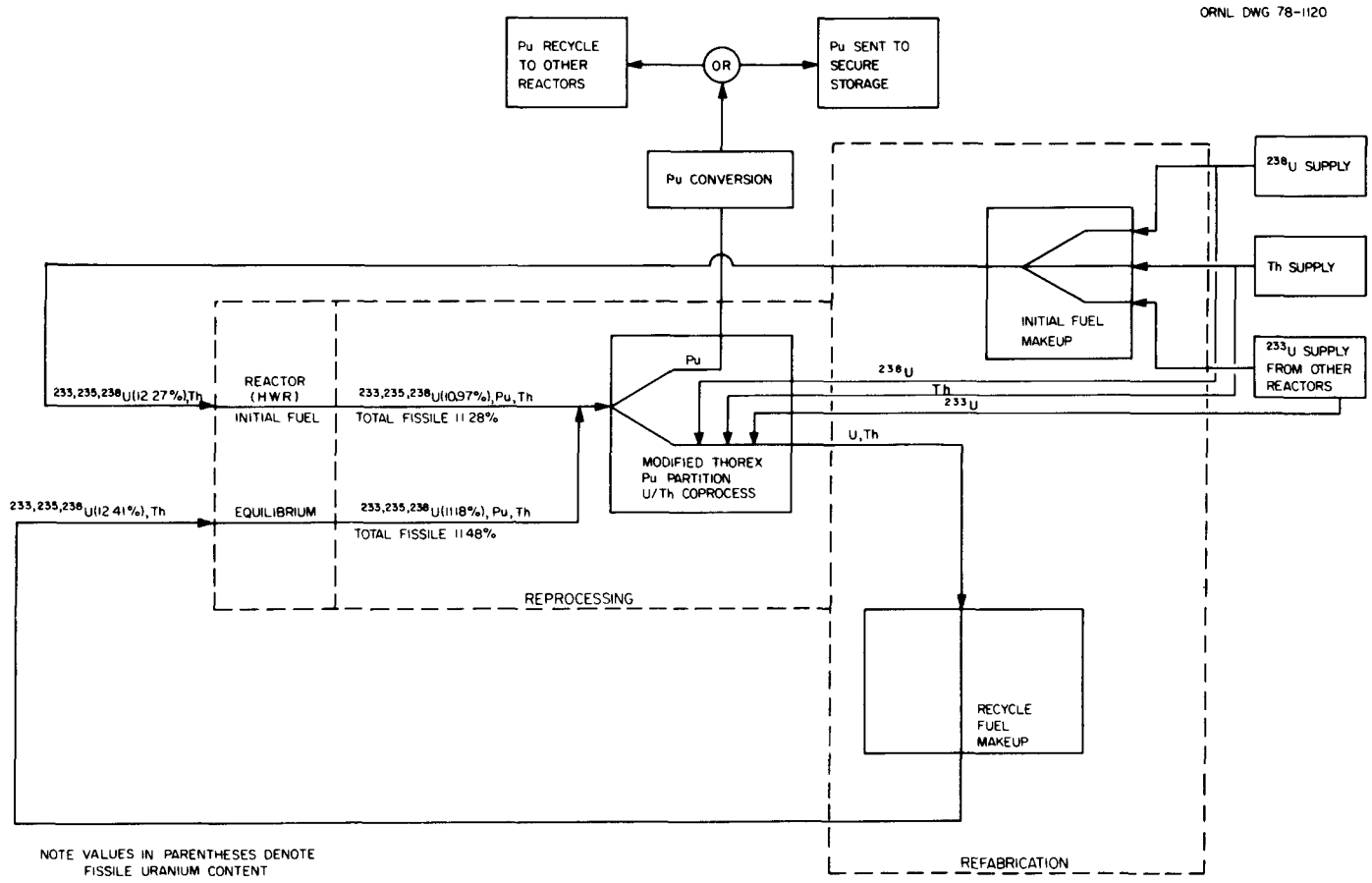


Fig. 2.7. Diagram showing heavy-metal flow for an HWR — denatured ^{233}U-thorium fuel cycle.

are coprocessed, blended with the required amounts of ^{238}U denaturant, thorium, and makeup ^{233}U, and sent to refabrication where recycle fuel is reconstituted.

This fuel cycle requires a secure center for both the reprocessing facility and the transmutation reactor that supplies makeup ^{233}U. A pure plutonium stream is produced in reprocessing, and it must be retained in the secure center. The isotopic enrichment of recycle fuel is kept sufficiently low to be inutile for a weapon.

2.8 LWR/HWR — UO_2 Fuel; Tandem Fuel Cycle[7]

Considerable energy remains in the approximately 0.6 to 0.8% enriched fuel that is discharged from LWRs, and conventional reprocessing may be used to recover this material for reuse. However, a potential option exists for recovering a large portion of the energy content of this irradiated fuel without conventional reprocessing by using spent LWR elements to fuel an HWR, which can operate efficiently on the low-enriched fuel. The LWR/HWR tandem fuel cycle is designed to fuel HWRs with irradiated LWR fuel via mechanical disassembly of LWR fuel elements and reassembly of the fuel-bearing portions into HWR fuel elements. Figure 2.8 depicts this fuel cycle.

The components that make up this fuel cycle are an enrichment facility, a fabrication plant, a remotely operated refabrication plant, an LWR, an HWR, and a spent fuel storage facility. The initial LWR fuel is enriched to about 2.2 wt % ^{235}U and given an average irradiation exposure of 33,000 MWd/MTHM before discharge; makeup fuel has a higher enrichment (~3.0 wt % ^{235}U) for overriding accumulated fission products. Irradiated fuel has a total fissile content (uranium + plutonium) of about 1.14 wt % for the initial fuel and 1.5 wt % for the equilibrium fuel. These enrichments are approximately twice that needed for operation of an HWR.

Some LWRs employ demountable fuel assemblies that permit ready removal of fuel rods; however, the difference in length between LWR assemblies and HWR assemblies (3.8 m and 0.5 m, respectively) will require a redesign of

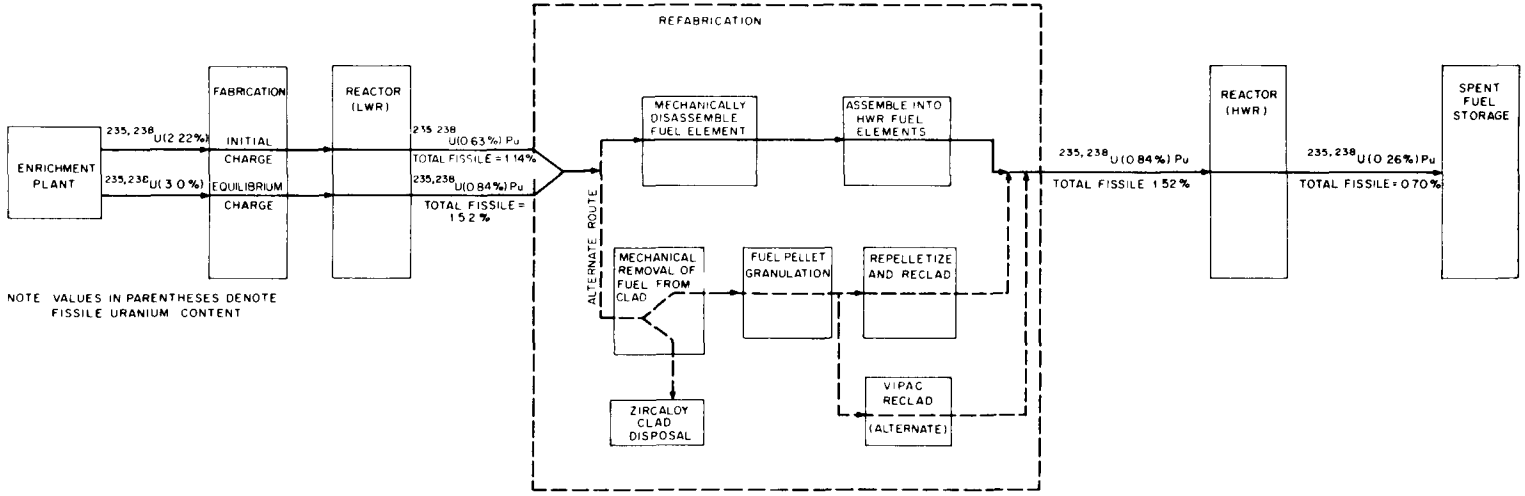


Fig. 2.8. Diagram showing the heavy-metal flow in the LWR-HWR tandem fuel cycle.

either the LWR or the HWR fuel elements if this reconfiguration option is used. Nevertheless, the direct exchange of fuel rods is the most feasible method because alternative procedures (Fig. 2.8) that require removal of fuel from the clad need extensive development to avoid intolerable losses and to reconstitute a fuel that meets HWR specifications. Furthermore, the refabrication cost would be relatively high, and this approach would not be economically justified. However, an estimated additional 15,000 MWd/MTHM of energy can be obtained from the spent LWR fuel before it is sent to spent fuel storage.

The tandem fuel cycle uses only low-enriched uranium; and, with the exception of the fresh fuel, all fuel transported is highly radioactive material. The only facility that needs to be located in a secure center is the spent fuel storage basin.

3. FRESH FUEL FABRICATION

Several generic processes are possible for the fabrication of water reactor fuels. Of these, three are considered commercially feasible: the conventional powder-pellet process, the Sphere-Cal process, and the Sphere-Pac process. Simplified functional flow diagrams for these processes are presented in Figs. 3.1-3.3. Principal functional areas are designated by numbers, and interfacing areas or activities are designated by dashed-line boxes.

Although the mechanical and nuclear designs of the fuel required for the LWR, SSCR, and HWR exhibit important differences, processing of feed materials into finished fuel assemblies for any type of reactor is similar from a functional point of view.

Brief descriptions of the functional areas of the three above-mentioned processes are presented in the following paragraphs.

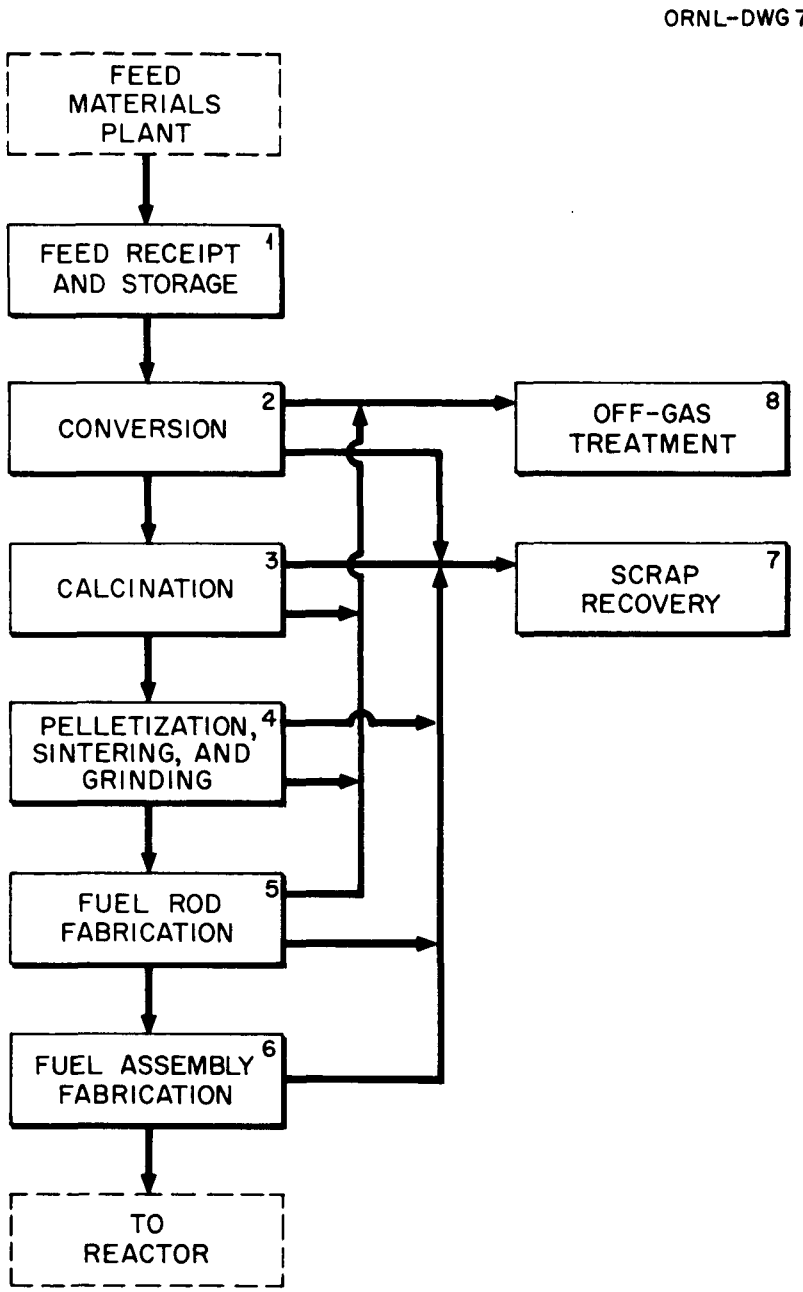


Fig. 3.1. Functional flow diagram for LWR, SSCR, and HWR fuel fabrication by the conventional powder-pellet process.

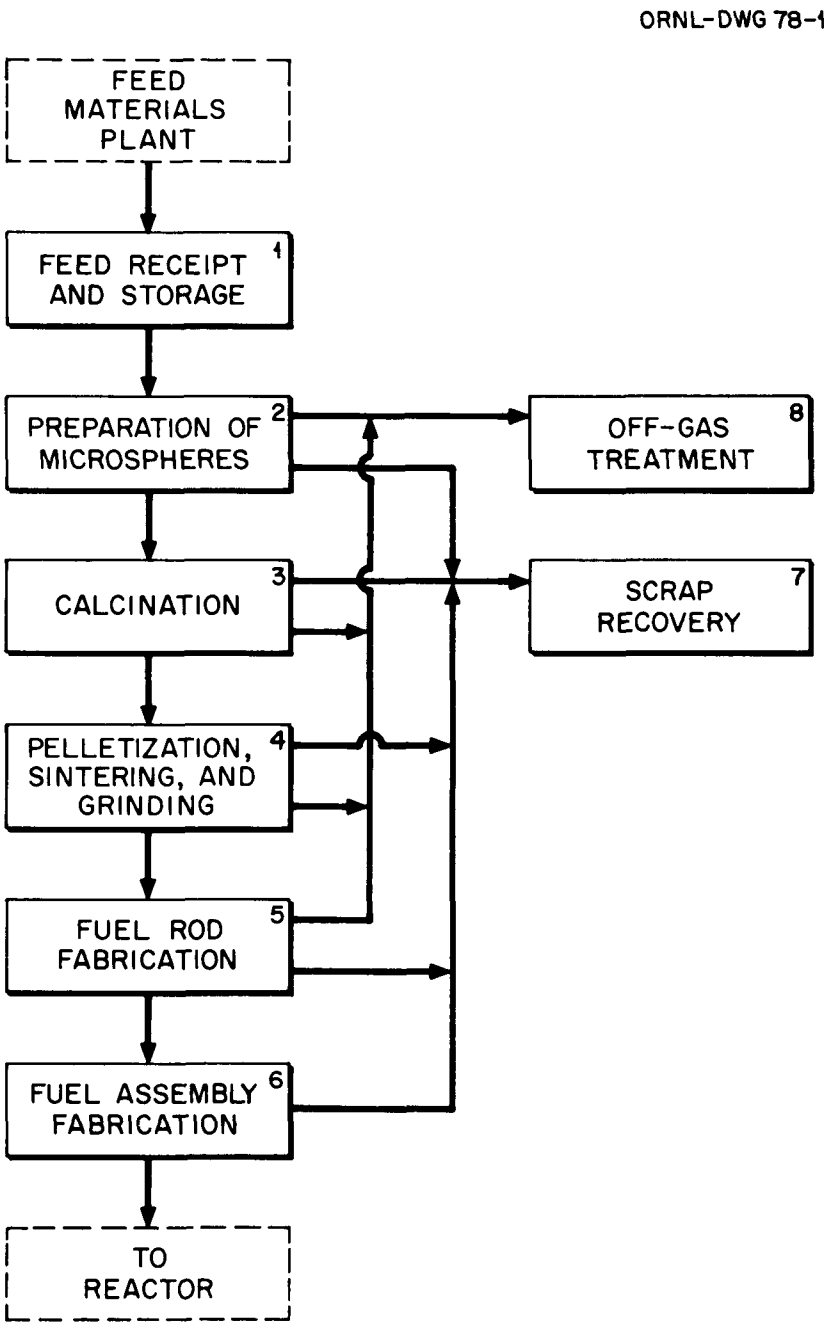


Fig. 3.2. Functional flow diagram for LWR, SSCR, and HWR fuel fabrication by the Sphere-Cal process.

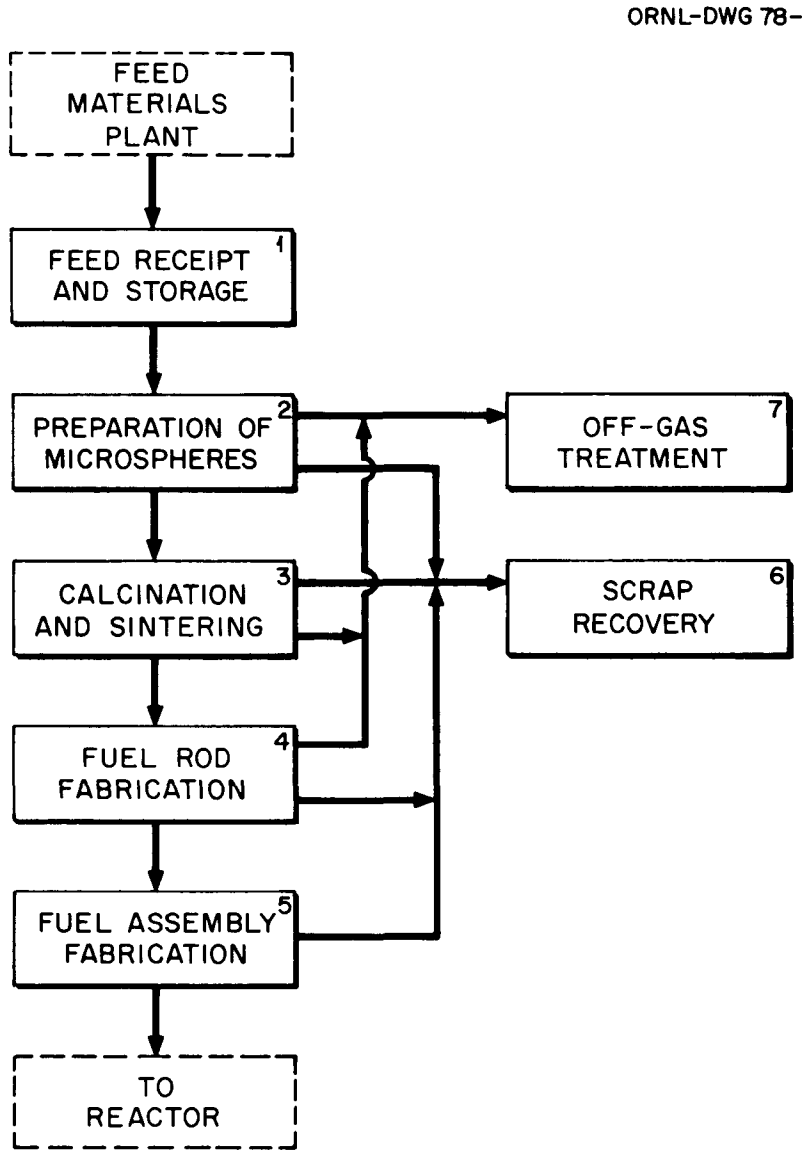


Fig. 3.3. Functional flow diagram for LWR, SSCR, and HWR fuel fabrication by the Sphere-Pac process.

3.1 Powder Pellet Process

3.1.1 Feed receipt and storage

The feed materials to a fuel fabrication facility are uranium as uranium hexafluoride (UF_6) and thorium as thorium nitrate [$Th(NO_3)_4$]. These materials are received at the fuel fabrication facility and stored until needed for processing into fuel.

3.1.2 Conversion

The basic process of this functional area is the conversion of feed materials to intermediate products that are suitable for calcination to metal oxide powders.

For the conversion of uranium, UF_6 is hydrolyzed to form uranyl fluoride (UO_2F_2) and the uranium is precipitated as ammonium diuranate (ADU) by addition of ammonia. The resulting ADU slurry is centrifuged for use as feed to the calcination furnace for conversion to uranium dioxide (UO_2).

Thorium, received as $Th(NO_3)_4$, is precipitated from aqueous solution as thorium oxalate by addition of oxalic acid. The thorium oxalate is filtered, washed, and air-dried to make suitable feed to the calcination furnace for conversion to ThO_2.

