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NONPROLIFERATION ALTERNATIVE SYSTEMS
ASSESSMENT PROGRAM

U.S. DEPARTMENT OF ENERGY
ASSISTANT SECRETARY FOR NUCLEAR ENERGY
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FOREWORD

The Department of Energy (DOE) Nonproliferation Alternative Systems Assessment Program (NASAP) is a planned program of studies of nuclear power systems, with particular emphasis on identifying and then evaluating alternative nuclear reactor/fuel-cycle systems that have acceptable proliferation-resistance characteristics and that offer practical deployment possibilities domestically and internationally. NASAP was initiated in 1977, in response to President Carter's April 1977 Nuclear Power Policy Statement.

The objectives of NASAP are to (1) identify nuclear systems with high proliferation resistance and commercial potential, (2) identify institutional arrangements to increase proliferation resistance, (3) develop strategies to implement the most promising alternatives, and (4) provide technical support for U.S. participation in the International Nuclear Fuel Cycle Evaluation (INFCE) Program.

NASAP is not an assessment of all future energy-producing alternatives. Rather, it is an attempt to examine comprehensively existing and potentially available nuclear power systems, thus providing a broader basis for selecting among alternative systems. The assessment and evaluation of the most promising reactor/fuel-cycle systems will consider the following factors: (1) proliferation resistance, (2) resource utilization, (3) economics, (4) technical status and development needs, (5) commercial feasibility and deployment, and (6) environmental impacts, safety, and licensing.

The DOE is coordinating the NASAP activities with the U.S. Nuclear Regulatory Commission (NRC) to insure that their views are adequately considered at an early stage of the planning. In particular, the NRC is being asked to review and identify licensing issues on systems under serious consideration for future research, development and demonstration. The Preliminary Safety and Environmental Information Document (PSEID) is the vehicle by which NASAP will provide information to the NRC for its independent assessment. The PSEID contains the safety and environmental assessments of the principal systems. Special safeguards measures will be considered for fuel cycles that use uranium enriched in uranium-235 to 20% or more, uranium containing uranium-233 in concentrations of 12% or more, or plutonium. These measures will include the addition of radioactivity to the fuel materials (i.e. spiking), the use of radioactive sleeves in the fresh fuel shipping casks, and other measures. The basis for the safeguards review by NRC is contained in Appendix A.

The information contained in this PSEID is an overlay of the present safety, environmental, and licensing efforts currently being prepared as part of the NASAP. It is based on new material generated within the NASAP and other reference material to the extent that it exists. The intent of this assessment is to discern and highlight on a consistent basis any safety or environmental issues of the alternative systems that are different from a reference light-water reactor once-through case and may affect their licensing. When issues exist, this document briefly describes research, development, and demonstration requirements that would help resolve them within the normal engineering development of a reactor/fuel-cycle system.

The preparation of this document takes into consideration NRC responses to the DOE preliminary safety and environmental submittal of August 1978. Responses to these initial comments have been, to the extent possible, incorporated into the text. Comments by the NRC on this PSEID were received in mid-August 1979 and, as a result of these comments, some changes were made to this document. Additional comments and responses were incorporated as Appendix B. Comments that are beyond the scope and resources of NASAP may be addressed in research, development, and demonstration programs on systems selected for additional study. The intent of this document (and the referenced material) is to provide sufficient information on each system so that the NRC can independently ascertain whether the concept is fundamentally licensable.

This PSEID was prepared for the DOE through the cooperative efforts of the Argonne National Laboratory, the Oak Ridge National Laboratory, and NUS Corporation.

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Chapter 1

MINING AND MILLING

1.1 URANIUM

1.1.1 GENERAL DESCRIPTION

In 1977, the domestic uranium mining industry produced the equivalent of about 14,000 tons of yellowcake, with production split almost evenly between open-pit and underground mines (Ref. 1). Open-pit mining is normally used when the ore body lies under easily moved overburden at depths as great as 500 feet; underground mining is employed when the ore body is at greater depths or when it lies beneath rock that requires much blasting. Minor methods of uranium recovery (about 3% of the total) include heap leaching from low-grade ores, recovery from mine water or mill raffinate, and in-situ leaching.

An underground mine has service buildings, a head frame with a truck-loading facility, a mine-waste pile, and a flow of water pumped to surface drainage from underground sumps. The area occupied by the hoisting and loading equipment, shop, warehouse, changehouse, and office may be only a few acres, but the underground workings may extend a mile or more. The volume of the mine waste pile can be equal to the volume of processed ore.

The groundwater that enters underground workings contains a variety of dissolved constituents, including radium, radon, and uranium. As it passes through the mine, its composition is not significantly affected, although it may lose some radon and gain some uranium. In some underground mines, the uranium is recovered from the water.

Underground uranium mines discharge much more air than do other mines because it is necessary to dilute the radon gas. Fresh air is usually downcast through the production shaft, distributed through ore haulage ways, and then discharged through vent holes or shafts at the extremities of the ore body and at intermediate locations. Discharged air may contain significant amounts of rock dust and radioactive gases, but the large quantities of diluent air and the natural atmospheric dispersion result in concentration levels at the site boundaries usually several orders of magnitude lower than the standards prescribed in 10 CFR 20. The discharged air does not contain significant amounts of noxious gases or smoke, and the mine air shafts do not intrude on the landscape.

Open-pit mining is characterized by a large open excavation, large piles of earth and rock overburden placed nearby, a network of operating roads and yards, and a flow of mine water pumped into the local surface drainage. Shop, warehouse, office, and changehouse structures are usually nearby, and an assortment of heavy earth-moving equipment is present. During much of the operating life of the mine, overburden is used for backfilling the mined-out areas, thus minimizing haulage and the period of the physical and topographical impact of the operations. It is not practical to begin restoration until sufficient mined-out area is exposed. During the later stages of mining and cleanup, the work of filling the final pit is not economic. The final pit areas are sometimes converted to small man-made lakes rather than restored to conditions similar to the surrounding region.

A uranium mill extracts uranium from the ore by mechanical and chemical processing. A semirefined product is sold in terms of its yellowcake (U_3O_8) content. There were 18 conventional uranium mills operating in the United States at the end of 1977, with a combined nominal processing capacity of 36,000 tons of ore per day (Ref. 1). All were in the western states: seven in Wyoming, five in New Mexico, two each in Colorado and Utah, and one each in Texas and Washington. Two new conventional mills were scheduled to start up in 1978, two existing plants are in the process of expanding capacity, and six new mills are planned for probable startup during the next 4 years.

In addition to the conventional uranium mills, six commercial or pilot-scale solution-mining facilities were in operation or planned in Texas, five development solution-mining installations are in operation, and several commercial-scale installations are planned in Wyoming. One central and one satellite plant for recovering uranium as a byproduct from phosphoric acid production are operating in Florida, with additional plants planned for recovering as much as 3,000 tons of yellowcake per year by 1980. Other plants that recover small quantities of uranium from mine water are in operation or planned in New Mexico, Wyoming, and Utah. Several heap-leach recovery plants are in operation or planned in western states. The total capacity of all the nonconventional uranium-recovery facilities is expected to remain a small fraction of U.S. milling capacity through the end of this century.

Two alternative methods are employed for uranium milling in the United States, the acid-leach and the alkaline-leach (sodium carbonate) methods, depending on the ore being processed. They differ in effluent chemical composition because of the reagents used and the higher concentration of impurities in the acid-leach tailings liquor. The acid-leach process consumes 3.5 times more water, giving correspondingly higher waste-discharge volumes. The alkaline-leach process dissolves slightly more of the radium in the ore than does acid leaching. Thus the acid-leach process discharges more of the radium with the tailings and creates a slightly more radioactive solid waste. Most of the ores are best suited to the acid-leach process.