3.1.3 Calcination

The ADU from the conversion furnace is fed to a calcination furnace, where it is calcined in a reducing atmosphere (hydrogen) to UO_2. The resulting UO_2 powder is milled and placed in interim storage for subsequent blending and pelletizing.

Thorium oxalate is calcined in air to form ThO_2 powder. This powder is also milled and placed in interim storage for subsequent blending and pelletizing.

3.1.4 Pelletizing, sintering, and grinding

Appropriate combinations of lots (batches) of UO_2 powder or, in the

case of uranium-thorium fuels, UO_2 and ThO_2 powders are blended, combined with binders, pore formers, and press lubricants, and then pressed into right circular cylindrical pellets. These metal oxide pellets are subsequently sintered in a hydrogen atmosphere to form high-density pellets which are ground to size on a centerless grinder.

3.1.5 Fuel rod fabrication

The sintered and ground metal oxide pellets are loaded into a Zircaloy tube that has one end cap, or plug, welded in place. The necessary plenum adjustment is made, and plenum hardware is inserted. The top of the loaded fuel rod is fitted with an end plug, which is welded to the tubing. The loaded HWR or BWR fuel rod is filled with helium to a slightly positive pressure prior to the final plugging and welding operation. In the case of PWR or SSCR fuel, the rod is pressurized to several atmospheres with helium prior to the plugging and welding operation. Alternatively, the pressurization may be accomplished by means of a pressurization hole that is sealed by welding.

3.1.6 Fuel assembly fabrication

The appropriate number of fuel rods is accumulated for fabrication into a fuel assembly. The PWR or SSCR fuel assembly is fabricated by loading the rods into a mechanical assemblage of spacer grids and control-rod guide tubes to form a square array. End fittings are then mechanically joined or welded to the control-rod guide tubes to complete the assembly. A slightly different technique is used for BWR fuels. In this case, the spacer grids are secured by means of one or two "water rods," which are empty tubes equipped with tabs to position the spacer grids. The BWR fuel assembly does not contain control-rod guide tubes.

The HWR fuel assembly is fabricated by placing the fuel rods in a circular array in a fixture and fixing this configuration firmly by brazing end plates to the assemblage. Rod spacing is maintained via spacer pads welded to individual fuel rods.

3.1.7 Scrap recovery

The considerable value of the various forms of scrap material generated in the fuel fabrication steps make a recovery system necessary. The recovery processes are subdivided, depending on the nature of the scrap. They include the simple recovery and recycle of materials such as the unloading of pellets from defective fuel rods, recycle of various scrap forms such as broken "green" pellets, dissolution of scrap materials to form new feed solutions, and dissolution and purification (by solvent extraction) of scrap materials to form new feed solutions. The employment of a scrap recovery system is very important to the profitable operation of a fuel fabrication plant.

3.1.8 Off-gas treatment

All major process steps except fuel assembly involve operations that may result in the production of gaseous wastes or airborne contamination. All gaseous streams are filtered through high-efficiency absolute (HEPA) filters to prevent the release of airborne particulates. Gaseous wastes from chemical operations are vented to gas treatment systems or scrubbers to prevent their release to the atmosphere.

3.2 Sphere-Cal Process

3.2.1 Feed receipt and storage

The feed materials for the Sphere-Cal process are uranium and thorium as uranyl nitrate and thorium nitrate respectively. These materials are stored prior to use in microsphere preparation.

3.2.2 Preparation of microspheres

Microsphere preparation is accomplished by the gel precipitation process in which solid gel microspheres are formed from aqueous solutions of the metal nitrates. Although the gel microspheres can be prepared by various approaches, an internal gelation process is chosen here as the reference method.

Feed solutions of uranyl nitrate and/or thorium nitrate are prepared by vacuum evaporation of the aqueous solutions. Nitric acid is removed by this technique, and acid-deficient uranyl nitrate or thorium nitrate solutions are formed. Alternatively, aqueous solutions of uranyl nitrate and/or thorium nitrate are contacted with an organic extractant to replace some nitrate ions with hydroxyl ions from the extractant in order to form acid-deficient solutions. The resulting aqueous solutions are concentrated by evaporation, cooled to 0°C, and mixed with an aqueous solution of urea and hexamethylenetetramine (HMTA). The gel microspheres are formed by ejecting the cold solutions from capillary tubes into an organic liquid held at 70 to 90°C. The gel microspheres are washed, dried in air at 70 to 80°C, and used as feed to the calcination furnace.

3.2.3 Calcination

For the preparation of UO_2 or UO_2-ThO_2 microspheres, the calcination is performed in an atmosphere of hydrogen to ensure reduction of the uranium. For ThO_2 microsphere preparation, the gel microspheres are calcined in air. Finished microspheres are collected in interim storage and held for the pelletizing operation.

3.2.4 Pelletizing, sintering, and grinding

The basic operations here are essentially the same as those involved in preparing pellets from powders. Batches of appropriate combinations of UO_2, ThO_2, or UO_2-ThO_2 microspheres are blended and combined with any necessary binders, pore formers, or press lubricants, and then pressed into right circular cylindrical pellets. The pellets are sintered in a hydrogen atmosphere and ground to size on a centerless grinder.

3.2.5 Fuel rod fabrication

Fuel rod fabrication using sintered pellets prepared from microspheres is identical to that using pellets prepared from powders (see Sect. 3.1.5).

3.2.6 Fuel assembly fabrication

Methods for fabricating fuel assemblies using microsphere fuel pellets are identical to those using pellet fuels prepared from powders (see Sect. 3.1.6).

3.2.7 Scrap recovery

Scrap or cull microspheres from the various microsphere forming or treatment operations are dissolved and returned to the feed solutions to be recycled through the process to form acceptable microspheres. Other scrap recovery operations are identical to those employed in the preparation of fuel by the powder-pellet process (see Sect. 3.1.7).

3.2.8 Off-gas treatment

An off-gas treatment system is required to handle gaseous wastes from the microsphere forming and drying operations; otherwise, the treatment of wastes in the Sphere-Cal process is the same as that used in the powder-pellet process (see Sect. 3.1.8).

3.3 Sphere-Pac Process

3.3.1 Feed receipt and storage

Feed materials for the Sphere-Pac process are the same as those for the Sphere-Cal process; also, the feed receipt and storage requirements are identical.

3.3.2 Preparation of microspheres

The preparation of microspheres for the Sphere-Pac process involves the same operations employed in the Sphere-Cal process, except that microspheres of two or more controlled size ranges are required.

3.3.3 Calcination and sintering

Dried microspheres from microsphere preparation are calcined and

sintered in a reducing atmosphere of hydrogen to form a high-density product. These high-density microspheres are used as the feed material to fuel rod fabrication.

3.3.4 Fuel rod fabrication

Batches of coarse and medium-sized microspheres are received from the sintering operation and blended to form a uniform lot. These blended microspheres are fed by gravity from a feed hopper into a vertically oriented rod that is gently vibrated during filling. A screen is placed over the stack of coarse and medium microspheres to secure the stack, and fine microspheres are poured through the screen and allowed to infiltrate into the interstitial positions of the coarse and medium microspheres. The screen is removed, appropriate plenum hardware is added, and the rod is plugged and welded in essentially the same fashion as that used for the pellet fuels except that the Sphere-Pac fuel rod is vertically oriented. Another method consists of feeding particles of three different sizes simultaneously into the fuel rod followed by vibration for packing.

3.3.5 Fuel assembly fabrication

The operations required for fabricating a fuel assembly are identical to those used for the powder-pellet and Sphere-Cal fuels (see Sect. 3.1.6).

3.3.6 Scrap recovery

Scrap recovery in Sphere-Pac fuel fabrication is essentially the same as that in Sphere-Cal fuel fabrication; however, sintered microspheres rather than sintered pellets are part of the scrap stream.

3.3.7 Off-gas treatment

Off-gas treatment is the same as in Sphere-Cal fuel fabrication operations (see Sect. 3.2.8).

4. SPENT FUEL REPROCESSING

Once-through fuel cycles, by definition, do not require reprocessing of the spent fuel. Therefore, this section describes only the reprocessing of denatured uranium-thorium fuels. The remaining system, the tandem PWR-HWR cycle, could be treated by chemical reprocessing, but the most practical treatment seems to be by the fuel reconfiguration option which does not require reprocessing. The reconfiguration option requires some type of separation of the fuel values from the fuel element hardware; for this reason, it is discussed last. Since the step sequence in fuel reprocessing is the same for the three reactor types (LWR, HWR, and SSCR), an overall description is given only once. Two fuel cycles, using ^{235}U-thorium and ^{233}U-thorium as the initial fuel, have been proposed for the LWR. The ^{233}U-thorium fuel imposes more limitations on reprocessing than does ^{235}U-thorium fuel. However, since the uranium concentration in the discharged fuel is about the same for both cycles (6.4% for ^{235}U and 7.0% for ^{233}U loading), the fuel reprocessing is not appreciably different; thus the process described applies equally to both cycles.

The spent fuel reprocessing has been divided into five operations: head end, solvent extraction, conversion, off-gas treatment, and waste treatment (Fig. 4.1) for convenient analysis.

4.1 Head End

The head-end operation (Fig. 4.2) is designed to convert the heavy-metal values in the used fuel element to a form suitable for solvent extraction. First, the heavy end sections of the fuel element are removed; then the fuel section is chopped into 1- to 2-in. lengths, which are treated for tritium removal. Although a method for removing tritium from thorium-based fuels has not been developed, laboratory experiments indicate that a high-temperature "soak" will reduce the tritium content to acceptable levels. The noble gases are also released during this

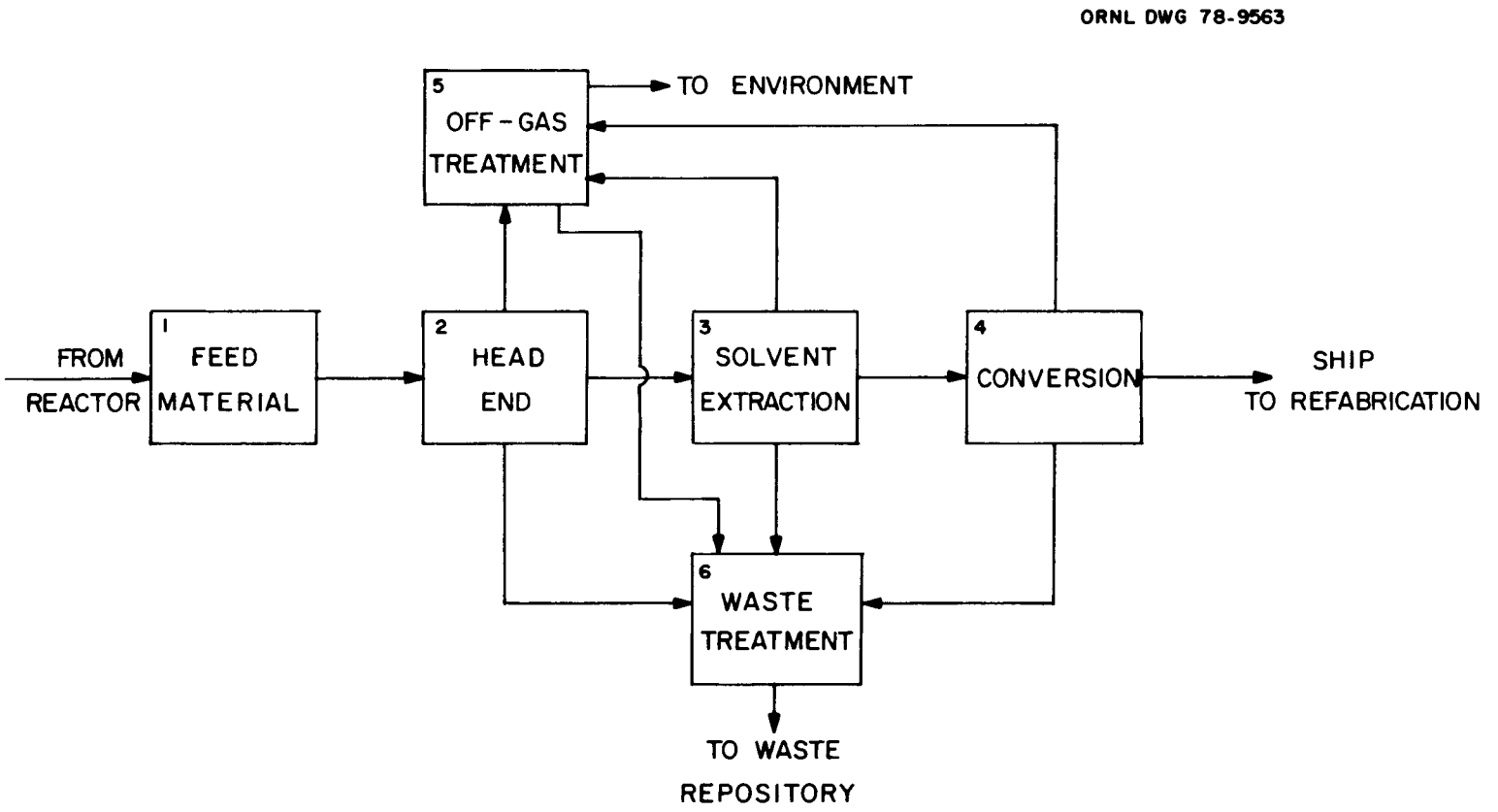


Fig. 4.1. Fuel reprocessing plant: Level 1 functional flow diagram.

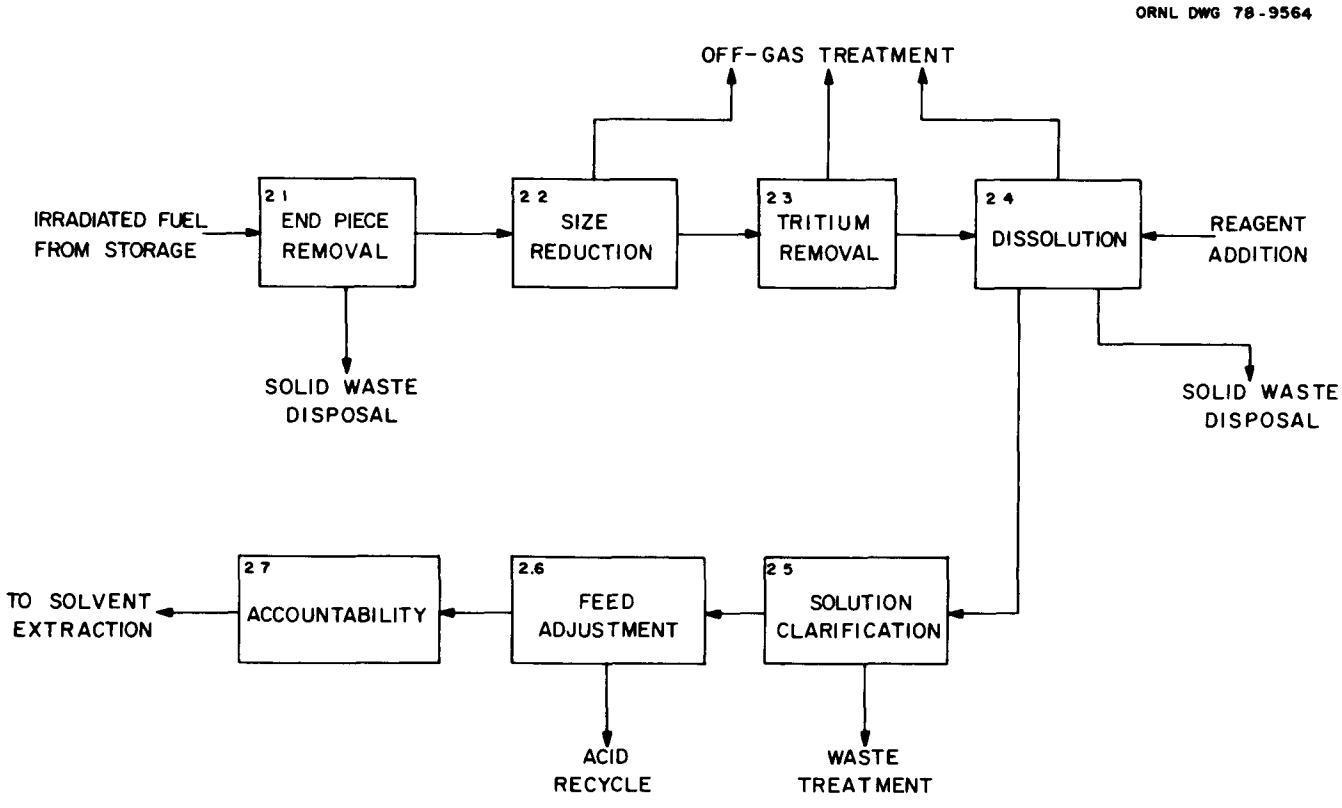


Fig. 4.2. Fuel reprocessing plant: head-end treatment of Zircaloy-clad $(U,Th)O_2$ fuels.

operation. The fuel is then dissolved in nitric acid using fluoride ion as a catalyst. This dissolution process has been demonstrated on a commercial scale for stainless-steel-clad thoria, but there has been concern as to whether problems would be encountered with Zircaloy-clad thoria due to the complexing of the fluoride with zirconium. Recent laboratory experiments, however, indicate that satisfactory results may be obtained by using slightly modified dissolution procedures. Conditions are also chosen to volatilize iodine during dissolution.

The cladding and fission products that do not dissolve (primarily molybdenum and noble metals) are removed from the solution by centrifugation. The clarified solution is adjusted to solvent extraction feed conditions and analyzed for metal values. This is the first point in the process at which satisfactory accountability of the fissionable content of the fuel can be obtained.

4.2 Solvent Extraction

Fuel values are extracted from the fuel solution with tributyl phosphate (TBP) in a kerosene-type diluent, leaving most of the fission products in the aqueous phase. The required operations are shown in Fig. 4.3. Additional fission products are removed by a nitric acid scrub. Plutonium, which is bred in the fuel by neutron capture in ^{238}U, extracts with the uranium and thorium and is selectively removed in a subsequent step by contacting the solvent with a suitable reductant. Plutonium is either transferred to safe storage or recycled to a plutonium-fueled reactor within the secured area. Uranium and thorium are costripped from the solvent and decontaminated further from fission products by a second solvent extraction cycle. Following the second cycle, the fission product activity of the product stream is less than the activity of the daughters of ^{232}U; thus further cleanup is unwarranted. The concentrated product is sent to the conversion process.

If the denatured ^{235}U-thorium LWR fuel cycle option of using highly enriched ^{235}U should be chosen (Sect. 2.2), then a partial partition of thorium from uranium will be required.

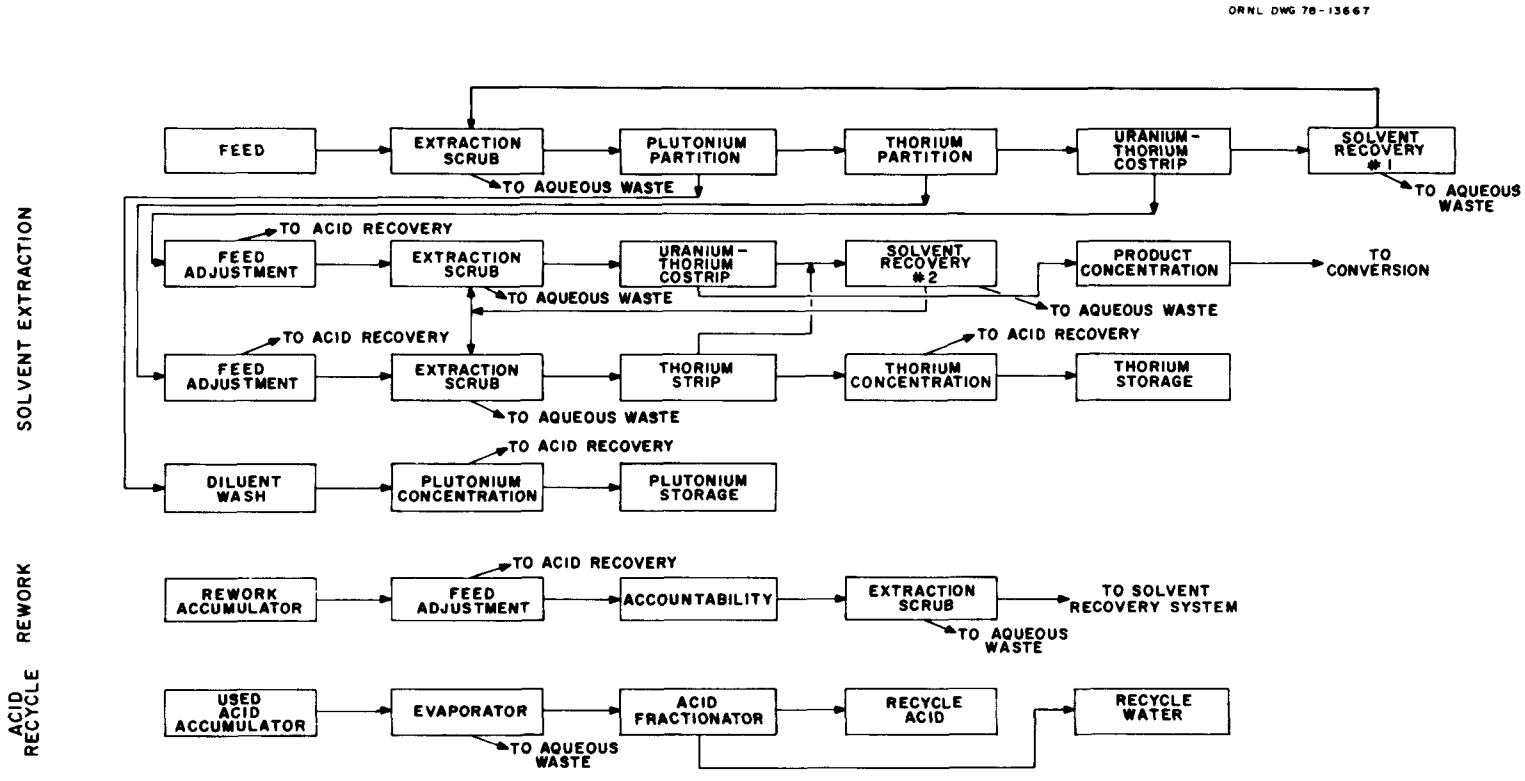


Fig. 4.3. Fuel reprocessing plant: solvent extraction - partial partition of thorium from uranium.