1.1.2 SAFETY CONSIDERATIONS[a]

Postulated accidents in mining operations include flooding, the collapse of mine walls, and fires. Flooding could result from the failure of drainage pumps or in open-pit operations from unusual rainfall. The amounts of dissolved and suspended material in flood water should be comparable to those in normal drainage water. Thus during the recovery operation, which requires increased pumping to empty the mine, the dissolved and suspended materials should be comparable to those released during normal operations.

In an underground mine, the resumption of mine ventilation after a power failure would result in a transient condition during which higher than normal concentrations of radon would be discharged to the atmosphere. Such an occurrence is not expected to affect the offsite environment measurably, particularly since there would be no increase in the total radon release.

The walls of the mine could collapse from instability resulting from rock fractures or underground water. In addition to the obvious risk to the workers, the collapse of earth into the mine could increase the concentration of suspended solids in the

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[a] See References 2, 3, and 4 for a more detailed treatment of safety considerations.

drainage water. The net effect would be negligible, however, because most of the mine drainage water is recycled to groundwater through natural seepage.

For mines using a settling pond, the failure of a retention-basin dike could result in the release of a large volume of mine drainage water, with an appreciable amount of settled slimes being discharged to nearby streams. Although these solids might contain small amounts of uranium and its daughter products, such a release would not be expected to have a significant effect on the environment.

An accidental fire involving petroleum products in the mine is credible but unlikely because of industrial safety precautions. A fire could release combustion products, but not radioactive material. Because uranium mines are typically remote from significant population centers, the environmental impact of such a fire would be negligible.

The most likely types of accidents associated with uranium milling that would be of sufficient magnitude to affect the offsite environment are inadvertent discharges of tailings to nearby surface waters or a major fire in a solvent-extraction circuit. Potential events that could result in offsite releases of tailings include the failure of a tailings dam as a result of earthquakes or flooding, equipment failure (e.g., the rupture of a tailings distribution pipeline), and operating errors, such as permitting the overflow of a tailings pond through inattention.

An earthquake-induced failure of a tailings dam is unlikely since most western mill sites are in Seismic Zone 1, a region of "minor damage." A flood-induced failure of a primary tailings dam is also unlikely since much of the industry is located in semi-arid regions. The risks from flooding would be decreased further by locating the dams below areas of limited drainage, by providing cutoff dams and diversion ditches above the pond area, and by the standard practice of requiring that a 5-foot minimum freeboard be maintained during operation of the tailings impoundment system.

If an operational or equipment-failure accident did occur, it would probably not involve a sudden catastrophic failure of the dam. The stored sand and slimes tend to be resistant to flow and are not readily transported by the water in the pond that is retained behind the low-head dam. The risks from flooding would be decreased further by locating the dams below areas of limited drainage and by providing cutoff dams and diversion ditches above the pond area. New U.S. Nuclear Regulatory Commission (NRC) guidance for the design and construction of tailings dams contains guidance on determining acceptable free board and requires that tailings dams be designed to prevent failure due to a probable maximum flood.

In the event of a tailings release, recovery would be relatively uncomplicated: the affected drainage system would be surveyed; waste solids and contaminated solids could be removed and buried or returned to the tailings impoundment system. The residual environmental impact, if any, would be small.

Several years ago, tailings-dike failures did occur. On evaluation of the dike construction, it was determined that the dikes needed strengthening. Mills having dikes similar in construction to those that failed were required to strengthen the dikes, and new mills were required to use new construction standards. Table IX.G.3-7 of Reference 3 contains a summary of tailings accidents from 1959 to 1971, assembled from the files of the U.S. Atomic Energy Commission.

The solvent-extraction circuit, where several thousand gallons of solvent (mostly kerosene) containing as much as several thousand pounds of natural uranium are used in the refining process, represents a potential for a serious fire. A large volume of dense smoke is characteristic of a petroleum fire, but otherwise the radiological or chemical contamination of the environment would be insignificant beyond a few hundred feet from the source. Recovery would entail surveying the surrounding area for uranium and removing or burying some contaminated soils. In 1968, two large fires involving solvent-extraction circuits occurred in two separate mills. Damages amounted to $300,000 in one case and almost a million dollars in the other. Although some 2,000 to 3,000 pounds of uranium were present in the circuits at the time of the fires, in neither case was there an appreciable release of uranium to the unrestricted environment. In both cases, essentially complete recovery of the uranium was expected.

Such accidents as overflows from process tanks, process-line failures, failure or malfunction of offgas filtration or scrubbing equipment, or even large spills of such reagents as sulfuric acid or kerosene from storage tanks may occur in uranium-mill operations as in any chemical process industry. The probability of such accidents having any significant effect on the offsite environment is negligible.

1.1.3 ENVIRONMENTAL CONSIDERATIONS

The principal environmental effects of uranium mining and milling result from the disturbance of land by mining operations, the disposal of mill tailings, and the release of radioactive elements, primarily radon gas (radon-222), from the ore body and the mill tailings.

Underground mining disturbs the land primarily through the disposal of the mine wastes, which can be equal to the volume of the mined ore. This material can be distributed in the vicinity of the mine so that it blends into the appearance of the surroundings. The use of land for service buildings and loading facilities is small. Minor impacts may be caused by the disposal of mine drainage water to surface drainage channels.

Open-pit mining has greater land impacts because of the large open excavations, the large piles of earth and rock overburden that must be stored nearby for extended periods, and the extensive network of roads and yards required for the operation. However, usually much of the overburden is used later to backfill the mined-out areas and to restore the land to its original condition. After the final closing of the mine, some open pits usually remain. They are often converted into small lakes. As in underground mining, land is used for service buildings and loading facilities, and mine drainage water may be discharged to surface drainage channels.

This environmental assessment is based on a 3,000-acre open-pit mining operation in a western state. Open-pit mining provides about 50% of the uranium produced today in the United States and should maintain close to this percentage in the future. The characteristics of a model open-pit uranium mine are shown in Table 1-1. The ore production rate of 1,600 metric tons (MT) per day for 300 days per year corresponds to a production of about 960 metric tons of yellowcake per year, assuming an average grade of 0.2%. The model mine has about 10 years of productive mine life. In addition to the land and water uses shown in Table 1-1, the model mine would require about 1,300 MW-hr of electricity and about 100,000 gallons of diesel fuel annually.

The environmental effects of chemical and radiological effluents from mining operations have been discussed in detail in References 2, 3, and 4. Chemical effluents include gaseous waste from the combustion of petroleum products, mine drainage water, and airborne dust from open-pit operations. The total gaseous effluents released annually by the model mine are given in Table 1-2.

The principal radioactive effluent is radon. Reference 3 derives a conservative radon release rate from the model mine of 3 Ci/day.[a] It also estimates an annual release to the watershed of 0.086 curie of uranium-238 and an equivalent amount of each of its daughter products, thorium-234 and uranium-234.

The principal impact of uranium milling is the impact on the land because most of the milled ore becomes tailings. The tailings are generally covered with earth, landscaped, and planted to create a natural appearance and to prevent water and wind erosion. Use of this land is currently restricted to minimize radiological exposures to the residual uranium and its decay daughters contained in the tailings.

The principal radiological concerns of uranium mining and milling are the release of radon gas (radon-222) from the ore and mill-tailings piles and the resultant occupational exposures of the miners and mill workers. Population doses from these releases are quite low because the mines and mills are generally in remote and sparsely populated areas.

The model uranium mill is chosen to process this output from one model uranium mine--that is, 1,600 metric tons of ore per day containing 0.2% yellowcake to yield 960 metric tons of yellowcake per year. A conservative estimate of the mill lifetime is 20 years. The model mill uses the acid-leach/solvent-extraction process. The requirements of such a mill for land, water, chemicals, and electrical energy are given in Table 1-3.

The environmental effects of chemical and radioactive effluents from milling operations have been discussed in detail in References 2, 3, and 4. Table 1-4 summarizes the effluent emissions to the atmosphere from the model mill from fossil-fuel combustion as well as the acid-leach and solvent-extraction processes. The principal radioactive effluent is radon. Reference 3 estimates that the radon gas releases for the model mill are $0.45 \times 10^6 \mu$Ci/day from ore piles, $0.74 \times 10^6 \mu$Ci/day from ore crushing, and $1.0 \times 10^6 \mu$Ci/day from the tailings-retention system. These figures correspond to a total release of 0.83 Ci/MT yellowcake.

More recent estimates (Ref. 5) based on a mix of acid-leach and solvent-extraction mills (85%) and alkaline-leach and solvent-extraction mills (15%) and a composite model of tailings areas (mix of pond and wet-beach areas and dry-beach areas) result in larger radon releases of 3.2 Ci/MT yellowcake during the period of mill operation. After milling operations stop, radon continues to be released from the tailings. The current objectives of the NRC are to reduce radon releases to about twice the background level by isolating or stabilizing the tailings. Reference 5 assumes a 5-year period to dry out the tailings and complete the stabilization. Over this 5-year period radon release is estimated at 1.4 Ci/MT yellowcake. After stabilization, the radon release is estimated to be 0.004 to 0.04 Ci/MT yellowcake annually.

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[a] This is equivalent to 1,095 Ci/yr, or 1.14 Ci/MT yellowcake. Estimates (Ref. 5) based on underground mining of much lower grade ores (0.1%) result in higher values, specifically, 16.6 Ci/MT yellowcake. This higher value is used in Table 1-2.

All liquids are retained in the tailings pond, which is constructed to prevent any discharge of liquids into the surface-water system. Therefore, no release of radioactivity to the watershed is estimated to occur.

1.1.4 LICENSING STATUS AND CONSIDERATIONS

The Federal Government has no licensing authority over uranium mines. The NRC licenses uranium mills and assesses the environmental impact of radiological emmissions from mines that are part of a mine-mill complex. Various Federal regulations apply, however, to the occupational health and safety of miners and mill workers and to environmental impacts on air and water quality. The states also have regulatory authority over mines through the environmental impact statement process.

Government regulatory authority over mill tailings is somewhat fragmented at present between the NRC, the U.S. Environmental Protection Agency, and individual states covered by the Agreement States Program. Efforts are under way (Ref. 6) to provide uniform and effective regulation of mill tailings. The NRC has the following studies of environmental effects of mining/milling in progress:

1. Preparation of a generic environmental impact statement (GEIS) on milling. This GEIS will develop detailed analyses of the impact of uranium-mill tailings.
2. Study of environmental effects of in-situ leaching of uranium on a large scale. The studies will address, among other things, long-term impacts from mill tailings and the control of these impacts; and potential impacts from large-scale in-situ uranium leaching.

These studies, together with the existing regulatory base and new data available from existing operations, should permit continued licensing of uranium mining and milling facilities.

1.1.5 DECOMMISSIONING AND DECONTAMINATION

There have been no standard procedures for decommissioning uranium mines and mills to date, although a number of mines and mills have been shut down and abandoned over the 30-year history of the industry. Legislation dealing with the reclamation of strip mines is presently under preparation and review in Congress and will apply to open-pit uranium mines.

The problem of abandoned mill tailings is recognized, and, as noted in Section 1.1.4, steps are now being taken to ensure safe onsite disposal of mill tailings (Ref. 7) and to establish land cleanup criteria for decommissioning uranium-mill sites (Ref. 8). Reference 9 presents a generic study of the costs and relative effectiveness of various methods of stabilizing mill tailings ponds by varying the type and thickness of the cover material used.

1.1.6 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

Most government-sponsored research, development, and demonstration in uranium mining and milling will be related to attempts to define the reserves and resource base. The National Uranium Resource Evaluation program is an example of a major effort under way at present. Government research on methods of recovering uranium from alternative sources, not presently exploited, can be expected to continue.

Research and development will be required to analyze and quantify the transport and dispersion of potentially harmful radioactive or toxic elements in the effluents from uranium mines and mills.

Table 1-1. Assumed characteristics of the model uranium mine

Ore production, MT/day	1,600
Yellowcake equivalent, MT yellowcake per year	960
Overburden moved, MT/yr	14,000,000
Water drainage, gal/min	~1,250
Land disturbed, acres/yr	100
Ten-year mine life totals	
Overburden moved, MT	144,000,000
Overburden stored, MT	19,400,000
Land disturbed, acres	1,000
Covered by overburden storage, acres	250
Exclusion area, acres	3,000
Land permanently disturbed, acres	120

Table 1-2. Effluents from the model uranium mine

Effluent	Total release
Particulates, MT/yr	0.72
Sulfur oxides, MT/yr	1.5
Carbon monoxide, MT/yr	12.4
Hydrocarbons, MT/yr	2.1
Nitrogen oxides, MT/yr	20.4
Radon-222 (to atmosphere), Ci/MT yellowcake[a]	16.6
Uranium-238 (to watershed), Ci/yr	0.086
Uranium-234 (to watershed), Ci/yr	0.086
Thorium-234 (to watershed), Ci/yr	0.086

[a] Based on an ore grade of 0.1% and measured data from underground mines.

Table 1-3. Assumed characteristics of the model uranium mill

Capacity, MT ore per day	1,600
Production rate, MT yellowcake per year	960
Land use, acres	300
Water use, gal/yr	170 x 10^6
Chemical use, MT/yr	
Sulfuric acid	13,000
Sodium chlorate	980
Sodium chloride	330
Ammonia	330
Manpower use, man-hr/yr	14,000

Table 1-4. Estimated effluents from the model uranium mill

Effluent	Emission rate
Sulfur oxides, MT/yr	0.23
Carbon monoxide, MT/yr	0.065
Hydrocarbons, MT/yr	6.6
Nitrogen oxides, MT/yr	64.0
Aldehydes, MT/yr	0.49
Organic acids, MT/yr	0.66
Ammonia, MT/yr	1.6
Kerosene, MT/yr	0.17
Particulates, MT/yr	160
Radon-222 (to atmosphere), Ci/MT yellowcake	
Active milling	3.2
Inactive mill, interim tailings pile	1.4
From stabilized tailings	0.004-0.04
Uranium-238 (to atmosphere), Ci/yr	0.39
Uranium-235 (to atmosphere), Ci/yr	0.017
Uranium-234 (to atmosphere), Ci/yr	0.39
Thorium-234 (to atmosphere), Ci/yr	0.18
Thorium-230 (to atmosphere), Ci/yr	0.18
Radium-226 (to atmosphere), Ci/yr	0.18

1.2 THORIUM

1.2.1 GENERAL DESCRIPTION

Most of the world's thorium production has been from monazite sands, which are mainly phosphates of the rare-earth elements and are formed by the weathering of rocks such as granites. Running water carries the sands to places where the heavier minerals settle in "placer deposits." Placer deposits may be formed in a river or in coastal locations. Monazite-bearing stream placers in the Piedmont region of North and South Carolina were the world's first source of thorium, with production starting in 1887. Production from this source ceased in 1917 and was supplanted by lower cost monazite beach placers in India, Malaysia, Brazil, and Australia. Recent U.S. production of thorium has been a byproduct of monazite recovered in mining titanium from beach placers in Georgia and Florida. Present world production of thorium is estimated to be less than 150 metric tons of ThO_2 equivalent per year. This might be increased by a factor of five if thorium were recovered from all the monazite mined.