Whenever practical, solutions are reused within the process. Used solvent is washed free of impurities, filtered, and recycled. Streams containing product materials that do not meet specifications are treated to recover uranium and thorium, which are then returned to the main process stream. Aqueous waste streams are evaporated to decrease their volume and to remove both acid and water. The concentration of acid in the distillate is increased by fractional distillation so that the acid can be recycled to the dissolution step. When practical, water and dilute acid are also recycled.

4.3 Product Conversion

The internal gelation process to produce gel microspheres (Fig. 4.4) has been chosen as the basis for conversion since this process yields a stable product suitable for shipment to a fuel refabrication plant. The gel microspheres prepared by this process are a satisfactory feed material for making either pellets or Sphere-Pac fuel rods. The process, which is highly flexible, can be used for any of the proposed fuel cycles. Off-specification materials are returned to the reprocessing plant for recovery, thereby eliminating all wet operations in the fuel refabrication plant.

The initial conversion operations are designated to remove acid from the feed via distillation or steam stripping. The acid-deficient product is mixed with HMTA and urea at about 0°C and injected into a hot organic solution (~85°C) to form gel microspheres. The microspheres are first washed to remove ammonium nitrate and then dried at about 100°C to prepare a shippable product. Finally, the hot organic solution is cleaned and recycled.

4.4 Off-Gas Treatment

The off-gas from the various operations (Fig. 4.5) is treated to remove nitrogen oxides, radioactive gases, and particles. In this study, it was assumed that tritium, krypton, and iodine would have to be

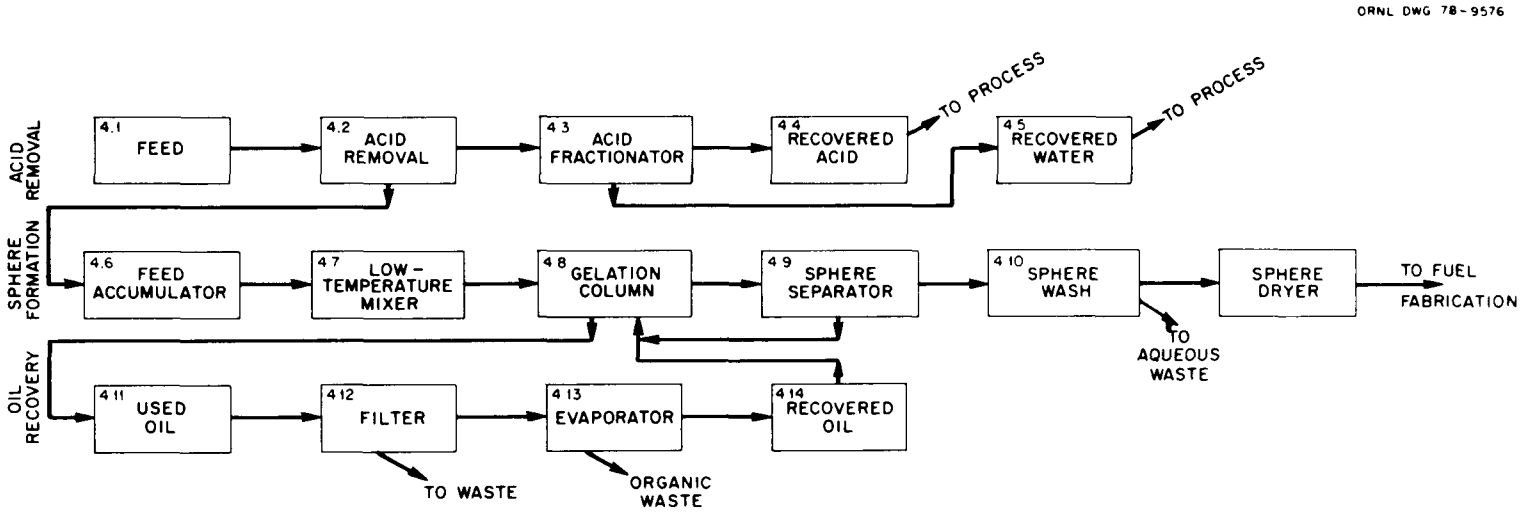


Fig. 4.4. Fuel reprocessing plant: functional flow diagram for product conversion.

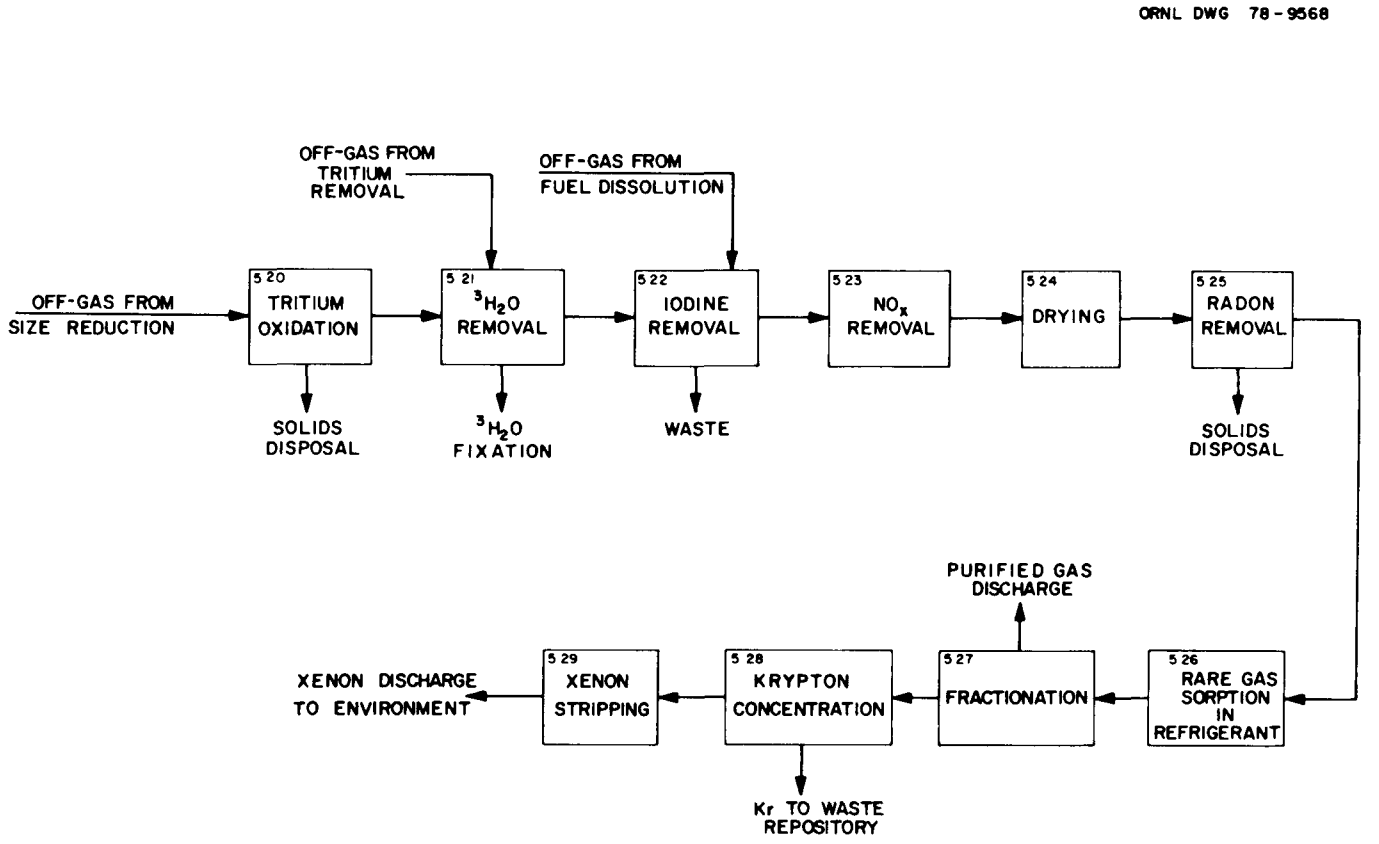


Fig. 4.5. Fuel reprocessing plant: off-gas treatment for $(U,Th)O_2$ fuels.

vaporized from the fuel prior to the dissolution step; these components must then be removed from the off-gas. The use of thorium-base fuel also requires that radon be held up for 10 to 15 min to allow ^{220}Ra (55.6-sec half-life) to decay to nongaseous daughters.

Tritium released during fuel shearing is oxidized to tritium oxide on a platinum catalyst, combined with the tritium oxide released by the high-temperature soak, and adsorbed on molecular sieves. Iodine may be removed by reaction with a mercury solution, by oxidation to iodate in highly concentrated nitric acid (Iodox), or by adsorption on heavy-metal zeolites. The treated gas is subsequently dried and passed through silica gel to adsorb radon. Krypton is separated from inert carrier gases by a low-temperature extraction process using a fluoro-chloro refrigerant as a solvent. The cleaned gas is filtered to remove entrained solids or liquids before being discharged to the environment.

4.5 Waste Treatment

Aqueous waste solutions from the reprocessing operations are evaporated and calcined; the removed water and nitric acid are recycled in the plant (Figs. 4.6 and 4.7). Combustible materials are burned, and the ash is combined with the calcine. Materials to promote the formation of stable glass are added, and the mixture is melted in stainless steel cans to produce vitrified waste for placement in a stable geological formation for long-term storage.

Radioactive solids, such as fuel element end pieces, leached hulls, stable solids containing radioactive gases, and failed process equipment, are compressed or volume reduced, packaged in stainless steel containers, and stored. The radioactive gases are either converted to stable solids, trapped in stable solid matrices, or contained in compressed gas cylinders. As is done for high-level wastes, these materials are prepared for storage in stable geological formations.

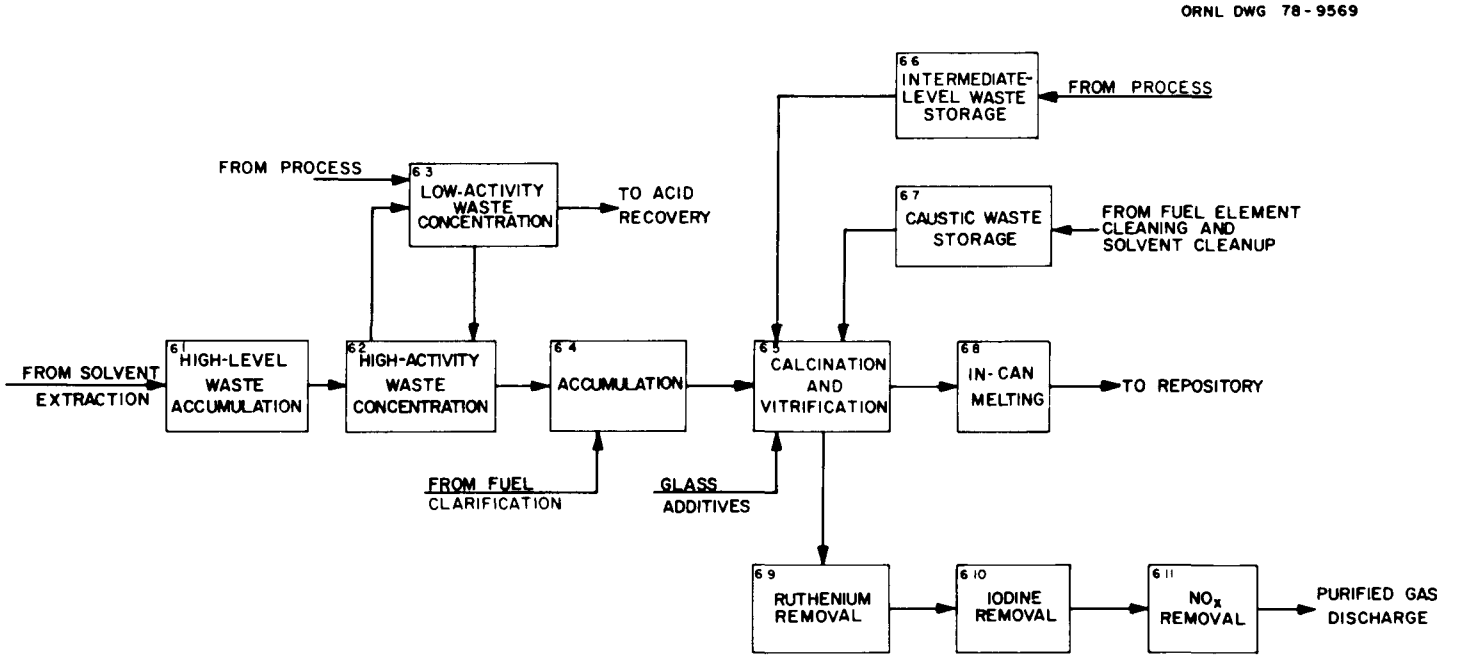


Fig. 4.6. Fuel reprocessing plant: treatment of reprocessing wastes.

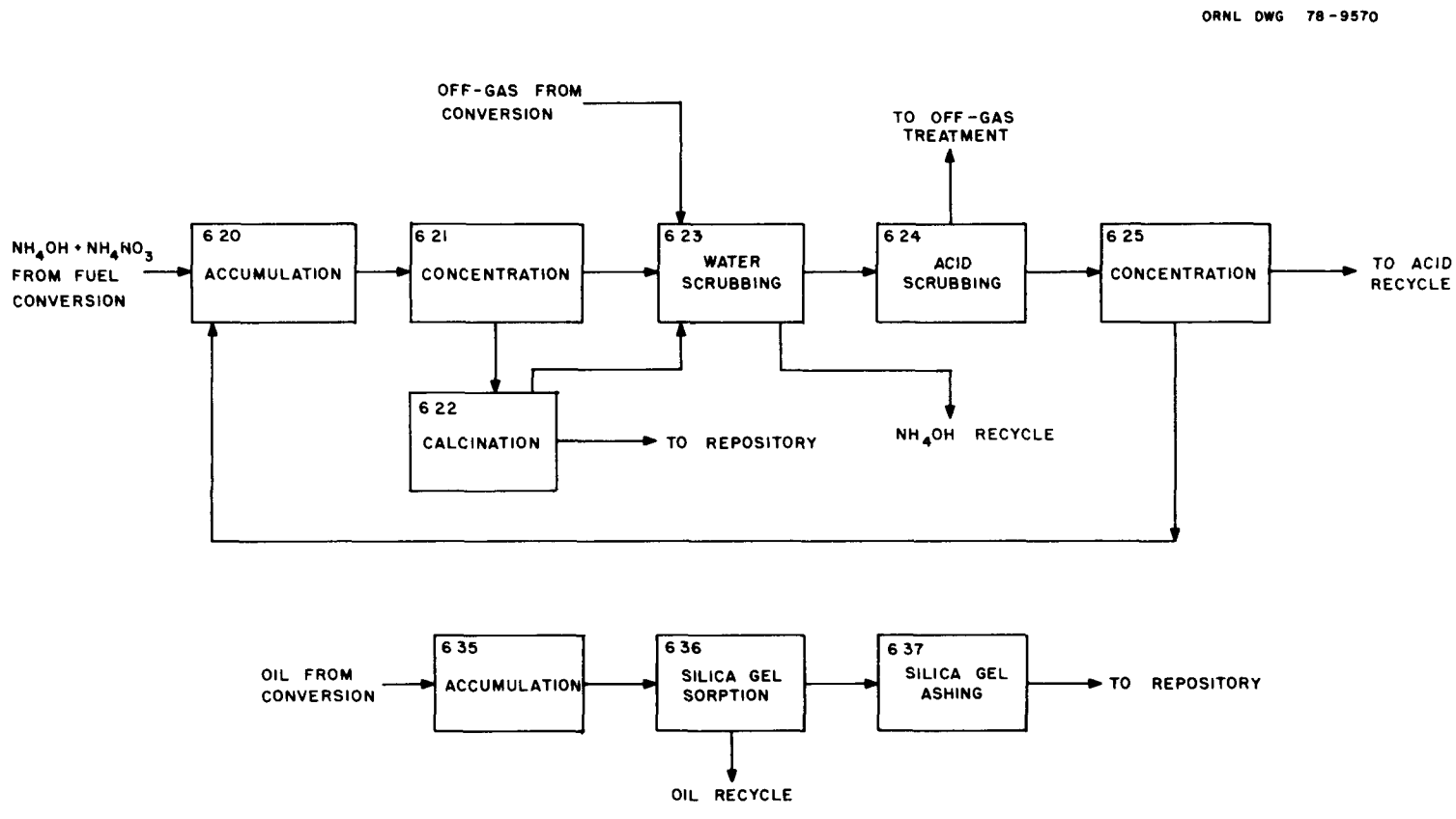


Fig. 4.7. Fuel reprocessing plant: treatment of conversion wastes.

5. RECYCLE FUEL FABRICATION (REFABRICATION)

The refabrication of nuclear fuel for water reactors (LWRs, SSCRs, and HWRs) may be accomplished by the same generic processes considered for fresh fuel fabrication, namely, the conventional powder-pellet process, the Sphere-Cal process, or the Sphere-Pac process. Essentially the same steps are involved in refabrication as in fabrication; differences are, for the most part, in the mode of operation. Fresh fuel may be fabricated in a facility in which contact operations and maintenance are employed. Recycled fuel requires shielding of the fuel materials and equipment to minimize operator exposure, as well as remote operation and maintenance of the equipment. Refabrication and reprocessing facilities may be co-located to allow refabrication wastes and recycle materials to be treated in the reprocessing facility. This arrangement obviates duplication of some facilities and brings about savings in operating and capital costs.

The flow diagrams presented in Figs. 3.1-3.3 for the fabrication of fresh fuel are also applicable to the fabrication of recycle fuel (refabrication). The following paragraphs, which present brief descriptions of the refabrication of fuel by the three generic processes, point out the differences from fabrication of fresh fuel.

5.1 Powder-Pellet Process

5.1.1 Feed receipt and storage

Feed materials to a refabrication plant include uranium (^{233}U, ^{235}U, and ^{238}U) as UO_2 powder, thorium as ThO_2 powder, and uranium-thorium mixtures as UO_2-ThO_2. These feed materials are received and stored at the refabrication facility. Because of the high gamma activity associated with the recycled ^{233}U and thorium, shielded storage facilities are required.

5.1.2 Conversion

Primary conversion is not a part of the refabrication operation. Conversion from nitrate solutions to the metal oxides is performed at the reprocessing plant to provide transportable solids. However, all scrap recycled within the fabrication plant must be processed through the conversion step using remotely operated and maintained equipment.

5.1.3 Calcination

Primary calcination is also performed at the reprocessing plant and thus is not required in the refabrication facility, although some powder conditioning operations are necessary. Calcination of some recycled scrap materials is mandatory.

5.1.4 Pelletization, sintering, and grinding

The metal oxides that are provided by the reprocessing plant are milled and blended, as required, to provide a suitable material for feed to the pellet forming operations. The pelletizing, sintering, and grinding operations are the same as those used in fresh fuel fabrication except that they are performed remotely; remote maintenance of equipment is also required.

5.1.5 Fuel rod fabrication

Fuel rod fabrication using recycled powder-pellet fuel follows the same flowsheet developed for fresh fuel fabrication; however, remote operation and maintenance of equipment are required.

5.1.6 Fuel assembly fabrication

Fuel assembly fabrication using recycle fuel is the same as for fresh fuel, except that remote operation and maintenance of equipment are required.