Beach placers are an unlikely source of new production in the United States because of the high population density of the coastal region. The most promising thorium deposits for large-scale exploitation are thorite-bearing veins such as those in Colorado, Idaho, Montana, and Wyoming.

Total U.S. thorium reserves have been estimated as 553,070 short tons as thorium dioxide. Probable potential resources are estimated at almost five times the estimated reserves. Approximately 35% of the U.S. thorium reserves producible at a forward cost of less than $30 a pound of thorium dioxide occur in vein deposits (Ref. 10). More than half of the $30 reserves in veins is estimated to be available in the Lemhi Pass district of Idaho and Montana, which lies astride the Continential Divide. Most of the remaining thorium reserves and resources in veins occur in the Wet Mountain district in Colorado. Flat or mildly dipping veins near the surface would be mined first by open-cut methods. Underground mining may have to be used later as surface deposits are depleted. The radiological effects on the environment of open-pit and underground mining should be similar.

The model mine for thorium is assumed to be in a mountainous region at 7,000 to 10,000 feet above sea level. It employs open-pit mining methods similar to the model uranium mine, except that smaller quantities of overburden are removed, a higher grade of ore is present (~0.5% thorium dioxide), and the ore is in veins rather than widely disseminated.

Mining of the thorite veins would progress in a series of small open pits as opposed to the single large pit typical of uranium mines. The average vein in the model mine is taken to be 25 feet thick, extending to a depth of 200 feet, and in working lengths on the order of 1,400 feet. The working width for the open pits would be about 100 feet.

There is at present no industry in the United States that produces thorium as a primary product and no U.S. industrial experience with producing thorium from thorite. A model plant for milling and refining thorium ore was patterned after typical mills for uranium ore. The process involves crushing and grinding the thorite to a small mesh to allow suspension in a slurry. A sulfuric acid leach is used to extract the thorium. Solids are separated by decanting, and the thorium-containing leach liquor is then purified and concentrated by an amine solvent-extraction process to produce a crude thorium product. This crude product is subsequently refined by solvent extraction with tributylphosphate to yield reactor-grade thorium nitrate tetrahydrate.

1.2.2 SAFETY CONSIDERATIONS

The accidents postulated for typical uranium mines in Section 1.1.2 are also appropriate for the conceptual thorium mine. These accidents, which include flooding, wall collapse, and fire, should not result in offsite environmental effects, even though the model thorium mine, because of its location in a mountainous region of high seismic risk, would have a greater probability of experiencing earthquake damage than would the uranium mine. The expected lower generation rate of mine drainage water and waste rock in the thorium mine as compared to the uranium mine should reduce the potential consequences. The likelihood of flooding in the thorium mine should also be lower because natural drainage is expected to be used for dewatering the mine.

The accidents postulated for a typical uranium mill in Section 1.1.2 are also appropriate for the conceptual thorium mill. Because of its location in a mountainous region, the conceptual thorium mill will have a greater probability of experiencing earthquake damage. Even so the tailings-dam failure and the solvent-extraction-circuit fire discussed in Section 1.1.2 would not be likely to have appreciable environmental effects beyond the boundaries of the site.

1.2.3 ENVIRONMENTAL CONSIDERATIONS[a]

The principal environmental effects of thorium mining and milling will result from the disturbance of land by mining operations, the disposal of mill tailings, and the release of radioactive elements, principally radon-220, from the ore body and the mill tailings.

This environmental assessment of thorium mining and milling is based on a conceptual open-pit mine and mill in a remote mountainous region of the western United States (Ref. 11). Mine and mill characteristics are listed in Table 1-5, which also shows the differences in tailings-impoundment characteristics that might be expected in the Lemhi Pass area (Montana) and in the Wet Mountain area (Colorado). Ore production and mill processing rates are assumed to be the same as for the model uranium mine and mill (1,600 MT/day for 300 days per year). Assuming an average grade of 0.5% equivalent thorium dioxide in the thorite ore and 91% mill recovery, the conceptual mill will produce about 4,500 metric tons of thorium nitrate tetrahydrate per year. The mine and mill are assumed to have a 20-year lifetime, and it is assumed that 3 acres of thorium-bearing ore are exposed in one or more pits.

The model thorium mine would use much less land and water than the model uranium mine. Land, water, chemical, and power use at the conceptual thorium mill would be approximately the same as those given in Table 1-3 for the model uranium mill. The principal environmental effect of thorium mining and milling results from the release of thorium and its daughter products. Radon-220 with a half-life of 55.6 seconds is the thorium decay product of most concern, whereas the uranium decay product of most concern is radon-222 (half-life = 3.62 days). Because of the shorter

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[a] See Reference 11; and Reference 3 (Appendix IX G.3) for a more detailed discussion.

half-life of radon-220, only ore very near the surface contributes to radioactive emissions to the atmosphere from in-place ore. Mined ore has a higher volumetric source strength, however and the overall release rate of radon-220 from thorium mining and milling is greater than that of radon-222 from uranium mining and milling.

In the absence of measurements on radon-220 release from vein-type ore deposits and from thorite mining and milling, Reference 11 presents calculated results for radon-220 released from the exposed ore, from fugitive dust in the mining operation, from the ore-storage pile at the mill, from mill and refinery operations, from the tailings beach and pond during 20 years of operation, and from the final evaporation of the pond and covering of the dry tailings after the mill is closed. The calculated source terms for all the isotopes in the thorium decay series are summarized in Table 1-6. Placing about 1 foot of earth cover over the dry tailings pile after mill operations have ceased would reduce source terms for the tailings pile and airborne particulates to negligible values compared with the values during operations. The stabilized thorium-mill tailings do not have a potential long-term source term as do the uranium mill tailings, which contain radium-226 (half-life = 1,622 years) and its secular equilibrium daughter radon-222 (half-life 3.62 days). The longest-lived daughter in the thorium-232 decay chain is radium-228 (half-life = 5.75 years). After a decay period of about 10 half-lives (57.5 years), the tailings pile will be equivalent to a thorium ore body containing 0.05% thorium oxide equivalent.

1.2.4 LICENSING STATUS AND CONSIDERATIONS

Since there is no domestic thorium mining and milling industry, there has been no licensing history for such activities. Licensing considerations for uranium mines and mills, however, would be directly applicable to thorium mining and milling, and no insurmountable problems unique to thorium are foreseen (Ref. 12). Attention would have to be given to the radon-220 release from in-process material, which is not present in uranium processing.

1.2.5 DECOMMISSIONING AND DECONTAMINATION

Decommissioning and decontamination considerations for uranium mines and mills (Section 1.1.5) are expected to be applicable to thorium mines and mills as well.

1.2.6 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

A modest amount of research and development will be required to analyze and quantify the transport and dispersion of potentially harmful radioactive or toxic elements in the effluents from thorium mines and mills.

Table 1-5. Characteristics of the open-pit thorium mine and the model thorium mill and refinery

Mine	
Approximate total area, m^2	49,000
Exposed thorium-bearing veins, m^2	12,000
Ore production, MT/day	1,600
Average thorium content, % ThO_2 equivalent	0.5
Water drainage, m^3/day	680
Average depth, m	23

Mill and refinery	
Ore capacity, MT/day	1,600
Days of operation annually	300
Thorium recovery efficiency, %	
Mill	91
Refinery	99.5
Production of thorium nitrate tetrahydrate, g/yr	4.5×10^{12}
Water required, m^3/day	2,400
Ore pile size	
Meters	100 x 32 x 15
Metric tons	96×10^6

Tailings impoundment	Montana	Colorado
Average area during 20-yr mill life, m^2		
Dry beach	4,000	36,000
Pond	57×10^4	49×10^4
Average area exposed after mill closing, m^2		
Dry beach	4,000	4,000
Pond	38×10^4	27×10^4
Air discharge from complex, m^3/sec		11.3
Filter losses, %		
Crusher dust		0.7
Thorium nitrate tetrahydrate product line[a]		0.05

[a] Bags plus high-efficiency particulate air filters.

Table 1-6. Estimated source terms for operation and closing of the model mill

Nuclide	Source term[a] (Bq/sec)					
	Ore handling	Mill and refinery	Tailings beach and pond		Covering dry tailings	
			Montana	Colorado	Montana	Colorado
Thorium-232	0.52	0.18	0.037	0.33	0.37	0.37
Radium-228	0.52	0.18	.33	3.0	3.3	3.3
Actinium-228	0.52	0.18	.33	3.0	3.3	3.3
Thorium-228	0.52	0.18	.26	2.2	2.4	2.4
Radium-224	0.52	0.18	0.26	2.2	2.4	2.4
Polonium-216	0.52	0.18	0.26	2.2	2.4	2.4
Bismuth-212	0.52	0.18	0.26	2.2	2.4	2.4
Lead-212	0.52	0.18	0.26	2.2	2.4	2.4
Thallium-208	0.18	0.07	0.09	0.8	0.8	0.8
Polonium-212	0.33	0.11	0.18	1.4	1.6	1.6
Radon-220	1.5×10^7	3.6×10^7	1.2×10^7	1.6×10^7	1.1×10^7	9.3×10^6

[a] 1 Bq/sec = 27 pCi/sec.

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Chapter 2

URANIUM HEXAFLUORIDE CONVERSION

Before nuclear fuel is enriched in uranium-235 by either gaseous diffusion or centrifugation, it must be converted to the volatile compound uranium hexafluoride. The chemical form before conversion is either yellowcake concentrates resulting from mining and milling of ore or uranyl nitrate from a chemical reprocessing plant (in the recycle of spent fuel). This chapter discusses the conversion of yellowcake concentrates to uranium hexafluoride in Section 2.1 and covers the conversion of uranyl nitrate to uranium hexafluoride in Section 2.2.

2.1 CONVERSION OF YELLOWCAKE CONCENTRATES TO URANIUM HEXAFLUORIDE

2.1.1 GENERAL DESCRIPTION

Two processes are used commercially in the United States to convert yellowcake concentrates to uranium hexafluoride. One method, fluorination and fractionation, consists of continuous successive reduction, hydrofluorination, and fluorination of the concentrates followed by fractional distillation of the crude uranium hexafluoride to obtain a pure product (Ref. 1). This process is used in the Allied Chemical plant at Metropolis, Illinois, with a nominal annual capacity of 14,000 metric tons of uranium (MTU). An alternative method, a wet process, uses a wet chemical solvent-extraction step at the head end of the process to prepare a high-purity uranium feed before reduction, hydrofluorination, and fluorination (Ref. 2). A wet process is used in the Kerr-McGee plant at Gore, Oklahoma, with a nominal capacity of 9,000 MTU/yr. The effluents from the two processes differ. Most of the impurities entering with the uranium concentrate feed are rejected from the fluorination-fractionation process as solids; in the wet process most of these impurities are dissolved solids in the raffinate stream (Ref. 1 and 2).

Selection of the manufacturing process to be used in future conversion facilities will depend on such factors as relative unit production costs of uranium hexafluoride, process technologies, and environmental control technologies and costs. Other studies have assumed a mixture of facilities, with the wet process accounting for 40 to 50% of the total capacity (Refs. 3 and 4). The plant capacities cited in published documents have also varied from 5,000 to 15,000 MTU/yr (Refs. 1-6).

The model plant in this document is assumed to process 10,000 MTU/yr, 5,000 MTU/yr by the fluorination and fractionation process, and 5,000 MTU/yr by the wet process. The base case is assumed to be derivable from the Environmental Survey of the Uranium Fuel Cycle (Ref. 3), with a doubling of environmental effects given in Table 0-5 of that document, even though the effects would probably be smaller and the potential technology for environmental control may be somewhat improved over the status assumed for that document. This report assumes that the environmental impact is the same as that in Reference 3 on a per-reactor basis, provided the reactor-fuel design and fuel mass flows remain unchanged.

2.1.2 SAFETY CONSIDERATIONS

Various postulated accidents have been analyzed for their potential for offsite effects. They include the following:

1. Fire in the solvent-extraction operations of a wet-process plant
2. Failure of a uranium hexafluoride product cylinder (or cylinder valve) releasing essentially all the cylinder's contents
3. Failure of a raffinate-pond dike in a wet-process plant
4. Failure of a uranyl nitrate evaporator in a wet-process plant
5. Rupture of a tank holding hydrofluoric acid
6. Other accidents typical of many manufacturing operations.

These accidents and their potential consequences are discussed in Reference 3, pages C-18 through C-23, and Reference 4, pages IVF-38 through IVF-39.

On the basis of evaluations made to date and the operating records of both U.S. facilities, there is no reason to believe that uranium hexafluoride conversion facilities cannot operate safely with minimal impact on the health and safety of both workers and the general public.

2.1.3 ENVIRONMENTAL CONSIDERATIONS

The environmental considerations for the model uranium hexafluoride conversion facility are discussed in Reference 3, Section C.4, and Reference 4, pages IVF-34 through IVF-41. These discussions are not repeated here; however, Table 2-1 summarizes the major environmental impacts of the model plant. This table has been derived from Table 0-5 of Reference 3 by doubling all values, under the assumption that doubling the plant size doubles the impact.

With regard to Table 2-1 it should be noted that doubling capacity does not double all the environmental effects identified in Reference 3. For the purposes of this document, a doubling of effects has two purposes:

1. The impact per reactor presented in Reference 3 is the same, provided the reactor and fuel-cycle characteristics do not change.
2. A doubling of all numbers is a conservative upper limit for the model facility. (See further discussion on conservatism below.)

It should also be noted that future improvements in environmental control technology have the potential to reduce some of the effects listed in Table 2-1 (Ref. 6). Furthermore, the data in Reference 3 are being revised by the NRC; hence, the impacts may change from those shown. These changes for uranium hexafluoride conversion are, however, expected to show little or no increase in the environmental effects of uranium hexafluoride conversion.

Different fuel cycles and reactor systems under study as part of the NASAP evaluations will not modify the effects of the model plant for converting yellowcake concentrates to uranium hexafluoride. The only impact of alternatives to the LWR on the once-through cycle is the total industry impact caused by a change in the total demand for conversion services, which changes the number of model plants required to support the industry.

The environmental impact of uranium hexafluoride conversion is small in comparison to that of the total fuel cycle (Ref. 3). The annual individual whole-body dose from the operation of a fluorination and fractionation plant with a 10,000-MTU/yr capacity has been estimated to be between 10 and 13 mrem/yr for the specific cases examined (Ref. 6). Therefore, future expansion of this industry should not result in major environmental impacts.