5.1.7 Scrap recovery

Scrap materials generated in a refabrication facility are essentially the same as those generated in a fresh fuel fabrication facility. The recovery of these materials is complicated by the requirement for remote operation and maintenance of the equipment used in the operation.

5.1.8 Off-gas treatment

In addition to the off-gas treatment requirements of a fresh fuel fabrication facility, the off-gas treatment facilities associated with refabrication operations require systems to ensure the removal of any radioactive particulates or gases that are peculiar to the recycle fuel materials.

5.2 Sphere-Cal Process

5.2.1 Feed receipt and storage

In the fabrication of recycle fuel materials, the primary feed to the refabrication plant is dried microspheres containing uranium (^{233}U, ^{235}U, and ^{238}U) and thorium. The microspheres are stored in a shielded facility for subsequent use in the fuel refabrication operations.

5.2.2 Preparation of microspheres

The preparation of recycle fuel microspheres is performed in the reprocessing facility using the procedure described for preparing fresh fuel microspheres. A similar capability for preparing microspheres from off-specification material generated in the fabrication plant is also required in that facility.

5.2.3 Calcination

The dried microspheres received from the reprocessing plant are calcined in a reducing atmosphere of hydrogen to form a metal oxide product. Remote operation and maintenance of equipment are required for recycle fuel.

5.2.4 Pelletization, sintering, and grinding

Pelletization, sintering, and grinding are identical for fabrication

and refabrication except that remote operation and maintenance of equipment are required for refabrication.

5.2.5 Fuel rod fabrication

Fuel rod fabrication of recycle Sphere-Cal fuel is the same as for fresh Sphere-Cal fuel except that remote operation and maintenance are required.

5.2.6 Fuel assembly fabrication

Fuel assembly fabrication of recycle Sphere-Cal fuel is identical to fuel assembly fabrication of recycle powder-pellet fuel.

5.2.7 Scrap recovery

Scrap recovery of refabricated Sphere-Cal fuel is identical to scrap recovery of fresh Sphere-Cal fuel except, for example, that remote operation and maintenance for refabricated fuel may necessitate a change in process equipment.

5.2.8 Off-gas treatment

Off-gas treatment is fairly similar to that described for refabricated powder-pellet fuel. However, additional gas treatment is required for the microsphere forming and drying operations. Radioactive particle control may also be needed.

5.3 Sphere-Pac Process

5.3.1 Feed receipt and storage

The primary feed for refabrication of fuel by the Sphere-Pac process is the same as that for the Sphere-Cal process, namely, calcined microspheres of uranium and thorium oxides. Alternatively, sintered microspheres of the metal oxides could be provided to the refabrication

facility. However, the sintering operation would probably be performed in the refabrication plant. The calcined microspheres are stored in a shielded facility prior to use in the refabrication operations.

5.3.2 Preparation of microspheres

Primary microspheres are prepared in the reprocessing facility, and only microsphere preparation from internal scrap recycle is considered as part of the refabrication operation.

5.3.3 Calcination and sintering

Calcination of microspheres is performed in the reprocessing facility. The calcination of microspheres prepared from recycled scrap materials requires remotely operated and maintained equipment. The calcined microspheres are sintered in a hydrogen atmosphere to a high-density product in a remotely operated and maintained shielded facility.

5.3.4 Fuel rod fabrication

Sphere-Pac fuel rod fabrication of recycle fuel is identical to Sphere-Pac fuel rod fabrication of fresh fuel except that it has been modified for carrying out refabrication operations with remotely operated and maintained equipment.

5.3.5 Fuel assembly fabrication

Fuel assembly fabrication of recycle Sphere-Pac fuel is performed in the same manner as fuel assembly fabrication of recycle powder-pellet and Sphere-Cal fuels except that it has been modified for carrying out the operations with remotely operated and maintained equipment.

5.3.6 Scrap recovery

The methods for recovering scrap materials from Sphere-Pac recycle fuel processing are the same as for fresh fuel except as modified for remote operations and maintenance of equipment.

5.3.7 Off-gas treatment

The off-gas treatment methods employed in Sphere-Cal recycle fuel fabrication are also applicable to the treatment of off-gases from recycle Sphere-Pac fuel fabrication.

5.4 Quality Control

Due to the requirements for remote operation and maintenance of equipment for refabrication of fuel by any of the alternative processes discussed here, assurance and control of the quality of the fuel become perhaps even more important considerations than in the case of fresh fuel fabrication. Increased complexity will be experienced in most, if not all, of the inspection and testing operations. Material control and accounting may be more difficult, and different nondestructive assay techniques will be needed. For example, the active gamma scanning of fuel rods commonly employed in fresh fuel fabrication must be modified for use in refabrication because of the high-level and varied spectrum of the gamma activity of the irradiated fuel. Although neutron interrogation techniques may be employed, remote operation on a production scale will require some development. Other inspections that are routinely employed in contact operations may become very complicated and time-consuming when applied to remote operations.

The assurance and control of quality are mandatory; and, regardless of the refabrication techniques employed, the techniques by which quality is assured must be demonstrable.

5.5 LWR-HWR Tandem Fuel Cycle

Two options appear to be feasible for the LWR-HWR tandem fuel cycle in order to convert LWR fuel into a form suitable for use in an HWR. The first option is to mechanically reconfigure modified LWR fuel in such a way that it can be accommodated by an HWR. In this reconfiguration

option, LWR fuel assemblies are disassembled, and the fuel rods are reassembled into HWR fuel bundles. The second option requires fabrication of LWR fuel that has been removed from the LWR fuel rods into HWR fuel rods and bundles. Several specific techniques are possible for either of the two options. Some general discussions of the various techniques that might be employed are presented below.

In the reconfiguration option, the basic problem is to disassemble an LWR fuel assembly and to incorporate the fuel rods in an HWR fuel assembly or bundle. The removal of the LWR fuel rods does not represent an overwhelming obstacle in this process since many LWR fuel assemblies can be partially or completely mechanically disassembled. (Only minor design changes would be required to allow most LWR fuel assemblies to be mechanically disassembled.) A typical LWR (BWR or PWR) has fuel rods that are longer than 3.5 m. The HWR fuel rods, which are about 0.5 m long, are incorporated into fuel bundles that are laid end to end in fuel channels of the HWR to form an overall fuel length of about 5.9 m. In order to directly utilize current LWR fuel rods in HWRs, the HWRs would have to be specifically designed to accept the longer fuel rods and fuel assemblies. For this reason, the reconfiguration option would not be the preferred technique.

An alternative in this option is to design the LWR fuel rods so that they can be mechanically disassembled and incorporated into HWR fuel bundles of essentially the same type as that currently being used. A technique that could be used to accomplish this is the use of segmented fuel rods in LWRs. A segmented rod is made by loading a short section of fuel tubing, inserting a connector plug that joins the loaded section of tubing with a second short section of tubing, loading the second section, and repeating the procedure until a finished fuel rod has been fabricated. The plugs that connect the loaded sections of the fuel rod have axial holes to provide a common atmosphere and pressure in all sections of the rod. A common plenum is provided in the uppermost section of the fuel rod. For reconfiguration, LWR fuel rods are removed from the fuel assembly and separated at the connector plugs of the fuel rod segments. The atmosphere of each fuel rod segment is adjusted, as

required, through the axial holes in the connector plugs. The holes are seal-welded after atmosphere adjustments have been completed, and the resulting HWR fuel rods are assembled into an HWR fuel bundle. A simplified flow diagram for the reconfiguration option using segmented fuel rods is presented in Fig. 5.1.

Modifications to the current HWR fuel bundle design should be minimized by use of the segmented fuel rod reconfiguration option. The LWR fuel rod segments could be specifically designed to meet the HWR fuel rod length requirements. The HWR bundle would probably be redesigned to accommodate more fuel rods because of the smaller diameter of the LWR rods. However, the impact of this change in HWR fuel bundle design should be much less severe than the impact of the change in design of and fuel management in the HWR itself, which is required in the reconfiguration option using full-length LWR rods.

The performance of LWR fuel may be influenced by use of segmented rods since small sections of each rod at each segment joint will not contain fuel; the overall fuel rod loading may be decreased for the same reason. Thus, it is anticipated that fuel performance testing of the segmented fuel would be required to qualify it for use in commercial reactors. The performance of spent LWR fuel in HWRs is also uncertain due to differences in operating histories of LWRs from which the fuel for the HWRs may be obtained, as well as differences in fuel burnup from a single LWR discharge batch. This uncertainty appears to exist for all the tandem cycle options but would be much less for reconstituted fuel if it is blended before refabrication.

An unanswered question that is common to both reconfiguration options is whether the integrity of the cladding can be maintained during disassembly and reassembly operations. Steps must be taken to ensure this integrity, as well as the overall quality of the fuel. Thickening the cladding wall will probably be necessary; however, this will tend to decrease the nuclear performance.

The fabrication of segmented fuel rods for LWRs is expected to be more complex and expensive than fabrication of unsegmented rods. On the other hand, satisfactory production rates, at slightly higher

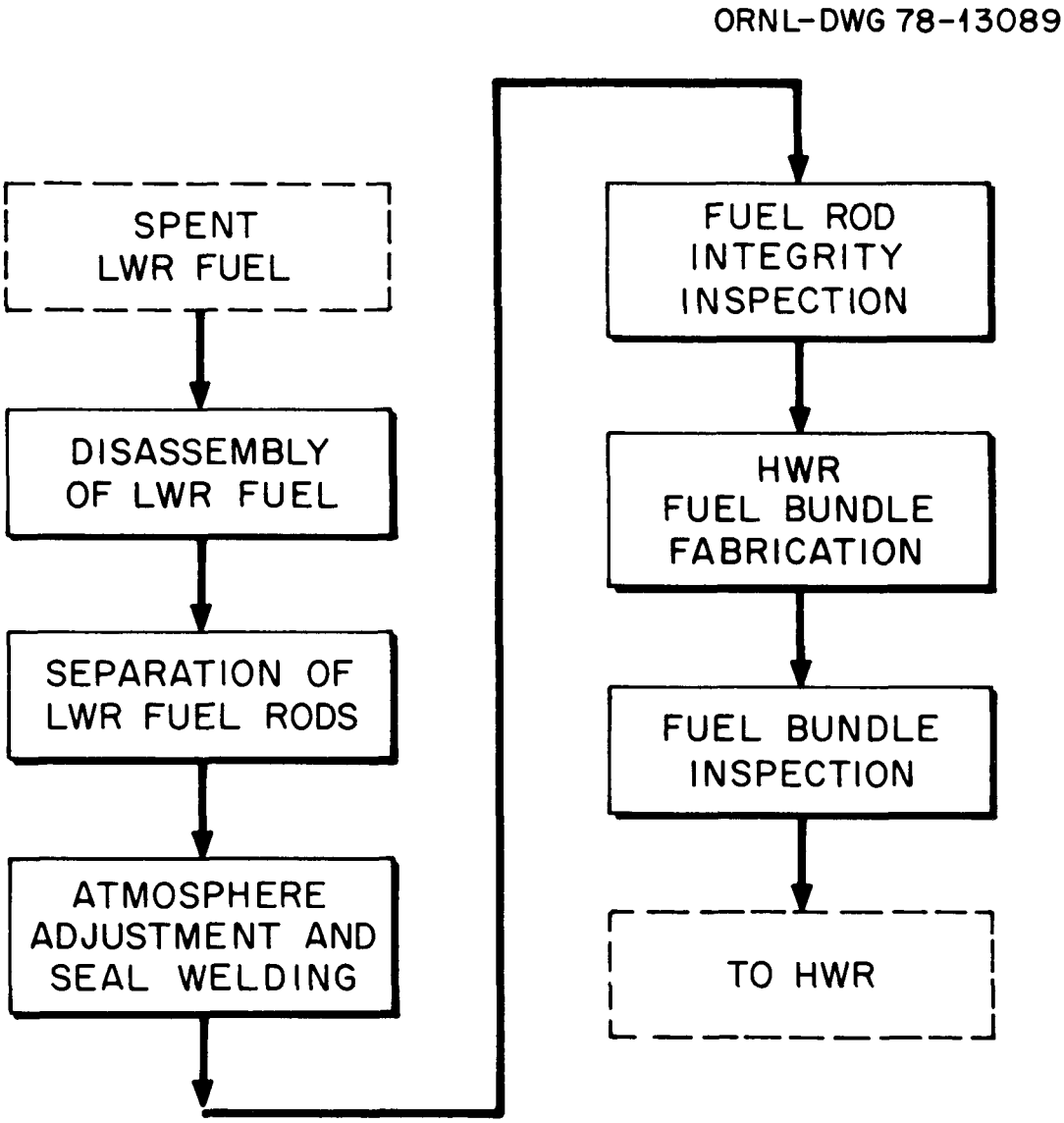


Fig. 5.1. Reconfiguration of LWR fuel into HWR fuel.

incremental costs, should be attainable since the LWR fuel is unirradiated and is fabricated in an unshielded contact facility. The benefits gained in the HWR portion of the cycle due to the use of segmented fuel rods would have to outweigh the penalties imposed by segmented rod fabrication in LWR fresh fuel plants in order to make this approach acceptable. However, power losses in the LWR operation may be economically controlling.

In the refabrication option the fuel from the LWR rods is removed from the cladding and refabricated into HWR fuel. A variety of more or less effective techniques may be used to remove the fuel material. For example, the LWR fuel rods may be cut into short lengths and the fuel particles mechanically removed. Unfortunately, studies at ORNL[8] suggest that fuel recovery is poor using this technique unless the rods are cut in very short segments. For example, 97.8% of the UO_2 in fuel rod segments was released when 0.25-in. cuts were made, whereas only 66.4% was released for 0.375-in. cuts. Oxidation of the UO_2 is another possibility for removal of the fuel. In this method, the fuel rods are cut into short segments and then heated in an oxidizing atmosphere. This treatment results in release of the volatile materials in the fuel and oxidation of the UO_2 to U_3O_8. (Note that this method is applicable only to uranium fuels or fuels that have a relatively high percentage of uranium.) The change in oxidation state is accompanied by a change from a compact pellet to a loose powder that should be relatively easy to remove from the cladding.

Regardless of the removal method, the fuel must be oxidized and then reduced to UO_2 for further preparation for loading into fuel rods. The UO_2 powder may be used for fabrication of pellets, or it may be milled, sized, and sintered as a powder, loaded into HWR fuel rods, and compacted via vibration (the Vipac process). Although the Vipac process has been used to produce reactor fuel, considerable work is anticipated to develop a process that can be routinely used in the preparation of fuel and can be qualified for commercial reactor use. From the standpoint of early commercialization, refabrication by oxidation of the UO_2 fuel contained in LWR rods, followed by reduction and pelletization, appears to be preferred over the Vipac process. A simplified flow diagram for the

refabrication option using fuel pellet refabrication is presented in Fig. 5.2. The refabrication cost is expected to be relatively high and generally not balanced by benefits from additional HWR exposure.

Based on the brief analyses presented above, the most direct method for utilizing LWR fuel in HWRs is to use segmented fuel rods in LWRs, which can be separated, inspected for cladding integrity, vented and purged with helium to adjust the atmosphere and pressure, seal welded, and incorporated into HWR fuel bundles. This technique avoids the redesign of HWRs required by use of full-length LWR fuel rods, greatly simplifies the refabrication process, and minimizes handling of the fuel. Furthermore, this method should be more economical than the separation of spent fuel from the cladding; however, the use of segmented fuel rods in LWRs will have to be demonstrated and licensed, and the spent fuel will have to be qualified and licensed for use in HWRs. The performance evaluation and licensing are required, of course, regardless of the reconfiguration or refabrication option that might be employed.

6. EVALUATION OF FUEL TRANSPORT TO AND FROM WATER REACTORS

Transportation forms a vital link in any fuel cycle. Transportation requirements for radioactive materials in the fuel cycle will continue to grow as the fuel cycle is completed and the industry expands. This section discusses the shipment of fresh or recycle fuel to, and spent fuel from, the reactors considered in the present study. The following treatment provides a somewhat limited view of the transportation of radioactive materials in each of the various fuel cycles since shipments of other materials are not considered. However, fuel element shipments are most significant in any considerations involving the operation of the reactors.

Spent fuel and other radioactive materials can be transported safely by utilizing currently available technology. Although it is possible that each type of reactor would employ its own cask design for

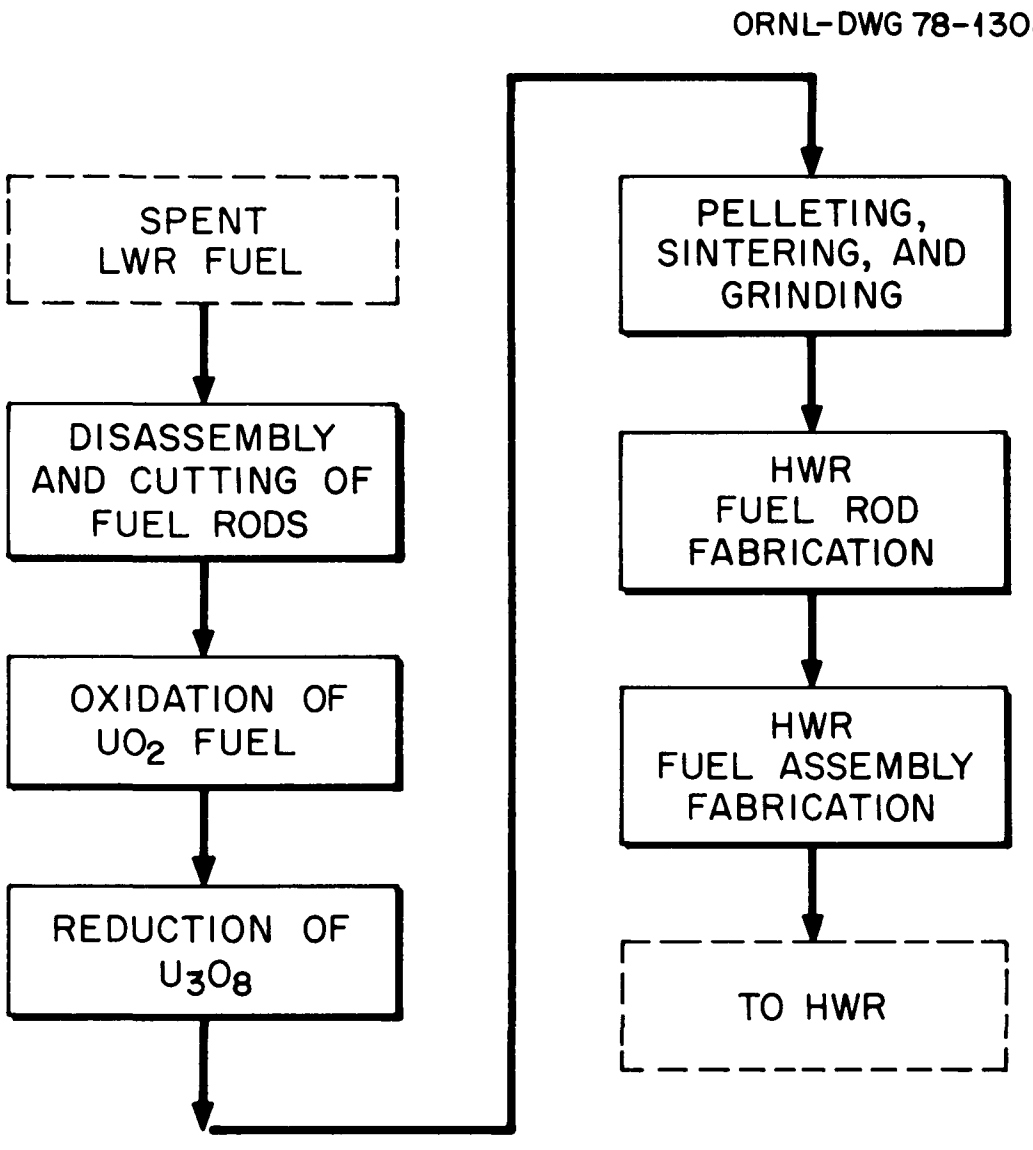


Fig. 5.2. Refabrication of HWR fuel from LWR fuel.

fresh, recycle, or spent fuels, there is sufficient similarity in current LWR fuels to enable existing package designs to be used as a basis for estimating fuel transportation requirements of the fuel cycles studied.

6.1 Federal Regulations[9]

In the United States, the shipment of radioactive materials is governed by the Nuclear Regulatory Commission (NRC) and the Department of Transportation (DOT). The actual division of responsibilities is described in the Memorandum of Understanding between the two agencies.

The NRC regulations establish criteria for the preparation of shipments of fissile and large quantities of radioactive material. These criteria stipulate the design of the container, limitations on contents and necessary safeguards to protect against overheating, release of radioactive material, loss of shielding, and accidental criticality. The DOT regulations include packaging requirements of radioactive materials, including allowable dose rates from the package, limitations of contents, and labeling requirements. Requirements for tie-down of containers and other matters affecting safety are also included.