2.1.4 LICENSING STATUS AND CONSIDERATIONS

The two U.S. facilities for uranium hexafluoride conversion have operated successfully and have both been expanded in output without any major licensing problems. It is anticipated that any further increase in capacity beyond that now existing can be licensed to operate safely and in an environmentally acceptable way.

One area that may require additional attention in the future is that of plant decommissioning and the long-term potential dose commitment from plant operations. These questions are discussed in Section 2.1.5.

2.1.5 DECOMMISSIONING AND DECONTAMINATION

The impact of decommissioning is difficult to assess because it depends on the method of decontamination, the methods used to predict the distributions of radiological source terms, assumptions regarding environmental pathways to man (e.g., resuspension in air, food chains, and migration of nuclides in the ground), and the model used to relate source terms to dose. For these reasons it should be noted that the information presented below is preliminary and subject to further refinement as more information and better models become available.

Reference 6 presents an analysis of these effects for a 10,000-MTU/yr fluorination and fractionation plant. Table 2-2, taken from Table 7.16 of that reference, lists the results of one calculation of source terms for the major long-lived nuclides. These source terms are developed for a base case (Case 1) in Reference 6 described as a facility with an effluent-treatment technique that "represents the minimum treatment necessary to operate the process." Improved treatment techniques might reduce the already small effects (see below).

In Tables 2-3 and 2-4 (Tables 7.17 and 7.18, respectively, of Reference 6), the source terms are converted to annual estimated doses to an average individual and to an assumed population within a 55-mile radius for a midwestern site, respectively. These data indicate that the long-term impacts of operating this type of plant are quite small, even considering the possible uncertainties in the results. Further improvements in treatment technology may also be available to reduce these small impacts even further if benefits are believed to outweigh the costs.

2.1.6 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

The research, development, and demonstration needs for converting yellowcake concentrates to uranium hexafluoride for alternative fuel cycles do not differ from the needs of the present LWR once-through system. In general, environmental and safety equipment is well developed.

The principal areas of radiological health and safety requiring a better understanding than now exists are the long-term impacts of plant operation and the impact of plant decommissioning. It is not expected that greater knowledge in this area will uncover any significant increase in effects. However, there is a need to reduce uncertainties in our current knowledge to better define future plant designs and operational tradeoffs. As stated earlier, this need is independent of the reactor and fuel-cycle system chosen for future development as far as the model uranium hexafluoride production plant is concerned.

Table 2-1. Summary of environmental considerations for a 10,000-MTU/yr uranium hexafluoride production plant

Natural resource use	
Land, acres	
Temporarily committed	2,800
Permanently committed	80
Water, 106 gal/yr	
Discharged to air	200
Discharged to water bodies	2,200
Total discharged	2,400
Fossil fuel	
Electrical energy, 10^3 MW-hr/yr	94
Equivalent coal, 10^3 MT/yr	34
Natural gas, 10^6 scf/yr	1,100
Effluents	
Chemicals, MT/yr	
Gases	
Sulfur oxides	1,600
Nitrogen oxides	560
Hydrocarbons	35.2
Carbon monoxide	11
Fluoride	6
Liquids	
Sulfur oxides	248
Nitrate	6
Fluoride	486
Chloride	12
Sodium and potassium	86
Nickel	2.4
Iron	2,000
Radiological, Ci/yr	
Gases--uranium	0.008
Liquids	
Uranium	2.42
Radium-226	0.187
Thorium-230	0.08
Solids (buried)	0
Other high-level solid waste	48
Thermal, 10^9 Btu/yr	1,000

Table 2-2. Long-lived radionuclides dispersed by terrestrial pathways during the 30-year life of a model fluorination-fractionation uranium hexafluoride plant[a]

Nuclide	Total airborne release during 30-year plant life (Ci)	Terrestrial activity concentration[b] (Ci/m^2)
Radium-226	4.1×10^{-2}	1.7×10^{-12}
Thorium-230	3.6×10^{-1}	1.5×10^{-11}
Uranium-234	3.3	1.3×10^{-10}
Uranium-235	8.0×10^{-2}	3.3×10^{-12}
Uranium-238	3.3	1.3×10^{-10}

[a] Data from Table 7.16 of Reference 6 for Case 1, a 10,000-MTU/yr plant. Aquatic releases are not included.

[b] Activity dispersed uniformly over an area of 2.461 x 10^{10} m^2 (i.e., within a 55-mile radius of the plant).

Table 2-3. Annual doses[a] received by the average individual after the model fluorination-fractionation uranium hexafluoride plant[b] closes until significant decay of radionuclides

Nuclide	Individual whole-body doses (mrem) per exposure mode				Adult organ doses (mrem) per exposure mode					
					Bone		Lung		Kidney	
	Contaminated ground	Inhalation	Ingestion	Total	Inhalation	Ingestion	Inhalation	Ingestion	Inhalation	Ingestion
Radium-226	1.9-6[c]	4.9-7	5.0-5	5.2-5	4.7-6	5.0-4	4.2-6	5.0-5	4.9-7	5.0-5
Thorium-230	3.2-5	1.5-5	1.7-6	4.9-5	5.7-4	6.1-5	7.0-5	1.7-6	8.2-5	1.7-5
Uranium-234	3.7-4	1.4-6	1.1-5	3.8-4	2.2-5	1.7-4	5.4-5	1.1-5	5.0-6	4.2-5
Uranium-235	2.0-4	2.8-8	2.5-7	2.0-4	4.7-7	3.9-6	1.2-6	2.5-7	1.2-7	9.1-7
Uranium-238	5.9-4	1.2-6	9.1-6	6.0-4	2.0-5	1.7-4	4.7-5	9.1-6	4.4-6	3.7-5
Total	1.2-3	1.8-5	7.2-5	1.3-3	6.2-4	9.0-4	1.8-4	7.2-5	9.2-5	1.5-4

[a] Doses after plant closes from radioactive materials dispersed in the terrestrial environment during 30-year operation of a model 10,000-MTU/yr fluorination-fractionation uranium hexafluoride plant, assuming a uniform distribution of the radioactive dusts within a 55-mile radius from the plant. In addition to these doses, there will be a long-term dose from radon to individuals living near the repository where the fluorination ash is buried.

[b] Data from Table 7.17 of Reference 6 for Case 1, a 10,000-MTU/yr fluorination-fractionation plant.

[c] $1.9\text{-}6 = 1.9 \times 10^{-6}$.

Table 2-4. Annual doses[a] received by the population after the model fluorination-fractionation uranium hexafluoride plant[b] closes until significant decay of radionuclides--midwestern site[c]

Nuclide	Population whole-body dose (man-rem)	Population organ doses (man-organ-rem)		
		Bone	Lung	Kidney
Radium-226	0.19	1.8	0.20	0.19
Thorium-230	0.18	2.4	0.37	0.47
Uranium-234	1.4	2.0	1.3	1.5
Uranium-235	0.72	0.74	0.73	0.72
Uranium-238	2.2	2.8	2.3	2.3
Total	4.7	9.7	4.9	5.2

[a] Doses after plant closes from radioactive materials dispersed in the terrestrial environment during 30-year operation of a model 10,000-MTU/yr fluorination-fractionation uranium hexafluoride plant, assuming uniform distribution of the radioactive dusts within a 55-mile radius of the plant. The doses are for the entire population within the 55-mile radius. In addition to these doses, there will be a long-term dose from radon to the population living near the repository where the fluorination ash is buried.

[b] Data from Table 7.18 of Reference 6 for a 10,000-MTU/yr fluorination-fractionation plant at a midwestern site.

[c] Population whole-body dose for a New Mexico site is about 2% of the dose at the midwestern site.

2.2 CONVERSION OF URANYL NITRATE FROM REPROCESSING TO URANIUM HEXAFLUORIDE

2.2.1 GENERAL DESCRIPTION

After nuclear fuel is chemically separated from waste products in a reprocessing plant, part or all of the recovered uranium may be converted to uranium hexafluoride for reenrichment by either gaseous diffusion or gas centrifugation. The plant takes the output from solvent-extraction steps in the reprocessing plant in the form of uranyl nitrate. It converts this material to uranium hexafluoride of the purity required for feed to the enrichment plant.

There are presently no reprocessing plants in the United States that are licensed to recover the residual uranium from spent LWR fuel. The need for this step in the fuel cycle does not presently exist. The need for converting uranyl nitrate to uranium hexafluoride will depend on future government policy on reprocessing, the form of product from the reprocessing plant, the economics of blending versus reenrichment, and the need for uranium with the enrichment typical of spent fuel relative to the quantity available from reprocessing.

The commercial reprocessing plants previously constructed were designed to produce either uranyl nitrate or uranium hexafluoride. One plant was to produce uranium hexafluoride directly by a fluoride volatization process, but the operability of the plant was never demonstrated. It is anticipated that most, if not all, future reprocessing capacity will produce uranyl nitrate as a product to be available for converting to uranium hexafluoride if desired.

For the purposes of this document, a model conversion plant is assumed to take purified uranyl nitrate from an adjacent reprocessing plant. The model conversion plant is assumed to process 1,500 MTU/yr. The fuel contains minute amounts of fission products and transuranium elements that have not been completely removed at the reprocessing plant. Illustrative flowsheets of this conversion plant are provided in Reference 7, especially Figures 3.2, 4.1, and 4.2. Although these flowsheets are not necessarily representative of a specific plant, they illustrate what is believed to be an operable facility capable of being licensed (Ref. 7).

2.2.2 SAFETY CONSIDERATIONS

The safety of converting uranyl nitrate to uranium hexafluoride in a plant separate from a reprocessing plant has not been evaluated and reported at present. It is expected, however that the potential for accidents and their consequences at such a plant will be similar to those already described for a plant converting yellowcake concentrates to uranium hexafluoride, especially those situations applicable to the wet process. The consequences of accidents can be estimated by referring to the presentation in Reference 3, Section C, or Reference 4, pp. IVF-38 and IVF-39. The results presented there may have to be adjusted to account for plant size differences, as appropriate.

On this basis the safety of such facilities should be comparable to that of the existing uranium hexafluoride production plants, which has been excellent to date.

2.2.3 ENVIRONMENTAL CONSIDERATIONS

The environmental considerations for the model plant are described in detail in Reference 7. It is assumed that the plant occupies the same site as a large chemical reprocessing plant. The material flows and potential nonradiological effluents are described in Chapter 4 of Reference 7 and are listed in Tables 4.1 through 4.5 of that document.

The radiological source terms and the estimated resulting doses, as developed in Reference 7, are presented in Tables 2-5 through 2-7 (Tables 4.9, 7.4, and 7.5 of Reference 7, respectively). It should be noted that the analysis has been done for two types of sites—one midwestern site and one coastal site--and for five different radwaste treatment cases (see Table 2-8).

The impacts presented here are for converting uranyl nitrate recovered from typical LWR spent fuel with average discharge burnups of 27,500 to 33,000 MWd/MTU. For other fuels or other fuel cycles, the results reported here may have to be modified. A key factor affecting the resulting impact is the decontamination achieved during reprocessing. To the extent that other fuels can be cleaned up by reprocessing to match the quality of feed to the model plant described in Reference 7, the results reported here for the model plant are applicable. The characteristics of this feed material are given in Section 4.2 of Reference 7.

2.2.4 LICENSING STATUS AND CONSIDERATIONS

With deferral of reprocessing there has been no urgency to assess the licensability of the associated uranyl nitrate to uranium hexafluoride conversion plant. It is reasonable to assume, however, that licensing of this plant should present no new questions when compared to a plant converting yellowcake to uranium hexafluoride.

As was pointed out above, a potential unresolved issue may be the long-term impact of operations and decommissioning of the conversion plant. This subject is discussed below.

2.2.5 DECOMMISSIONING AND DECONTAMINATION

Reference 7 has attempted to evaluate the long-term effects of plant operations after its closure. The analysis contained in Reference 7 is analogous to that presented in Reference 6 for a yellowcake to uranium hexafluoride conversion plant. The comments made previously in Section 2.1.5 regarding the degree of uncertainty apply here also.

Tables 2-9 through 2-12 present information developed in Reference 7. These data show that the impacts are expected to be quite small compared to natural background (compare with data presented in Sections 2.1.3 and 2.1.5).

2.2.6 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

Research, development, and demonstration relative to the model plant are needed primarily in the following areas:

1. Effect on plant design of feed from a reprocessing plant with characteristics differing from those of the reference feed
2. Improvement in the data base and in the methods used to study decommissioning and long-term radiological effects

These areas for research, development, and demonstration do not appear to present any major problems for the implementation of this step in the fuel cycle once a decision is made to allow the reprocessing of spent fuel from reactors.

Table 2-5. Source terms for the model 1,500-MTU/yr recycle uranium hexafluoride plant: calculated release of radioactive material in airborne effluents[a]

Nuclide	Case 1 Concentration (μCi/ml)	Case 1 Amount (Ci/yr)	Case 2 Concentration (μCi/ml)	Case 2 Amount (Ci/yr)	Case 3 Concentration (μCi/ml)	Case 3 Amount (Ci/yr)	Case 4 Concentration (μCi/ml)	Case 4 Amount (Ci/yr)	Case 5 Concentration (μCi/ml)	Case 5 Amount (Ci/yr)
Sr-90	1.1-14[b]	3.5-5	4.3-15	1.4-5	2.9-15	9.5-6	2.0-15	6.5-6	1.6-17	5.2-8
Zr-95	2.9-13	9.4-4	1.2-13	3.7-4	8.4-14	2.6-4	5.5-14	1.8-4	4.4-16	1.4-6
Nb-95	5.7-13	1.8-3	2.2-13	7.1-4	1.5-13	4.9-4	1.1-13	3.4-4	8.5-16	2.7-6
Tc-99	8.6-12	2.8-2	3.4-12	1.1-2	2.4-12	7.5-3	1.6-12	5.2-3	1.3-14	4.1-5
Ru-103	8.4-14	2.7-4	3.3-14	1.1-4	2.3-14	7.3-5	1.6-14	5.0-5	1.3-16	4.0-7
Ru-106	5.6-13	1.8-3	2.2-13	7.1-4	1.5-13	4.9-4	1.1-14	3.4-4	8.5-16	2.7-6
Cs-134	2.9-14	9.4-5	1.2-14	3.7-5	8.0-15	2.6-5	5.5-15	1.8-5	4.4-17	1.4-7
Cs-137	1.5-14	4.9-5	6.1-15	1.9-5	4.2-15	1.3-5	2.9-15	9.2-6	2.3-17	7.4-8
Ce-144	9.7-14	3.1-4	3.8-14	1.2-4	2.7-14	8.5-5	1.8-14	5.9-5	1.5-16	4.7-7
Eu-154	1.0-15	3.2-6	3.9-16	1.3-6	2.7-16	8.7-7	1.9-16	6.0-7	1.5-18	4.8-9
Th-234	1.2-12	3.9-3	4.8-13	1.5-3	3.3-13	1.1-3	2.3-13	7.4-4	1.8-15	5.9-6
U-232	2.2-13	7.0-4	8.7-14	2.8-4	6.0-14	1.9-4	4.1-14	1.3-4	3.3-16	1.1-6
U-233	9.2-16	2.9-6	3.6-16	1.2-6	2.5-16	8.0-7	1.7-16	5.5-7	1.4-18	4.4-9
U-234	1.5-11	4.7-2	5.8-12	1.8-2	4.0-12	1.3-2	2.8-12	8.8-3	2.2-14	7.0-5
U-235	2.4-13	7.7-4	9.5-14	3.0-4	6.6-14	2.1-4	4.5-14	1.5-4	3.6-16	1.2-6
U-236	5.5-12	1.8-2	2.2-12	7.0-3	1.5-12	4.8-3	1.0-12	3.3-3	8.3-15	2.6-5
U-237	1.4-11	4.5-2	5.5-12	1.8-2	3.8-12	1.2-2	2.6-12	8.4-3	2.1-14	6.7-5
U-238	4.9-12	1.6-2	1.9-12	6.2-3	1.3-12	4.3-3	9.2-13	2.9-3	7.3-15	2.3-5
Np-237	2.4-14	7.6-5	9.5-15	3.0-5	6.5-15	2.1-5	4.5-15	1.4-5	3.6-17	1.1-7
Np-239	4.9-14	1.6-4	1.9-14	6.1-5	1.3-14	4.2-5	9.2-15	2.9-5	7.3-17	2.3-7
Pu-238	6.5-14	2.1-4	2.6-14	8.2-5	1.8-14	5.6-5	1.2-14	3.9-5	9.7-17	3.1-7
Pu-239	5.1-15	1.6-5	2.0-15	6.4-6	1.4-15	4.4-6	9.5-16	3.0-6	7.6-18	2.4-8
Pu-240	7.4-15	2.4-5	2.9-15	9.3-6	2.0-15	6.5-6	1.4-15	4.5-6	1.1-17	3.5-8
Pu-241	1.6-12	5.0-3	6.3-13	2.0-3	4.3-13	1.4-3	3.0-13	9.5-4	2.4-15	7.6-6
Cm-244	3.7-16	1.2-6	1.5-16	4.7-7	1.0-16	3.2-7	7.0-17	2.2-7	5.5-19	1.8-9