The regulations require that both packages and casks function in a specified manner when exposed to a series of tests or conditions. These tests and conditions, described below, will produce a response in the package similar to that which would be expected during exposure to a normal transport environment or a severe accident sequence.

6.1.1 Normal conditions of transport

The normal conditions of transport specified in both the NRC and the DOT regulations include ambient temperatures that vary from 55°C in the sunshine to -40°C in the shade, an ambient pressure of 0.5 atm, vibration normally incident to transport, a 30-min water spray, a short free fall onto an unyielding surface, a penetration test, and a compression test. During exposure to such conditions, the package must operate in the manner intended by the designer and approved

in writing by the approving authority. No leakage is permitted either from or into the package; the dose rate is limited to 10 mrem/hr at a distance of 6 ft from the accessible surface of the vehicle. The package must be designed to prevent a criticality reaction from occurring, and the temperature of the accessible surface of the package cannot exceed 82°C. Although normal conditions of transport involve environmental extremes, their impact on package design is seldom as severe as that of accident conditions.

6.1.2 Design criteria for tests simulating accident conditions

The hypothetical accident damage conditions given below are those specified in the regulations to be applied to the packages and casks considered in this study. Each package must be designed and constructed so that, if subjected to these conditions in sequence, most of the shielding would be retained and only a limited amount of contaminated coolant or inert radioactive gases could escape.

Free drop. The free-drop test is defined as a fall of the package through a distance of 30 ft (an impact velocity of 30 mph) onto a flat, essentially unyielding, horizontal surface, where it strikes in a position most likely to produce maximum damage.

Puncture. The puncture test is defined as a free drop of the package through a distance of 40 in. where it strikes, in a position to produce maximum damage, the top end of a vertical cylindrical mild steel bar mounted on an essentially unyielding, horizontal surface. The bar shall be 6 in. in diameter, with the top surface horizontal and its edge rounded to a radius of not more than 1/4 in., and of such length (not less than 8 in.) that maximum damage will be sustained by the package. The long axis of the bar shall be perpendicular to the unyielding horizontal surface.

Thermal exposure. The thermal test consists of exposure of the package to a heat input not less than that which would result from exposure of the entire package to a radiation environment of 800°C for 30 min with

an emissivity coefficient of 0.9, assuming that the surfaces of the package have an absorption coefficient of 0.8. The package shall not be cooled artificially until 3 hr after the conclusion of the test period unless it can be shown that the temperature on the inside of the package has begun to fall before this time has elapsed.

Water immersion. The water immersion test consists of submerging all portions of the package under at least 3 ft of water for a period of not less than 8 hr.

Summary. The above tests, although quite severe, are not intended to simulate any specific transportation accident. They do, however, provide a means for producing damage similar to that which might be sustained by the same container if it were involved in a series of severe transportation accidents.

6.2 Licensing[10]

Applicants seeking NRC approval of a packaging design must provide a detailed safety analysis which demonstrates that the design meets the packaging standards and criteria designated in 10 CFR 71.* The demonstration may consist of an engineering analysis, full-scale tests, model tests, extrapolation from tests performed on packages of similar design and construction, or other evidence. The safety analysis, in the form of an application, is the principal document whereby the applicant provides information to demonstrate that the design meets the requirements. The application also serves as the basic document in the NRC staff review of the capability of the design for meeting the requirements of 10 CFR 71. In cases where analytical methods are presented in the application, the methods should be described and justified; and the margin of safety, where applicable, should be stated. Also, the calculational or test accuracy is required in those instances where the design is optimized. Details of the calculations performed should be included in the appropriate

*Code of Federal Regulations, Title 10, Part 71.

section of the application. The complete input data should be provided for those calculations performed by a computer. In all cases, the package model, boundary conditions, assumptions, and material properties used in the calculations should be fully explained and detailed. The adequacy of computer programs should be demonstrated, and modifications to basic programs made in a particular application should be described and justified. The method used to determine data should also be described if the data have been derived experimentally and are not available in the open literature.

6.3 Insurance[11]

The Price-Anderson Act of 1957 relies on nuclear liability insurance provided by two casualty insurance pools and government indemnity to provide assurance of prompt and fair payment of public liability claims resulting from a nuclear incident. In 1975, the Act was extended for 10 years to 1987, and a second layer of financial protection (with funds provided by reactor operators through the payment of a retrospective premium of up to $5 million per reactor) was added to the basic layer of insurance. Provisions were made for increasing the present $560 million limit of liability and for phasing out government indemnity and replacing it totally with private insurance as more reactors are licensed in the future.

Nuclear materials being transported to and from nuclear reactors and other indemnified facilities are covered under the Price-Anderson Act through the insurance purchased by the facility operator and through the indemnity agreement entered into by it with the NRC. Other supplementary nuclear insurance is available to suppliers and transporters of nuclear materials. The 1975 amendments to the Price-Anderson Act extended coverage of the system beyond the territorial limits of the United States for (1) shipments of fuel moving outside the U.S. territorial limits during transit from one NRC-licensed nuclear facility to another, and (2) stationary nuclear reactors licensed by the NRC

beyond the U.S. territorial limits. These changes did not apply to material being exported from or imported to the United States.

In its rule-making procedures in 1976, the NRC exercised its discretionary authority to extend Price-Anderson coverage to certain licensees of plutonium processing and fuel fabrication plants. The NRC has also extended indemnity coverage to irradiated fuel from one reactor that is stored in a second reactor at another site. The NRC chose, in its rule-making proceedings, not to provide separate indemnity coverage for transportation to carriers such as railroads.

The coverage of shipments of nuclear material with respect to insurance policies and indemnity agreements varies, depending on the segment of the fuel cycle in which the shipment is taking place and the type of coverage applicable to the shipper and receiver. In some cases of shipment of radioactive material, coverage is obtained through conventional liability policies rather than specific nuclear policies.

6.4 The Radiological Assistance Program[12]

Some agencies of the federal government have a mechanism that can provide immediate, competent, and experienced assistance to deal with radiological incidents such as those which might occur during the transportation of radioactive material. The Atomic Energy Act of 1954 assigned general responsibility for protecting health and safety of the public to the U. S. Atomic Energy Commission (AEC)*. During the earlier stages of nuclear energy development and use, the AEC and the Department of Defense (DOD) were the federal agencies most deeply involved and, consequently, most knowledgeable in the safety ramifications of radiological accidents. The DOD-AEC Joint Agreement of February 27, 1958, formalized the mutual radiological assistance activities of the AEC. In June 1958, the AEC formally initiated its Radiological Assistance Plan.

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*Now the U.S. Department of Energy.

Under it, the specialized capability of the AEC to deal with radiological incidents was provided as needed and upon request. At about this same time, the U.S. government recognized the expanding non-military use of nuclear energy and the need to coordinate the growing radiological assistance capabilities of many other federal agencies. This resulted in the Interagency Radiological Assistance Plan (IRAP), which assigned overall administrative responsibility for the plan to the AEC (now the DOE) and generally defined the responsibilities of the other signatory agencies.

While the DOE continues to have a statutory responsibility to protect the public health when source, by-product, or special nuclear materials are involved, the aggressiveness of other federal and state agencies in recognizing the need for adequate emergency response capability is adding to the overall radiological emergency capability to such an important extent that coordination is imperative. The increasing use of radiation-emitting materials and equipment not under DOE license or contractual control has required both the U.S. Public Health Service and individual states to effect comprehensive surveillance and control programs that provide a technical capability useful in planning for, assessing, and controlling radiological emergencies.

6.5 Modes of Transport

Truck and rail are the principal modes for transporting waste and fuel materials from the back end of the nuclear fuel cycle. Most radioactive material shipments are presently being carried on trucks; however, these shipments are limited by state governments to a maximum payload weight of about 22 MT. (Shipments exceeding this weight must be made in accordance with overweight permits issued by state agencies.) Payloads of 100 MT or more, on the other hand, can be accommodated in rail shipments. In many cases, improved payload-to-container weight ratios can be achieved via this route because of the increase in size and weight of the shipping containers which can be transported.

Transfers of radioactive material must be made in compliance with federal, state, and local regulations. Federal transportation regulations were discussed in Sect. 6.1. State agencies regulate vehicle sizes and weights and, in some cases, transportation routes and times of travel.

The physical properties of some fuel cycle materials can pose unique constraints on the design of shipping containers and transport modes. Spent fuel, for example, requires containers with large amounts of shielding and some heat removal capacity. The quantities shipped must be limited in order to keep the container weights and dimensions in compliance with regulations.

6.5.1 Transportation of spent fuel

After irradiation to a specified burnup in the reactor, the fuel is transferred to a storage pool where it is held approximately 6 months. Subsequent to this cooling period, it is transported either to a reprocessing plant or to a fuel storage pool to await reprocessing or disposal (if the decision is made not to reprocess the fuel). The irradiated fuel assemblies contain residual fissile material, activated structural materials, transuranic elements, and large amounts of highly radioactive fission products. The decay of the radioactive species produces large amounts of heat; spent fuel shipping casks must therefore provide relatively large amounts of radiation shielding and have a large heat removal capacity.

Spent fuel shipping casks have not been designed for the reactor cases of this study; however, the fuels utilized in the study are sufficiently similar to current LWR fuel that it may be assumed that present-generation casks could be used to ship it after discharge from the reactor. A description of a currently licensed spent fuel cask, the General Electric IF 300, is given below:

> The IF-300 is a water-cooled cylindrical cask designed to carry 7 PWR or 18 BWR assemblies. Shielding is provided by 10.2 cm of depleted uranium metal in the cask shell and an exterior water-filled neutron shield. Heat is removed from the fuel to the cavity walls by

> natural circulation of water in the cavity and transmitted through the cask sides and ends by conduction. The outer steel shell is surrounded by an annulus of water retained by a corrugated jacket forming the neutron shield. The corrugated jacket is cooled by a forced air cooling system. The loaded cask weighs about 61 MT. The cooling equipment weighs an additional 13.6 MT. The cask is normally transported by rail on a 100-MT capacity flatcar. It may be transported for short distances on highways on a specially designed truck. Such truck shipments require special overweight permits. The method is used primarily to move the cask from reactors without rail facilities to the nearest rail siding.

Spent fuel from the heavy water reactors is assumed to be shipped in casks similar to those used for shipping CANDU fuel. These casks[13] are described as follows:

> The cask is designed in the shape of a rectangular parallelepiped weighing approximately 63,000 kg. It is capable of carrying about 200 spent fuel bundles in water to provide efficient heat transfer to the walls of the cask cavity. The cask is finned for heat transfer purposes and has a heat removal capacity of about 70 kW.

6.5.2 Transportation of fresh or recycled fuel

Fresh LWR fuel that contains only ^{235}U, ^{238}U, or virgin thorium does not require shielding. Consequently, packages similar to those used to ship fresh PWR fuel assemblies can be used to estimate the number of shipments required annually to operate the reactors considered in this study. In the case of HWR fuel, it is assumed that the methods used to ship unirradiated CANDU fuel can be used to predict the number of fresh fuel shipments to that type of reactor. However, if the recycled fuel contains ^{233}U, recycled thorium, actinides, a spikant, or any fission products, shielding of the assemblies will be required for shipment.

Unshielded fuel packages. These steel packages are built in the shape of a cylinder about 1.25 m in diameter and 5 m long. Each weighs about 700 kg empty and 1800 kg with two LWR assemblies.

CANDU fuel, which is about 10 cm in diameter and 0.5 m long, is packed in Styrofoam blocks and bonded onto wooden pallets; each package

contains 100 unirradiated assemblies. Two such pallets can be shipped to the reactor in one truck.

Shielded fuel packages. It is assumed that recycle fuel which is supplied to a reactor and requires shielding will be transported in a cask having about one-half the shielding required for the spent fuel and weighing about one-half that of the spent fuel cask. However, for colocated facilities, the carrier could transport refabricated fuel in the same cask used for spent fuel.

6.6 Transportation Requirements

Estimates for the number of annual shipments required for each of the fuel cycles in this study were made using the above assumptions. Table 6.1 gives these estimates on a per-reactor basis.

7. SPENT FUEL STORAGE FACILITIES

Light water reactors typically have spent fuel storage facilities sized to contain a full core plus the spent fuel removed from the reactor during 1 year of operation. Most BWR fuel management plans are based on replacing the core approximately every 4 years (i.e., one-fourth of the core discharged as spent fuel per year); PWR plans are based on replacing one-third of the core per year. The average spent fuel storage space at reactor sites will accommodate at least 1.33 PWR cores and 1.4 BWR cores for single reactor sites, assuming no expansion of at-reactor (AR) storage capacity. Pools with substantially larger capacities will probably be constructed in the future.

The design bases for sizing of spent fuel storage basins at currently operating plants allow space for at least one annual discharge, which is allowed to decay for up to 1 year prior to shipment to a reprocessing plant. The reprocessing plant stores spent fuel on-site for a few months, then separates the uranium, plutonium, and waste. The recovered uranium and plutonium are further processed and fabricated into fuel assemblies that are returned to the reactor. With this cycle repetition

Table 6.1. Transportation requirements for the eight fuel cycles[a]

Case	Reactor	Fuel cycle	Annual reload (kg HM)	Amount of HM per Assembly (kg)	No. of assemblies reloaded per year	Shielding required for fresh or recycle fuel?	Package capacity		No. of shipments per year	
							Fresh fuel	Spent fuel	Fresh fuel	Spent fuel
1	SSCR	Once-through uranium	31,062	388.3	80	No	2	7	40	12
2	SSCR	Denatured ^{233}U-thorium	31,062	388.3	80	Yes	7	7	12	12
3	HWR	Denatured ^{233}U-thorium	73,530	17.0	4325	Yes	200	200	22	22
4	HWR	Once-through uranium	55,850	18.7	2990	No	200	200	15	15
5	LWR	Improved once-through	20,555	426.4	48	No	8	7	6	7
6	LWR	Denatured ^{235}U-thorium	28,717	389.8	74	No	2	7	37	11
7	LWR	Denatured ^{233}U-thorium	31,186	389.8	80	Yes	7	7	12	12
8	LWR/HWR	Tandem								
	LWR	Standard[b]	63,807	426.4	149	No	8	7	19	22
	HWR	Standard	63,807	20.5	3110	Yes	200	200	16	16

[a]All flows based on equilibrium cycle.

[b]Assumes that 1.86 standard LWRs are required to feed 1 HWR.

every year, existing reactors should have adequate spent fuel storage facilities. Under conditions of planned reactor operation and with reprocessing of spent fuel, the reactor's spent fuel basin would be only partially loaded most of the time.

Additional capacity will be required to store spent fuel in the once-through fuel cycle if plans are carried out for terminal encapsulation and storage facilities based on nominal LWR fuel (33,000-MWd/MT exposure) with a decay of about 5 years. Longer decay times might permit added savings in fuel encapsulation and geologic space requirements. Several years of additional storage capacity will be required if full-core-reserve (FCR) status in basins is retained.

Further considerations include the proposed changes to optimize once-through fuel cycles in the LWR and SSCR and increase the fuel burnup to ~50,000 MWd/MT. The curves shown in Fig. 7.1 illustrate that an additional 5 years is needed for the fuel to decay to a radioactivity level equivalent to the 33,000-MWd/MT exposure (dashed line on figure). The net effect is to increase the interim fuel storage requirement by 20%. For HWR CANDU fuel (0.7% ^{235}U), the entire core is discharged annually at 8000-MWd/MT exposure. With the enrichment increased to 1.2%, the quantity of fuel discharged annually is approximately halved but still produces a larger storage requirement (for 3-1/2 year decay) than for 33,000-MWd/MT LWR fuel after a 5-year decay period.

The reactor cases covered in this evaluation have no major impact on the spent fuel storage methods presently used in the normal LWR fuel cycles. Since the basic fuel geometries are not affected by the change in fuel composition, the facilities, fuel handling, and storage arrangements normally associated with these standard reactors or recycle plants are satisfactory. The spent fuel disposition, however, does change with the various cases and will have an impact on the quantity and locations of spent fuel storage. Once-through fuel cycles require a significant increase in spent fuel storage capacity. This can be provided by increasing the storage density of the reactor pools or constructing away-from-reactor (AFR) storage basins. Modification of the fuel cycle to increase core life, thereby increasing burnup, will also affect the storage

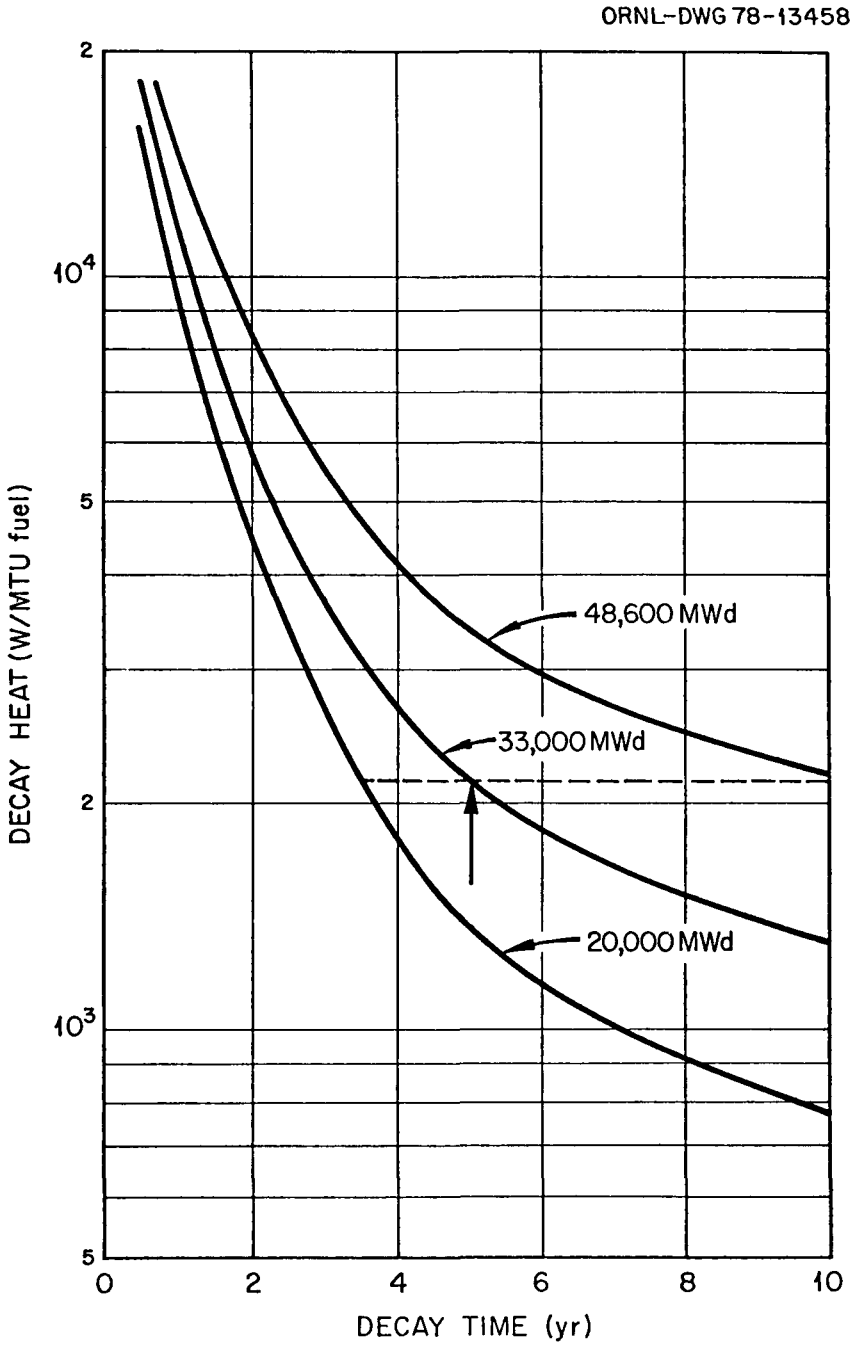

Fig. 7.1. Power decay of LWR-UO_2 spent fuel.

by requiring longer residence for increased decay time; modification of water treatment systems to handle the larger heat loads may also be necessary.

The amount of spent fuel storage needed through the year 2000, based on an earlier estimate of 383 GW(e) total installed capacity by that time, is shown in Table 7.1. Current estimates of ~345 GW(e) installed capacity are lower and will require about 10% less storage.

The additional storage required by the alternate fuel cycles may make it necessary to consider the use of compact storage at the reactor plants or the use of a generic AFR storage facility. Figure 7.2 shows the AFR storage requirements based on various storage options relative to pool expansion or the use of FCR.

Qualitative comparisons of alternatives for spent fuel storage are given in Table 7.2. These include descriptions of water- and air-cooled fuel storage facilities.

7.1 Storage in Water Basins

Spent fuel storage in water basins (Fig. 7.3) is developed technology. The use of water offers the following benefits:

1. It serves as an excellent heat transfer medium for removing decay heat from the fuel and provides a substantial heat sink under most accident conditions.
2. It is a transparent radiation shield that allows visual inspection and direct manipulation of the fuel.
3. It provides partial containment of fission product gases and essentially full containment of any particulate radioactive material that may escape from a fuel assembly.

Water basins are designed to retain their water-tight integrity for all credible accidents, including the design-basis tornadoes and earthquakes. They are designated as Category 1 seismic structures and, as such, are designed (1) to resist rupture and excessive loss of water; and (2) to prevent all massive equipment, such as cranes, from falling into the basins.

Table 7.1. Schedule of nuclear generating capacity and spent fuel discharges — annual and cumulative

Year	Capacity of reactors discharging spent fuel [GW(e)]	Spent fuel discharged annually[a] (MTHM)	Cumulative spent fuel discharged (MTHM)
1976	38	820	2,500
1977	43	940	3,400
1978	53	1110	4,500
1979	60	1260	5,800
1980	68	1410	7,200
1981	74	1650	8,900
1982	96	1950	11,000
1983	101	2110	13,000
1984	105	2260	15,000
1985	105	2340	18,000
1986	105	2340	20,000
1987	105	2250	22,000
1988	150	3210	25,000
1989	167	3560	29,000
1990	186	3960	33,000
1991	205	4360	37,000
1992	223	4740	42,000
1993	244	5210	47,000
1994	265	5660	53,000
1995	285	6090	59,000
1996	307	6520	65,000
1997	328	6940	72,000
1998	347	7310	80,000
1999	366	7630	87,000
2000	383	7920	95,000

[a]Data through 1985 are from individualized reactor characteristics at 0.6 plant factor; for years 1986-2000, data are based on 0.7 plant factor.

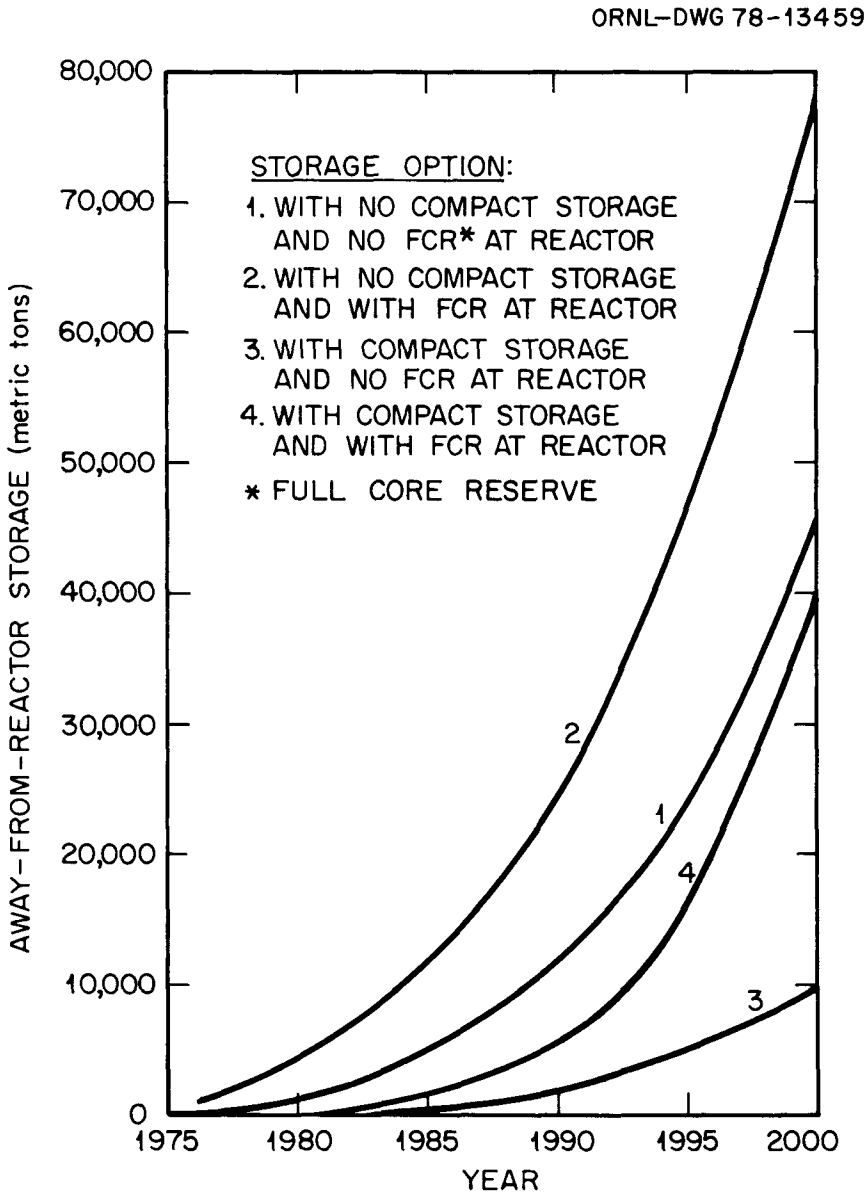

Fig. 7.2. Away-from-reactor storage requirements.

Table 7.2. Summary comparison of spent fuel storage alternatives

Storage alternative	Confinement barrier in addition to cladding	Means of heat removal	Method of controlling fuel cladding corrosion	Maintenance requirement	Surface reading
Unpackaged storage					
Water-cooled basin[a,b]	Water[b,c]	Forced circulation of basin water	Low temperature and water quality control	High	Low
Air-cooled vault[a]	Filters	Forced circulation of air	Low temperature	Moderate	Moderate
Packaged storage					
Water basin[a]	Water and package[a,b]	Forced circulation of basin water	Packaged in inert or noncorrosive medium	High	Low
Air-cooled vault[a,b]	Package	Natural circulation of air	Packaged in inert or noncorrosive medium	Low	Moderate
Concrete surface silo (surface storage cask)[a,b]	Package	Natural circulation of or conduction to air	Packaged in inert or noncorrosive medium	Low	High
Geologic formations[a,b]	Package, hole liner[a]	Conduction to earth	Packaged in inert or noncorrosive medium	Moderate	Low
Near-surface caisson[a,b]	Package, hole liner	Conduction to earth	Packaged in inert or noncorrosive medium	Low	High

[a]Alternative studied for LWR fuel.

[b]Alternative studied for CANDU fuel.

[c]Filtration of effluent ventilation air may be used to provide an additional containment barrier.

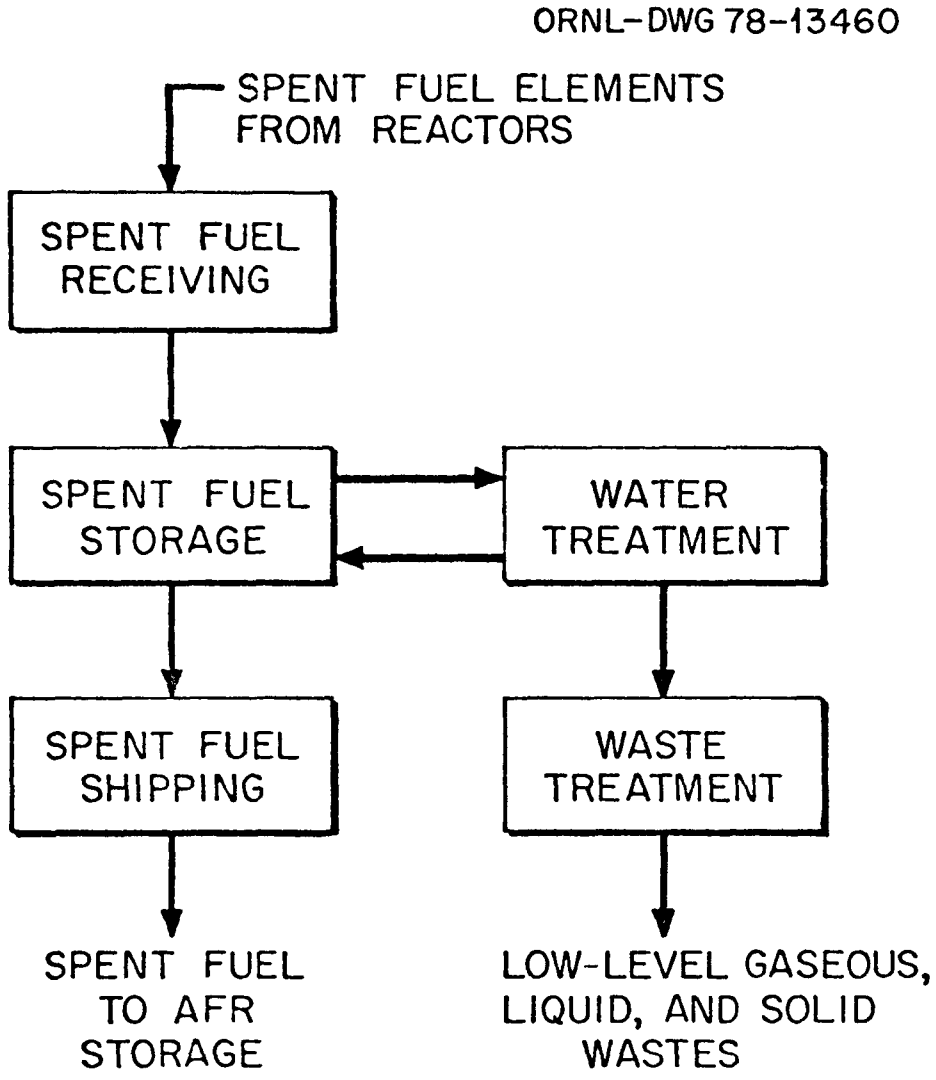

Fig. 7.3. Simplified schematic of the storage basin process.

The basin facility contains the following process areas: cask-carrier handling, cask processing, underwater handling and storage, and support. These generic areas, which are designed to receive and store LWR and CANDU fuels, are described briefly in the following paragraphs.

7.1.1 Cask-carrier handling

Spent LWR and CANDU fuel is received in the cask-carrier handling area in licensed casks from commercial carriers using either rail or truck shipments. Each is identified, inspected for shipping damage, and monitored for radiation.

7.1.2 Cask receiving

After the casks have been removed from the carrier, they are prepared for insertion into the fuel unloading pool. This preparation includes washing the casks to remove road dirt, venting the casks to the off-gas system, and cooling and flushing the casks as required. The waste from cask rinsing, flushing, and cooling is collected for treatment.

7.1.3 Underwater handling and storage

Cleaned and cooled casks are inserted into the unloading pool in a vertical position. The depth of the water in these pools is sufficient to allow vertical unloading of the spent fuel from the casks while shielding the spent fuel. After the cask has been lowered into the pool, the cask lid is removed, and the individual fuel assemblies are withdrawn and stored underwater. When the casks contain LWR fuel, the individual fuel assemblies are transferred to multiple-assembly storage racks in the pool. In the case of CANDU fuel, the baskets used in cask transport also serve as storage baskets. CANDU baskets are transferred directly from the shipping casks to the appropriate storage basin. All spent fuel is handled by remote control under approximately 3.6 m of water to shield the operating personnel. Fuel assembly leakers are placed in a special container to control release of radionuclides during handling and storage.

After all fuel assemblies have been removed, the empty cask is inspected (to ensure that it contains no fuel or nonfuel items) and the cask lid replaced. As the cask is removed from the pool, sprays of high-velocity demineralized water flush away pool water and contamination from the exterior of the cask. When the cask reaches the parapet level, the head bolts are replaced in the lid, and the cask is transferred to the decontamination area.

The multiple-assembly storage racks for LWR fuel are fastened to the basin floor. The racks are designed to maintain the spacing of the baskets even during extreme natural phenomena disturbances.

Basins that store CANDU fuel contain racks to support stacked arrays of CANDU baskets. Four baskets of CANDU fuel (~1 m high by 1 m in diameter) are stored in the racks in a vertical position. These racks are designed to limit motion of the baskets and possible fuel damage during extreme natural disturbances. There is no requirement to maintain the spacing of CANDU fuel, which is subcritical in light water.

7.1.4 Support operations

The basins have support facilities which dissipate the heat, control the quality of water in the basins, ventilate the building, treat the radioactive waste generated, and provide services such as electricity and water.

7.2 Generic AFR Storage Basin

The major processing facilities that are located within a secured area include a cask unloading and fuel handling building, an emergency cooling water pond, and the fuel storage basin. Environmental release points are the 45-m (150-ft)-high stack for discharge of airborne effluents, the cooling tower for heat dissipation, the facility air conditioning system, and the radwaste treatment area for solidification of nonsolid wastes for off-site burial.

The size and number of process areas for handling and storing spent fuel are determined by the expected delivery rate for the spent fuel and

by the amount of spent fuel that might require storage in the basin. A typical AFR storage facility would probably be designed for receiving and shipping 2500 MT of spent fuel per year. The calculated number of process areas required for the fuel cycles of this study is shown in Table 7.3. An on-stream time of 300 days per year was assumed.

7.3 Dry Storage Facilities

Several concepts for air-cooled spent fuel storage facilities have been assessed.[14] One of the preferred arrangements utilizes the convection vault[15] concept, where air is drawn in from below the fuel, is heated by the decay energy, and exits at the top. No external powered blowers are required. Cooling by convection occurs over the outside of the storage wells.

This concept has limited applicability to recently discharged fuel that has a high heat generation rate; however, terminal storage of 5-year or older fuel could probably be accommodated.

7.4 Fuel Encapsulation and Geologic Storage

Terminal disposal of nuclear waste and spent fuel in the United States is assumed to require a geologic repository. Since no repository has yet been built, this description is based on the current concepts of waste isolation. The repository described is capable of encapsulating and storing from 60,000 to 90,000 MT of spent fuel, which is the quantity expected to be discharged by 50 GW(e) of nuclear power over about 50 years of reactor operation. Such a repository will require 10 to 15 years of development and construction before being operable. Longer delays will require additional AFR storage facilities.

In view of the extensive studies on salt formations as the initial geologic host and the several design alternatives developed, the generic repository was assumed to be located in salt. The reference repository is the short-room-and-pillar concept intended to ensure ceiling support

Table 7.3. Minimum number of cask process handling areas at a fuel storage facility for a 2500-MT/year throughput facility

Assumptions on cask handling	
Casks/day, rail	1.8
Casks/day, truck	6.7
Number of handling areas	
Preparation area and cask unload/load area	4
Cask cool and washdown area	2
Fuel unloading pools	2
Cask decontamination area	2

and mine stability. Waste or spent fuel is buried in the floor to simplify retrievability if this is required. The typical repository is designed to receive, encapsulate, and place spent LWR and CANDU fuel elements, high-level waste, and transuranic (TRU) waste in mined locations 450 to 600 m (1500 to 2000 ft) below the surface.

The location of the repository is selected on the basis of the long-term geologic stability and a favorable geologic and hydrologic evaluation of the selected site. The prime concern in selecting a repository location is the absence of intruding forces that might interact with isolated waste. Hence, an area characterized by seismic stability and very low or no water movement is required. The absence of natural resources such as gas, oil, or valuable mineral deposits will ensure a minimum commitment of natural resources to the waste management program. In addition, the absence of valuable resources will prevent future generations from drilling or mining extensively near the repository. The sequential steps in the development and operation of the repository are as follows:

1. Extensive exploration and testing of the region and of the specific repository area for geologic and hydrologic stability and for acceptability from both technical and societal consideration.
2. Development of a detailed facility design and the accompanying safety analyses.
3. Licensing of the facility by the NRC.
4. Construction of the receiving and mining facilities, as well as the deep tunnels and waste isolation rooms for reception of the waste.
5. Receipt, examination, and encapsulation of the spent fuel, prior to its placement in the designated areas in the repository.
6. Proof testing the impact of the radioactivity, heat, and mine operation on the repository with maintenance of a retrievability option for all the initial waste placed in the repository. After verification of the repository's performance, additional repository rooms will be mined and additional waste will be

placed in the repository.

7. Backfilling of filled waste burial rooms and tunnels with material mined as additional areas are prepared for waste placement.
8. Removal of all surface facilities and sealing of all below-ground areas (with material previously mined from the area) when the repository is filled.
9. Sealing all shafts and openings to the repository.
10. Long-term monitoring in the vicinity of the repository to ensure that isolation is being maintained.
11. Marking surface areas and keeping technical records to indicate the content and location of the repository; limiting surface usage above the repository to prevent drilling or mining in the immediate area.

Current proposals suggest that the repository be operated in a trial phase for 5 years while long-term acceptability of the site is being established. During this period, all wastes will be emplaced in a retrievable manner by lining the storage holes with steel sleeves to prevent contact between the waste containers and the salt and by storing less than the maximum allowable amount of fuel per acre of room. Once this trial phase has been completed and the site acceptability verified, the storage density of the spent fuel will be increased.

Major surface structures included within a secured area at the generic repository are the canistered waste building, low-level waste building, mine operation building, and mine salt storage building. The major functions of these buildings are:

1. The canistered waste building handles all waste that requires shielding. This building contains facilities for receiving, unloading, encapsulating, and preparing waste for disposal.

Spent fuel is encapsulated in this building in steel containers to provide additional containment of the fuel as well as improved control of contaminants.

2. The low-level waste building provides unshielded facilities for receiving, unpackaging, and handling of all low-activity wastes.
3. The mine operations building provides support for the nonradiation-related mining operations.
4. The mined salt storage building provides interior surface storage of 1.2×10^6 m^3 of mined salt.

Other surface structures include an administration building, warehouses, and a gate house. In addition to the secured facilities, excess salt tailings mined from the underground repository will be stored just outside the fenced area. This salt storage facility will occupy about 10 hectares (200 acres) when filled with approximately 24 million MT of salt. The salt will be 18 m deep and covered with 3 m of earth.

Surface structures which handle materials that could potentially release radionuclides are provided with installed ventilation to protect the environment and the personnel in the facilities. The exhaust air from the canistered waste building and low-level waste building is filtered through HEPA systems to reduce particulate radionuclide releases. Because of the low release potential, no equipment is provided to recover any gaseous (nonparticulate) fission products that might be released.

Subsurface structures include shafts, hoists, and mined disposal areas. Four types of shafts are required to accommodate the following: canistered waste, low-level waste, men and materials, and ventilation exhaust. The canistered waste shaft is used to lower encapsulated spent fuel and all other wastes, except low-level TRU waste, to the appropriate mine level for disposal. The low-level shaft is provided for low-level TRU waste. The men and materials shafts provide a means of transporting personnel and equipment to the mine, supplying ventilation air to the personnel, and moving the mined salt from the mine to the surface. The ventilation exhaust shaft houses fans and filters for discharging ventilation air through the 110-m (360-ft) mine exhaust stack.

The spent fuel disposal areas have a room-and-pillar design with vertical holes in the floor of each room. The encapsulated spent fuel and other wastes are transported to and emplaced in these holes. The TRU waste containers are stacked on pallets in an open matrix in low-level waste rooms.

Spent fuel and other wastes are received at the surface facilities of the repository. They are transported in licensed shipping casks and segregated according to waste type. Following unpackaging, inspection, and encapsulation or containerization when needed, the spent fuel and the waste are then lowered into the repository. The spent fuel is placed in the vertical holes within the appropriate chamber; the low-level waste is stacked on pallets within each low-level waste room.

The exhaust air from the subsurface working areas of the repository will be passed through HEPA filters before being released to the environment through the 110-m (360-ft) stack. Releases of radioactive materials which may occur from the working areas of the repository are discussed in the subsequent sections of this analysis. No release of radioactive materials from inactive and backfilled sections of this subsurface mine is assumed.

8. ENVIRONMENTAL ASSESSMENT AND LICENSABILITY

All of the fuel cycles of this study require examination to assess environmental effect and licensability. The facilities and procedures of each recycle operation must be designed to minimize environmental insults and meet licensing requirements. The once-through fuel cycles may have fewer environmental concerns than the cycles that employ reprocessing and refabrication since radioactive waste materials are not purposely released. Nevertheless, all fuel cycles must comply with strict standards to ensure that containment barriers are not breached.

8.1 Environmental Assessment

In general, the potential environmental effects associated with the denatured ^{233}U-thoria fuel cycle are similar to those of the ^{235}U-thoria fuel cycle. Current regulations governing fuel reprocessing and fabrication require that the site boundary dose due to radioactive releases from the facility be effectively less than the dose due to background radiation. Basically, then, there will be no difference in the environmental effects of different fuel cycles; however, the engineering requirements to confine radioactivity in order to meet regulations will be slightly different. The recycle and storage facility designs will follow the concept of total isolation from groundwater by appropriate building design to withstand the effects of natural phenomena expected for the specific site location.

All liquid and solid wastes potentially contaminated with radioactivity will be treated to reduce the volume and fixed into a nonleachable form for ultimate repository storage. All gaseous effluents will be treated for removal of radioactive isotopes such as ^{3}H, Rn, ^{85}Kr, and ^{129}I. Treatment beds will be fixed in permanent media and disposed of as solid waste in a repository. Gaseous krypton will be bottled and stored until decayed or may be fixed by techniques currently under development. Radon will be retained for a suitable decay period, and the long-lived daughter products will be treated as solid waste.

Uranium-233 associated with the thorium fuel cycle will require special handling of fabrication plant effluents as well. The environmental effects for plutonium conversion will be similar to those for the reference uranium-plutonium cycle. Chemical effluents will follow the same route through liquid and solid waste fixation as the radioactive effluents; therefore, the environmental effects may be less severe than for many types of standard industrial plants that employ less restrictive standards.

Because the thorium fuel cycle produces ^{233}U and because ^{233}U is a better fuel than ^{235}U, the requirements for mining, milling, and enrichment will eventually be decreased, thus reducing the volume of chemical

and radioactive effluents to be treated from the front end of the fuel cycle.

8.2 Licensability

Facilities for the safe storage, reprocessing, and fabrication of spent denatured uranium-thorium fuels must meet the same regulatory requirements for protection of the general public from radioactive exposure as other nuclear fuel cycles. The safety considerations for denatured ^{233}U-thoria would not be significantly different from those for uranium-plutonium facilities.

All structures, systems, and components whose failure might result in exceeding specified doses at the plant boundary must be identified as "important to safety" and analyzed for possible failure modes and effects. All such structures, systems, and components will be designed, fabricated, installed, and tested to specified quality standards. These quality-assured structures, systems, and components constitute the primary containment for radioactive materials. Secondary containment includes limited or no-access areas and controlled ventilation areas. Access will be restricted by a combination of physical barriers and security measures. All structures, systems, and components important to safety will be designed to withstand the effects of natural phenomena likely to occur at the specific facility site. Redundant and/or diverse emergency backup systems will maintain uninterrupted operation of all systems related to containment of radioactive materials. Special nuclear materials will be geometrically or administratively controlled to prevent criticality. Potentially combustible or explosive materials will be handled in a special way to prevent incidents that might result in radioactive release. Potentially volatile or leachable waste materials will receive special treatment and fixation.

In addition to regulatory requirements for protection of the general public, operating personnel must be protected for efficient, economical plant operation.

The denatured uranium-thorium cycle will require the handling of ^{233}U in shielded areas. If plutonium is left with the waste, it will be contained by the shielding used to protect workers from radioactivity. If plutonium is recovered for refabrication, it will be handled in a controlled-environment glove-box area. Thorium will require light shielding for fresh fuel fabrication.

All of the above protective measures except thorium handling have been demonstrated for existing LWR fuel cycles in licensed facilities that met regulatory requirements at the time of construction. A licensed facility for manufacturing thorium fuels exists at the General Atomic Company in San Diego, California. Future regulatory requirements may be more restrictive and introduce additional design constraints.

9. TECHNOLOGY STATUS AND R&D REQUIREMENTS

The eight fuel cycles under study can be divided into two groups for this evaluation: (1) the once-through cycle for the LWR, HWR, SSCR, and tandem PWR/HWR, and (2) the denatured uranium-thorium cycle for the LWR (^{235}U and ^{233}U), SSCR, and HWR. Since much of the required R&D for the LWRs applies to the other cycles within each group, these common areas are discussed first; then any additional work required for specific fuel cycles is pointed out.

A short review of the status of existing LWR recycle technology will provide background information for a discussion of the R&D required for the eight cases mentioned above. Although plutonium recycle as mixed oxide (MOX) is not being considered in this study, the status of plutonium recycle is mentioned because some of the technology for the plutonium recycle can also be applied to the denatured uranium-thorium cycles.

9.1 Status of Existing LWR Recycle Technology

The basic technology for recycling LWR fuels containing uranium and uranium-plutonium already exists. It is based on the Purex reprocessing

operation that has been used for many years in government-owned plants in the United States and other countries, and on the fuel fabrication/refabrication experience of various countries and private organizations. However, neither a commercial reprocessing plant nor a refabrication plant that conforms to current federal and state environmental, safety, and safeguards requirements has been operated in the United States. In fact, these requirements have not yet been fully defined. Fuel recycle licensing has been suspended, as has issuance of the Generic Environmental Statement on Mixed Oxides (GESMO) except for that part relating to safeguards. The technology for recycle has been developed through engineering-scale or pilot-plant-scale work for all important components. The operability, reliability, licensability, and costs of an integrated plant must be demonstrated. It is certain that some additional development and demonstration work would be required for any integrated fuel recycle plant. It is also certain that an integrated plant can be built and operated within anticipated licensing requirements.

9.1.1 Fuel refabrication

The refabrication of recycled fuel has not been commercialized in the United states. The development of the commercial processes is dependent, in part, upon a favorable decision by the NRC and the President to permit plutonium to be recycled in power reactors. However, fuels made from mixtures of uranium and plutonium oxides (MOX) have already been prepared in small-scale pilot plants operated both in the United States and abroad, using high-grade plutonium produced for the weapons program. Analysis of power reactor-grade plutonium shows an order-of-magnitude increase in short-lived plutonium isotopes which, because of spontaneous fission and increased gamma activity, are more hazardous to process. The usual procedure in MOX fabrication plants is to prepare UO_2 powder (from natural uranium, recycled uranium, or slightly enriched uranium), to mix the UO_2 with PuO_2 powder prepared from plutonium nitrate solution, and then to proceed through the same fabrication operations as for fresh fuel. Thus the principal difference between fabrication and

and refabrication is that refabrication of plutonium is complicated because of its radioactivity and requires special considerations. Plutonium must be contained with a high degree of integrity, and shielding must be provided to protect workers from radiation. Kilogram quantities of weapons-grade plutonium have been processed in alpha-contained facilities with only light gamma shadow shielding. The increased activity of the plutonium produced in power reactors will require remote or semiremote handling techniques, which have not been fully developed. Refabrication experience with plutonium of the isotopic composition expected in recycle fuels is limited. The full extent of the problems associated with MOX fuel refabrication will not be known until more experience with remotely operated engineering-scale equipment has been gained. Further, to facilitate fuel dissolution in nitric acid without fluoride catalyst, the MOX fuels may need to be made by coprecipitation methods to ensure that the PuO_2 is in solid solution in the UO_2.

The qualification of fresh fuel for reactor use has, of course, been established. The qualification of recycle fuel has not been completed; present development has largely been based on fuel fabricated in laboratory- or engineering-scale facilities. Additional work is required to demonstrate that fuel refabricated in commercial facilities performs satisfactorily. Additional development work to clean and recycle most process radioactive waste streams appears to be necessary to prevent the release of plutonium from fuel refabrication plants.

The major refabrication plants presently in operation in the United States are shown in Table 9.1.

9.1.2 Fuel reprocessing

LWR fuel reprocessing operations include shearing the fuel, voloxidation, fuel dissolution, feed clarification and adjustment, off-gas cleanup, recovery of the fissile material largely free of fission products, and conversion of nitrate products to the oxide form for refabrication. LWR fuel reprocessing has undergone significant commercial development as evidenced by the operation of the Nuclear Fuel Services, Inc. (NFS) plant in West Valley, New York, during the period 1966-1971. Other plants

Table 9.1. Major U.S. LWR-uranium-plutonium fuel refabrication facilities

Organization	Location	Recycle capacity MT/year	Recycle capacity MT/day[a]
	Existing		
B & W — NUMEC	Apollo, Pa.	20-25	0.1
Exxon	Richland, Wash.	~20	0.1
Westinghouse	Cheswick, Pa.	20-25	0.1
	Planned		
Westinghouse	Anderson, S.C.	200	1.0
B & W — NUMEC	Apollo, Pa.	72	0.4

[a]Approximate capacity, assuming 200-days/year operation.

intended for commercial operation have been built at Morris Illinois, by the General Electric Company, and at Barnwell, South Carolina, by Allied-General Nuclear Services. The capacity and status of each of these plants are given in Table 9.2.

None of the plants listed in Table 9.2 are in operation today. Only the AGNS plant is being considered for use, and plans for its start-up are indefinite. The plant is presently in an advanced-construction and equipment-checkout state; however, fuel conversion and waste treatment facilities are still needed, and other modifications may be required for NRC approval. Extensive modifications of the MFRP and NFS plants would be required before they could become operable. While the NFS plant provided commercial reprocessing experience which met safety requirements for its period of operation, this experience might not be directly applicable to the more highly irradiated fuels (up to 40,000 MWd/MT) anticipated in the future, or to the shorter-cooled fuels. Further, neither the effluent control operations of the plant nor its design for earthquake and tornado resistance would conform with current regulations.

The extensive work carried out at various government installations and the commercial experience with the NFS plant, although limited, have provided a good understanding of reprocessing operations. With regard to the initial mechanical processing, cropping of the nonfueled metal from the fuel assembly is reasonably successful; on the other hand, shearing of the fuel still needs development, particularly with regard to practical remote maintenance of the shear. Work needs to be continued on bundle disassembly prior to pin shearing. If voloxidation to remove tritium is required, this would necessitate extensive development. The technology for the dissolution and separation of uranium and plutonium is well developed for fuels at modest burnups (~30,000 MWd/MT); however, a highly radioactive residue in amounts that increase with increasing burnup remains when more highly irradiated fuels are dissolved, and practical separation of the residue from the solution may be difficult. Plutonium loss to this residue, while thought to be low, must be determined in plant practice. Further, the dissolution of MOX fuels has only been demonstrated in laboratory-scale operations.

Table 9.2. Commercial reprocessing plants for low-enriched uranium fuels

Plant	Site	Type of fuel	Uranium capacity (MT/year)	Oxide fuel processed (MT)	Status of operation
Allied-General Nuclear Services (AGNS)	Barnwell, S.C.	Oxide	1500	-	Has not started; startup dependent on NRC decision on plutonium recycle and additional facilities.
Nuclear Fuel Services, Inc. (NFS)	West Valley, N.Y.	Oxide	750	222	Shut down; startup dependent on relicensing by NRC and certain economic factors.
Midwest Fuel Reprocessing Plant (MFRP)	Morris, Ill.	Oxide	300	-	Has not gone on-stream; startup dependent on major revisions to plant and various economic factors.

Off-gas treatment. Off-gas cleanup technology has been demonstrated, but new requirements may be imposed for tritium and ^{85}Kr removal; new processes for iodine removal from the off-gas and from liquids to be released after evaporation are being developed. Considerable additional work relative to off-gas treatment needs to be done because disposal restrictions can be expected to be more stringent in the future, with emphasis on the containment of ^{85}Kr, ^{14}C (as CO_2), ^{129}I, and tritium. In the case of thorium fuels, ^{220}Rn must also be contained. Research and development, which has been carried out on various separation processes for the above nuclides, includes cold (tracer) pilot-plant-scale operation of a fluorocarbon absorption process for ^{85}Kr removal, bench-scale studies of iodine absorption in nitric acid, and vaporization and absorption of tritium on activated carbon and on molecular sieves. Removal of ^{14}C could be accomplished by fixation of CO_2 as $CaCO_3$. Radon can be held on molecular sieves until it decays to nonvolatile daughters.

Liquid waste treatment. In a fuel recycle plant waste treatment facility, acidic aqueous wastes are concentrated so that acid and water may be recycled. The concentrated fission products are stored but must eventually be calcined and incorporated into vitrified materials for ultimate disposal. (Disposal per se is not considered in this study.) Cladding hulls, other alpha-containing wastes, and numerous low-level wastes from reprocessing and refabrication operations must also be treated for disposal. The radioactive off-gases mentioned above and perhaps the traces of other noble gases need to be collected and treated for storage. Waste-treatment operations would be largely associated with the effluents of the reprocessing operations of the recycle plant, but refabrication wastes cannot be neglected.

Experience with treatment of high-level wastes generated by the reprocessing of power reactor fuels is limited. Idaho National Engineering Laboratory (INEL) has made engineering/pilot-plant and larger-scale studies on calcination of high-level wastes, while Battelle-Pacific Northwest Laboratory (PNL) has carried out extensive R&D activities on calcination and vitrification. Operating experience on the storage of intermediate-level wastes has been confined to the interim storage of liquids in tanks. The technology for handling wastes

from operations involving the removal of Zircaloy cladding is also limited. Extensive experience exists on the treatment of low-level wastes, but its adequacy has recently been questioned. Also, modification of the fuel dissolution procedure to accommodate MOX or thorium-base fuels may add certain chemicals, including fluoride, which complicate waste treatment. Long-term containment of radioactive noble gases would presumably involve storage in gas cylinders or incorporation into solids.

9.2 Required R&D for Once-Through Cycles

The various R&D requirements for the once-through fuel cycles of LWR, SSCR, and HWR systems are discussed in the following subsections.

9.2.1 Improved LWRs

The improved fuel cycle utilizes fuel of slightly higher initial enrichment than is typical of present-day PWRs (4 wt % vs 3 wt %) and longer irradiation times, which results in smaller quantities of fuel being removed from the reactor on an annual basis. The higher fuel exposure results in a higher decay heat rate, but this should require only small modifications of the storage facility. Since no reprocessing is involved in the once-through cycles, the main R&D requirements are for fuel fabrication procedures and qualification for the higher-burnup fuel and for waste treatment.

Each of the five LWR fuel suppliers in the United States has proposed (or has considered proposing), with utility participation, high-burnup demonstrations in an operating reactor. The first of these programs has just begun. As the initial phase of this program, a demonstration is planned in which five irradiated assemblies will be examined and, if found satisfactory, will be irradiated an extra cycle and then reexamined. The objective is to increase the burnup from 33,000 MWd/MT to 38,000 MWd/MT, but four of the assemblies are now expected to attain 40,900 MWd/MT. In the second part of this program, fuel capable of achieving a burnup of 45,000 MWd/MT will be designed and tested.

Effort is required to develop fuel capable of achieving higher discharge burnups and to demonstrate that it will give satisfactory irradiation performance. The initial phase of this development program should consist of few-assembly irradiations in which the peak discharge exposures anticipated in the improved design are attained. The purpose of these irradiations is to provide an early indication of the performance of fuel irradiated to high exposures, and to generate data on parameters required for the design of high-burnup fuel (e.g., fission gas release and fuel/cladding mechanical interaction). This part of the development program can most quickly be accomplished by irradiating several assemblies for one or more extra cycles beyond their currently planned core residence time.

The next phase of the development program would consist of the design, construction, and irradiation of demonstration assemblies to be used in high-burnup studies. The design should be similar to that of present fuel but incorporate changes in detail to accommodate the modified performance parameters identified earlier. These demonstration assemblies would be followed by an entire batch loading of high-burnup fuel.

For this fuel cycle, the waste treatment R&D refers to testing associated with the long-term storage of reactor fuel elements; it includes determination of containment performance, of the adequacy of heat removal methods, and of material compatibility under storage conditions. For the present LWRs, the only R&D requirement would be concerned with fuel storage. The testing period specified is associated with resolving technical issues and does not consider long-term demonstration.

9.2.2 SSCRs

The SSCR consists basically of a PWR nuclear steam supply system whose reactivity control system has been modified to utilize heavy water rather than soluble boron to compensate for reactivity changes during the operating cycle.

Fuel fabrication technology for the SSCR would be largely available by adaptation of LWR technology; however, because the enrichment of the SSCR fuel would be lower and distributed differently and because the specific element design will probably not be exactly the same as that for the LWR, a small R&D effort is required. Waste treatment R&D is estimated to be essentially the same as that for LWRs. Because the SSCR fuel would have different fuel enrichments and specifications than LWR fuels, fuel qualification testing is needed but would largely consist of verification studies. The fresh-fuel fabrication plant available to LWRs was considered to be available to SSCRs by small adaptations.

Other R&D required (such as that involving the D_2O upgrader) is not concerned with the fuel cycle and therefore not included in this study.

9.2.3 HWRs

The HWRs considered here differ somewhat from the typical CANDU-PHWR. Significant among the differences are the use of slightly enriched uranium fuel, higher pressures, and certain modifications to increase the generating capacity and to meet licensing requirements.

For HWRs using slightly enriched uranium, some R&D is needed for the fabrication of slightly enriched fuel which would allow higher exposures (relative to natural-uranium fuel). However, the estimated effort is small and would largely be associated with verification studies. Research and development is required in fuel qualification to demonstrate that fuel elements using slightly enriched uranium fuel perform adequately under commercial conditions. Storage R&D for such fuel elements is estimated to be about the same as that for HWRs with natural-uranium fuel since the heat removal requirements are relatively low in both cases and the element designs are similar. A fresh-fuel fabrication demonstration plant is included for this reactor on the basis that the present plants would not be converted to fabrication of slightly enriched uranium fuel.

9.2.4 Tandem LWRs/HWRs

The two options that have been proposed for converting discharged LWR fuels to a form that can be used directly to fuel HWRs in a tandem cycle are reconfiguration and refabrication. In the reconfiguration option, fuel rods are removed from an LWR fuel assembly, segmented, and loaded in an assembly which can be charged to an HWR. In the refabrication option, the fuel is removed from the LWR fuel rods and reloaded in rods to be used in the HWR fuel assembly. An analysis for these options indicated that the reconfiguration option has fewer unknowns, would require less development, and could be accomplished in a shorter time period. The fuel cycle would also be less expensive and less subject to fuel diversion. However, small power penalties in either type of reactor could considerably reduce the savings.

The reconfiguration procedure proposed in Sect. 5.5 involves the removal of the fuel rods from irradiated LWR fuel assemblies, cutting the rods into sections at the divider plug assembly provided in each rod, and then reloading into assemblies that are compatible for loading into existing HWRs.

Experimental assemblies in which the rods may be removed individually have been made by commercial fuel fabricators and are now being included in commercial LWR reactors. Licensing for this limited use has been approved. Similar fuel assemblies would need to be fabricated on a commercial basis and licensed for use in full cores. Previous experience with these assemblies leads us to anticipate few difficulties, but experiments to confirm this would be required.

The reconfiguration of the irradiated LWR fuel for loading into the HWR will be more difficult. Due to the high activity level, all of the operations will have to be done in remotely operated and maintained equipment. Experimental fuel assemblies which have previously been irradiated in LWRs have been remotely disassembled and examined on a research basis; however, these operations would need to be engineered for commercial operation. No previous experience exists in the segmenting, resealing, and loading of irradiated fuel rods into assemblies for an HWR.

Since the fuel rod diameter for the LWR is much less than that for the HWR, the HWR fuel assembly will have to be redesigned to accept a larger number of rods so that it will contain the same amount of fuel. Extensive testing will be required to qualify these fuel assemblies under the licensing process.

It is estimated that the development of the various steps in the tandem LWR/HWR fuel cycle using the reconfiguration option would require about 15 years and cost about $150 million. Qualification of the fuel for commercial use would require about 10 years for licensing within the 15-year period.

The refabrication option would require all of the R&D listed above plus the development of a method for physically removing the fuel from the LWR fuel rods and fabricating this material into HWR fuel rods. At present, no method is available for physically removing enough of the fuel from the rods so that the cladding would meet current discharge standards. If an acid leach were to be used to clean the cladding, the resulting solution would contain fissile material and thus be attractive for diversion. Refabrication of the irradiated fuel containing about 4% fission products into pellets or into materials for Vipac loading involves many unknown factors. It is estimated that the R&D for these operations will add about $50 million to the cost and about 2 years to the development time that would be required for the reconfiguration option.

9.3 Required R&D for Denatured Uranium-Thorium Cycles

The R&D requirements for denatured uranium-thorium fuels in LWRs, SSCRs, and HWRs are discussed in the following subsections.

9.3.1 LWRs

The denatured ^{235}U (^{235}U-thorium) and the denatured ^{233}U (^{233}U-thorium) fuel cycles are discussed in this section. In the ^{235}U-thorium fuel cycle the core is initially comprised of thorium fuel enriched with uranium containing ~20% ^{235}U. Uranium-235 is utilized to supplement recycle fissile uranium recovered from the spent fuel of previous cycles

of operation. Beginning in the fourth cycle of operation, the uranium (containing ^{235}U, ^{238}U, and small quantities of other uranium isotopes) is recycled to reduce the requirements for ^{235}U.

Since the fissile content of this recycle material is less than that required to achieve the specified cycle length (because of the depletion of fissile material during the prior cycle of irradiation), ^{235}U must be added. In this report it is assumed that fully enriched uranium (93 wt % ^{235}U) is used to "top" the recycle material. (The enriched material is added until the enrichment limit for denatured uranium is reached.) The topped recycle uranium is then mixed with thorium in the proper proportions required to achieve the specified cycle length; the remaining fuel consists of 20 wt % ^{235}U mixed with thorium (fresh denatured fuel).

The ^{233}U-thorium cycle is similar, except that ^{233}U is utilized in place of ^{235}U to provide the initial and makeup fissile inventories. Since ^{233}U is not found in nature, it must be produced via irradiation of thorium-bearing fuels.

The technology for the utilization of thorium-bearing fuels and for the recycle of ^{233}U is less well developed than that of the uranium fuel cycle. Use of the denatured thorium fuel cycle will therefore necessitate significant R&D effort in the areas of fuel fabrication and reprocessing, and as a prerequisite to reactor-related data base and verification-type development.

The radioactivity resulting from trace quantities of ^{232}U, which is produced along with the fissile ^{233}U from thorium fuels, makes the fabrication of ^{233}U-bearing fuels significantly different from that of ^{235}U-bearing fuels. However, fabrication of ^{233}U fuels is not significantly different from the fabrication of fuels from reactor-grade plutonium. Although the ^{233}U contains more gamma activity than the plutonium, both fuels would require remotely operated and shielded process facilities. While such remote and shielded facilities are easy to visualize conceptually, the engineering involved in preparing detailed designs is expected to be quite complex. One important problem which must be addressed is the maintenance of the remote equipment, since

long down-times for repair would compromise the economics of the fabrication process. The complexity of the pelletization process may make it desirable to fabricate ^{233}U-bearing fuels using Vipac or Sphere-Pac technologies — technologies which appear more amenable to remote operations. The use of Vipac or Sphere-Pac fabrication would, of course, necessitate additional R&D for process development as well as for in-reactor performance qualifications, since neither of these alternative fabrication technologies has been employed for the manufacture of commercial-grade fuels.

The denatured thorium fuel cycle also introduces significant new requirements for fuel reprocessing and waste treatment R&D. Reprocessing of thorium-based fuels is based on the Thorex process. Although parts of this process have been commercially demonstrated for lower-radiation-exposure fuel, it is much less developed than the Purex process, which is utilized for reprocessing uranium-based fuels. Since spent denatured thorium fuels will contain significant quantities of plutonium, in addition to uranium and thorium, a modified version of the Thorex process will have to be developed and tested. Reprocessing of the thorium-based fuels is also complicated by the fact that, unlike uranium, thoria dissolves very slowly in nitric acid unless fluoride ion is present. (Similar dissolution procedures may be required for MOX fuels.) The introduction of fluoride complicates the treatment of waste from the fuel-dissolving process and will necessitate additional R&D in this area. The fluoride ion also complexes with the zirconium cladding so that thoria dissolution may be retarded. However, recent laboratory experiments indicate that satisfactory dissolution may be obtained by making small modifications in the dissolution procedure. Another approach is to remove the cladding before dissolving the thoria by some chemical or mechanical method. Again, additional R&D will be required both to develop the dissolution process per se and to find ways to treat the waste that is generated during dissolution.

Although some experience has been gained with the irradiation of thorium-based fuels in LWRs, additional R&D will also be required in this area. Research and development in two major areas, data-base

development and fuel performance qualification, is necessary to ensure that fuel performance meets licensing requirements and to develop the information required for licensing thorium-fueled cores. Information on in-reactor densification and swelling behavior, fission gas release, thermal conductivity of fuel, pellet cladding interaction, and coefficients of reactivity must also be obtained. Subsequent R&D would consist of in-pile irradiation demonstrations where significant quantities of thorium-based fuels, fabricated with processes and equipment representative of commercial fabrication technology, would be irradiated to provide a demonstration of in-reactor fuel performance.

With regard to solvent extraction, the separation of thorium and uranium from fission products for fuels of high irradiation exposure in laboratory and pilot-plant tests has been demonstrated using the Acid Thorex process; however, highly irradiated fuel has not yet been processed in engineering-scale equipment. Denatured uranium-thorium fuels will have high fuel exposures (~40,000 MWd/MT) and will contain plutonium. Modifications to the Thorex process appear to be required in order to accommodate the extraction and separation of fuel products in the denatured uranium-thorium fuel cycle. Also, in using products from the solvent extraction steps, care must be taken so that ^{220}Rn (radon) does not lead to radiation release problems. Radon is formed by the decay of ^{228}Th and, while it has a relatively short half-life, must be controlled by the development of effective off-gas treatment methods.

New waste treatment requirements may arise because of the thoria dissolution problem. Wastes resulting from any chemical decladding of the Zircaloy (or from other treatment) would need to be handled and treated as high-level wastes. In addition, the presence of fluoride in the wastes from fuel dissolution will influence the choice of waste treatment process. A significant R&D effort appears to be needed in these areas.

9.3.2 SSCRs

This cycle differs only slightly from the previously described cycle for the LWR - denatured uranium-thorium system. Some R&D would be

required on heavy water use as described in the SSCR reactor paper.[4] These differences do not significantly affect the required fuel cycle R&D, which is essentially as outlined in Sect. 9.3.1 for LWRs in the denatured uranium-thorium cycle.

9.3.3 HWRs

The denatured uranium-thorium fuel cycle in an HWR requires the same fuel cycle R&D as is outlined in Sect. 9.3.1 for LWRs in the denatured uranium-thorium cycle. In addition, reactor R&D (discussion not included in this report) similar to that for the SSCR will be required.

9.4 Estimated Schedules and Costs

The preceding discussion confirms that some of the steps in the various fuel cycles are very similar, if not identical, and that an integrated program to develop several fuel cycles simultaneously would be both cost- and schedule-effective. To date, however, no decision has been made with regard to the sequence in which these fuel cycles might be required, or even which fuel cycles will be selected for development. Thus, in constructing the schedules and costs presented here, it has been assumed that the cycles will be developed independently of each other.

The information resulting from the R&D for the various recycle steps will be used to design, construct, and operate a fresh-fuel fabrication plant and/or a recycle demonstration plant. Each plant would operate on a semicommercial scale, would contain all elements of the development program, and would demonstrate the safety and licensing requirements of the commercial plants, as well as generate useful products. Because of the importance of installing prototypic processes and equipment in the plants, the schedule assumes that all R&D efforts will be completed prior to plant construction. With this approach, there is less risk that the plants will require modifications, although delays in the

overall schedule will probably be experienced due to the timing of construction. The throughput of the demonstration plant for fresh-fuel fabrication would be about 2 MT/day. The recycle plant costs considered a reprocessing capacity of about 2 to 5 MTHM/day based on modification and use of the Barnwell reprocessing facility (the lower throughput is associated with thorium fuel cycles). A refabrication capacity of 0.2 MT/day was considered for the demonstration facility.

Using the above approach, R&D schedules and costs for each fuel cycle considered are broken down into the following broad categories: fuel fabrication, refabrication, reprocessing, waste treatment, demonstration facility for fresh fuel fabrication, demonstration facility for fuel recycle, and fuel qualification. The categories of fabrication, refabrication, reprocessing, and waste treatment cover the conventional fuel recycle R&D areas. Fuel qualification provides acceptable proof that R&D has produced a fuel that meets commercial power reactor specifications, the evidence having been obtained through appropriate fuel irradiation testing. Fuel qualification cannot proceed until fuel specifications as well as fabrication processes and equipment have been developed to at least an engineering scale. The time to complete a fuel qualification program has been estimated to be 8 years after reasonably assured specifications are established. In some fuel cycles it will be appropriate to carry out capsule or small-scale fuel irradiation testing to establish specifications.

The demonstration facilities provide (for fresh fuel fabrication as needed, and for fuel reprocessing and refabrication) the step between R&D efforts and commercial facilities; because of the large capital investment and operating costs, the construction of demonstration facilities is undertaken only after the economic promise of the selected fuel cycle has been carefully considered. The time required to place a fuel recycle demonstration facility on line is generally estimated to be 10 years, including 3 years for Title I and II design, 5 years for construction, and 2 years for cold shakedown. The facility would be cycle specific

and must account for appropriate disbursement of all fissile, fertile, and waste materials associated with spent reactor fuel elements that are received. A good engineering cost estimate for the demonstration plants can be made only after Title I design has been completed. Such information was not available for the schedules presented here and could not be developed with the resources available. Thus the estimates are quite preliminary and based primarily on engineering judgment and experience; in addition, they consider effluent controls which appear attainable with present technology, although that technology is not fully developed. The estimates could be much higher if new technologies have to be developed.

The estimated time schedules and costs for R&D for the various fuel cycles in the different reactors are presented in Table 9.3. As has been pointed out earlier, it is difficult to accurately predict the costs and times required to resolve the inherent problems. Further, the costs of demonstration facilities cannot be estimated very accurately without performing detailed plant design studies, and these could not be carried out in the time available. Nonetheless, the schedules and costs (1978 dollars) were developed on a relatively consistent basis and should be useful for overall guidance. At the same time, the large uncertainty in absolute costs needs to be emphasized. Thus Table 9.3 also gives estimated ranges of R&D costs, which are roughly based on an uncertainty of -10% to +30% for base costs, of -10% to +50% for fresh fuel fabrication facilities, and of -10% to +100% for fuel recycle demonstration facilities. In addition, these ranges consider that the fuel for HWRs using uranium of low enrichment might be fabricated in present natural-uranium fuel fabrication facilities with relatively little modification. These ranges are not intended to imply lower and upper limits but are believed to be reasonable estimates of probable uncertainties.

10. ESTIMATED CAPITAL AND OPERATING COSTS

The following economic information has been derived from recent studies of the fuel fabrication, reprocessing, and refabrication portions of the

Table 9.3. Summary of estimated fuel recycle R&D costs and schedules for selected reactor types and fuel cycles

Fuel cycle	Estimated costs (millions of dollars)									Time to complete demonstration plant (years)		
	R&D					Demonstration facilities[a]						
	Fuel qualification	Fabrication	Refabrication	Reprocessing	Waste treatment	Fresh fuel (capital/oper.)	Recycle plant (capital/oper.)	Total	Estimated range	R&D	Fresh fuel fabrication	Fuel recycle
Once-through												
Improved LWR	30	10	-	-	50	-	-	90	40-70	6	Available	-
SSCR	30	10	-	-	50	-	-	90	80-120	7	Available	-
HWR	30	20	-	-	35	100/150	-	335	300-500	7	13	-
Tandem	60	-	150	-	35	-	520/320	1085	1000-2300	10	Available	20
Denatured U-Th												
LWR-^{233}U	60	-	225	220	350	100/150	750/320	2175	1900-3100	14	~10	~20
LWR-^{233}U	60	-	225	220	350	100/150	750/320	2175	1900-3100	14	~10	~20
SSCR	60	-	225	220	350	100/150	750/320	2175	1900-3100	14	~10	~20
HWR	60	-	225	220	350	100/150	750/320	2175	1900-3100	14	10	20

[a]Based on demonstration plant capacities of about 2 MTHM/day for the fresh fuel fabrication facility; the demonstration recycle plant costs correspond to use of the Barnwell Nuclear Fuel Plant at a capacity of 2.5 MT/day for reprocessing and a new facility of about 0.2-MT/day capacity for refabrication. The operating costs for the demonstration plants consider 4 years of facility operation.

fuel cycles. Although the absolute values have a reasonable degree of uncertainty, the relative values are considered to be more reliable because of the consistent manner in which the estimates were prepared.

10.1 Ground Rules and Economic Factors

Basic design assumptions for the various processing plants had to be established to ensure consistency in the cost estimates. Table 10.1 gives selected assumptions, which reflect the implications that the given fuel cycle is commercially established and that the plant is not the first of a kind, but rather an early commercial version of moderate capacity.

Table 10.2 provides the economic assumptions associated with three types of financing. Three sets of economic ground rules were used to provide a range of costs representing the minimum (government financing), a reasonable figure for use in fuel cycle economic comparisons (typical industrial financing), and a high value (high-risk industrial financing).

Table 10.3 presents the methodology used to convert the basic cost estimates into unit prices for each of the sets of economic ground rules. While there are various complex methods of relating the various cost areas of capital, operating, and capital utilization to arrive at a final unit price, the formula proposed in Table 10.3 presents a simple, flexible system. Basically, the approach is a discounted cash flow analysis to obtain an average, or levelized, product price. The capital utilization costs are defined in the last three entries for each set of economic ground rules and can be derived from the assumptions listed above.

10.2 Unit Cost Estimates for Fabrication and Reprocessing

The basic cost component estimates were derived from (1) an analysis of the functional flow diagrams to provide space requirements for equipment in each functional area and process support area, (2) cost estimates for each set of equipment, (3) definitions of hardware and expendable

Table 10.1. Design assumptions for fabrication and reprocessing plant cost analysis

Reference plant capacity:
Fabrication — 2 MT/day
Reprocessing — 5 MT/day
Effective full-production days per year:
Reprocessing — 300
Fabrication
Contact operation — 260
Noncontact operation — 240
Average shipping distance (base case): 1000 miles
On-site storage at reprocessing and refabrication plant: 30 days
Cooling time before reprocessing: 180 days
Recycle out-of-reactor time: 2 years
Reprocessing and refabrication shall be in separate facilities in the base case.
Limited storage of uranium and thorium products
Licensing requirements shall meet current NRC-ALARA criteria.
Effluents shall meet ALARA base case; if not, basis should be defined (including document reference).
Design criteria for shielding: 0.25 mR/hr at outside surface

Table 10.2. Typical assumptions used for discounted cash flow analysis

	Type of financing		
	Government	Typical industrial	High-risk industrial
Project life, years			
Construction period	6	6	6
Operating period	20	20	20
Decommissioning period	3	3	3
Capital structure			
Equity, %	0	65	100
After-tax return on equity, %/year	0	14	15
Debt, %	100	35	0
Interest rate on debt, %/year	7.5	8.3	0
Weighted avg. cost of money, %/year	7.5	12.0	15.0
Taxes			
Federal income, %	0	48	48
State income, %	0	3	3
Property taxes and insurance, %	0	3	3
Federal investment tax credit, %	0	7	7
Tax depreciation method	-	SYD[a]	SYD[a]
Tax depreciation life, years	-	16	16
Equipment replacement and maintenance charge, % of initial equipment cost/year	5	5	5
Charge rate during construction, %/year	7.5	10.5	10.5
On-stream efficiency, %			
Years 1-6	0	0	0
Year 7	33	33	33
Year 8	67	67	67
Years 9-26	100	100	100
Owner's cost during construction, % of annual operating cost			
Year 1	5	5	5
Year 2	10	10	10
Year 3	20	20	20
Year 4	30	30	30
Year 5	40	40	40
Year 6	40	40	40
Capital costs, % of total			
Year 1	2.5	2.5	2.5
Year 2	6.5	6.5	6.5
Year 3	18.2	18.2	18.2
Year 4	44.2	44.2	44.2
Year 5	27.1	27.1	27.1
Year 6	1.5	1.5	1.5
Derived fixed-charge rate	0.108	0.226	0.316
Charges during construction, fraction of total cost			
Capital expenditures, IDC_D	0.249	0.366	0.366
Owner's cost, IDC_O	0.209	0.303	0.303

[a]Sum-of-years digits.

Table 10.3. Unit price analysis formula

The formula used for computing unit production cost ($/kg, in constant dollars) is:

$$\$/kg = \frac{(C_D + C_O + C_C)\ R + O + M + E_R + D}{T} ,$$

where

$$C_D = C_F + C_E$$

$$C_C = IDC_O \cdot C_O + IDC_D \cdot C_D$$

$$E_R = A_R \cdot C_E$$

$$T = XF$$

and

C_D = design and construction costs, $ million

C_F = facility cost (excludes process equipment), $ million

C_E = equipment cost, $ million

C_O = owner's cost during construction, $ million

$C_O + C_D$ = direct capital

C_C = charge on direct capital during construction, $ million

IDC_D = fractional charge on design and construction cost during construction

IDC_O = fractional charge on owner's cost during construction

R = annual fixed-charge rate on capital, fraction per year

O = annual operating cost, $ million/year

M = annual hardware and expendable material cost, $ million/year

A_R = annual maintenance and replacement rate on equipment, fraction per year

E_R = annual maintenance and replacement cost, $ million/year

D = annual payment to establish fund for decommissioning, $ million/year

T = annual throughput achieved, millions of kg per year

X = design capacity of plant, millions of kg per year

F = average fraction of design capacity achieved

materials, and (4) an analysis of the manpower requirements. This procedure was repeated for each process for each reactor fuel cycle. The cost of uranium, plutonium, or thorium is not included since the plants operate as toll processors.

The net result of these estimates and the subsequent economic analyses are summarized in Table 10.4. The unit prices associated with the economic assumptions for a typical industry are recommended for comparative analysis of power generation costs for the various reactor cycles. The LWR and SSCR are estimated to have essentially the same unit fuel recycle costs, pending the evolution of a detailed design of the SSCR fuel element.

Table 10.4. Summarized cost estimates and derived unit prices for water reactor fuel fabrication and reprocessing[a]

Case(s)	Reactor fuel cycle	Process	Economic set[b]	Estimated costs: Facility cost, C_F ($10^6)	Equip. cost, C_E ($10^6)	Annual hardware and materials cost, M ($10^6)	Annual operat. cost, O ($10^6)	Derived costs: Owner's cost during const., C_O ($10^6)	Charge on direct capital during const., C_C ($10^6)	Annual equipment replac. cost, E_R ($10^6)	Annual payment to decommissioning fund, D ($10^6)	Annual production rate (kg of HM/yr)	Unit price ($/kg HM)
1, 1A	LWR/SSCR	Fabrication	A	32.0	34.2	23.0	14.1	20.4	20.7	1.7	0.6	520,000	100
	Improved		B	32.0	34.2	23.0	14.5	21.0	30.6	1.7	0.7	520,000	130
	Once-through		C	32.0	34.2	23.0	14.8	21.4	30.7	1.7	0.7	520,000	150
2	LWR-denatured U-Th ^{235}U makeup	Fabrication	A	34.8	46.5	24.5	14.6	21.1	24.7	2.3	0.7	520,000	110
			B	34.8	46.5	24.5	15.0	21.8	36.4	2.3	0.7	520,000	140
			C	34.8	46.5	24.5	15.3	22.2	36.5	2.3	0.7	520,000	170
		Reprocessing	A	795	335	7.7	39.4	57.1	293.3	16.7	1.8	1,500,000	150
			B	795	335	7.7	39.4	57.8	431.1	16.7	1.8	1,500,000	330
			C	795	335	7.7	40.2	58.3	431.3	16.7	1.9	1,500,000	385
		Refabrication	A	509.8	265.7	27.4	25.9	37.6	201.0	13.3	1.2	480,000	370
			B	509.8	265.7	27.4	26.5	38.4	295.5	13.3	1.2	480,000	660
			C	509.8	265.7	27.4	26.9	39.0	295.6	13.3	1.2	480,000	870
3, 3A	LWR/SSCR denatured U-Th ^{233}U makeup	Refabrication	A	509.8	265.7	27.4	25.9	37.6	201.0	13.3	1.2	480,000	370
			B	509.8	265.7	27.4	26.5	38.4	295.5	13.3	1.2	480,000	660
			C	509.8	265.7	27.4	26.9	39.0	295.6	13.3	1.2	480,000	870
		Reprocessing	A	789	333	7.7	39.4	57.1	291.3	16.6	1.8	1,500,000	150
			B	789	333	7.7	39.9	57.8	428.2	16.6	1.8	1,500,000	330
			C	789	333	7.7	40.2	58.3	428.3	16.6	1.9	1,500,000	380
4	HWR-LEU	Fabrication	A	21.3	33.2	11.2	11.4	16.5	17.0	1.7	0.5	520,000	65
	Once-through		B	21.3	33.2	11.2	11.6	16.9	25.1	1.7	0.5	520,000	90
			C	21.3	33.2	11.2	11.8	17.1	25.1	1.7	0.5	520,000	110
5	HWR-denatured U-Th ^{233}U makeup	Refabrication	A	453.9	247.3	17.7	18.5	26.8	180.2	12.4	0.9	480,000	310
			B	453.9	247.3	17.7	18.9	27.4	264.9	12.4	0.9	480,000	570
			C	453.9	247.3	17.7	19.2	27.8	265.0	12.4	0.9	480,000	760
		Reprocessing	A	864	364	7.9	40.5	58.7	318.0	18.2	1.9	1,500,000	160
			B	864	364	7.9	41.0	59.5	467.5	18.2	1.9	1,500,000	360
			C	864	364	7.9	41.4	60.0	467.6	18.2	1.9	1,500,000	420
6	LWR-HWR tandem	LWR-fabrication	A	32.0	34.4	26.2	14.4	20.9	20.9	1.7	0.7	520,000	110
			B	32.0	34.4	26.2	14.8	21.5	30.8	1.7	0.7	520,000	140
			C	32.0	34.4	26.2	15.1	22.0	31.0	1.7	0.7	520,000	160
		HWR-fabrication reconfiguration	A	500.8	216.7	4.3	17.3	25.1	183.9	10.8	0.8	480,000	280
			B	500.8	216.7	4.3	17.5	25.4	270.3	10.8	0.8	480,000	550
			C	500.8	216.7	4.3	17.7	25.7	270.4	10.8	0.8	480,000	740
		or HWR-fabrication reconstitution	A	605.1	250.3	17.1	20.2	29.3	219.1	12.5	0.9	480,000	350
			B	605.1	250.3	17.1	20.6	29.9	322.1	12.5	1.0	480,000	680
			C	605.1	250.3	17.1	20.9	30.3	322.3	12.5	1.0	480,000	900

[a] All costs and prices are given in constant 1978 dollars.

[b] A = government, B = standard industrial; C = high-risk industrial.

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