[a] Data from Table 4.9 of Reference 7.

[b] 1.1-14 = 1.1 x 10^{-14}.

Table 2-6. Maximum doses[a] received by individuals[b] from airborne effluents from the model 1,500-MTU/yr recycle uranium hexafluoride plant[c]

Radwaste treatment case[d]	Maximum whole-body dose (mrem)	Maximum adult organ dose (mrem)									
		GI tract	Bone	Thyroid	Lungs	Muscle	Kidney	Liver	Spleen	Testes	Ovaries
						Midwestern site					
1	6.2-2[e]	9.1-2	5.4-1	6.1-2	2.8-1	5.8-2	1.3-1	6.3-2	4.9-2	5.6-2	4.8-2
2	2.5-2	3.7-2	2.2-1	2.5-2	1.2-1	2.4-2	5.4-2	2.6-2	2.0-2	2.3-2	2.0-2
3	1.8-2	2.6-2	1.6-1	1.8-2	8.2-2	1.7-2	3.8-2	1.8-2	1.4-2	1.6-2	1.4-2
4	1.2-2	1.8-2	1.1-1	1.2-2	5.7-2	1.2-2	2.6-2	1.3-2	9.8-3	1.1-2	9.7-3
5	9.6-5	1.4-4	8.6-4	9.7-5	4.5-4	9.2-5	2.1-4	1.0-4	7.7-5	8.8-5	7.6-5
						Coastal site					
1	5.5-2	8.4-2	4.7-1	5.5-2	2.4-1	5.3-2	1.2-1	5.6-2	4.4-2	5.0-2	4.3-2
2	2.2-2	3.3-2	1.9-1	2.2-2	9.4-2	2.1-2	4.6-2	2.2-2	1.7-2	2.0-2	1.7-2
3	1.5-2	2.3-2	1.3-1	1.5-2	6.5-2	1.5-2	3.2-2	1.5-2	1.2-2	1.4-2	1.2-2
4	1.0-2	1.6-2	8.8-2	1.0-2	4.4-2	9.8-3	2.2-2	1.0-2	8.2-3	9.4-3	8.0-3
5	8.4-5	1.3-4	7.3-4	8.5-5	3.7-4	8.1-5	1.8-4	8.6-5	6.7-5	7.7-5	6.6-5

[a] Fifty-year dose commitment from exposure to effluents from 1 year's operation of the model plant.

[b] Maximum dose to individuals at 1.5 miles and downwind of the prevailing wind direction; maximum doses at 0.5 and at 1.0 mile are 2.3 and 1.5 times higher. All food is assumed to be produced and consumed at the location of exposure. Daily intakes are 1 liter of milk, 0.25 kg of vegetables, and 0.3 kg of beef.

[c] Data from Table 7.4 of Reference 7.

[d] See Table 2-8 for a description of the radwaste treatment cases.

[e] $6.1\text{-}2 = 6.1 \times 10^{-2}$.

Table 2-7. Summary of annual doses[a] received by the population from airborne effluents from the model 1,500-MTU/yr recycle uranium hexafluoride plant[b]

Radwaste treatment case[c]	Whole-body dose (man-rem)	Population organ doses (man-organ-rem)									
		GI tract	Bone	Thyroid	Lungs	Muscle	Kidney	Liver	Spleen	Testes	Ovaries
						Midwestern site					
1	1.39	2.03	1.14+1	1.40	5.72	1.33	2.83	1.39	1.10	1.27	1.08
2	5.51-1[d]	8.04-1	4.52	5.54-1	2.27	5.27-1	1.12	5.50-1	4.35-1	5.02-1	4.28-1
3	3.81-1	5.56-1	3.13	3.84-1	1.57	3.65-1	7.76-1	3.81-1	3.01-1	3.48-1	2.96-1
4	2.65-1	3.86-1	2.17	2.66-1	1.09	2.53-1	5.39-1	2.64-1	2.09-1	2.41-1	2.06-1
5	2.09-3	3.04-3	1.71-2	2.10-3	8.59-3	2.00-3	4.25-3	2.08-3	1.65-3	1.90-3	1.62-3
						Coastal site					
1	7.53-1	1.08	6.38	7.57-1	3.34	7.20-1	1.56	7.67-1	5.97-1	6.87-1	5.87-1
2	2.97-1	4.26-1	2.51	2.98-1	1.31	2.84-1	6.12-1	3.01-1	2.35-1	2.71-1	2.31-1
3	2.04-1	2.94-1	1.72	2.06-1	8.95-1	1.96-1	4.22-1	2.07-1	1.62-1	1.86-1	1.59-1
4	1.40-1	2.02-1	1.18	1.41-1	6.07-1	1.34-1	2.88-1	1.42-1	1.11-1	1.28-1	1.09-1
5	1.13-3	1.62-3	9.51-3	1.13-3	4.96-3	1.08-3	2.32-3	1.14-3	8.92-4	1.03-3	8.77-4

[a] Fifty-year dose commitment from exposure to effluents from 1 year's operation of the model plant. The doses are for the entire population within 55 miles of the model plant; daily intake assumed to be 0.25 kg of vegetables, 0.3 kg of beef, and 300 ml of milk. It is assumed that 100% of the food consumed is produced or grown at the location of exposure.

[b] Data from Table 7.5 of Reference 7.

[c] See Table 2-8 for a description of the radwaste treatment cases.

[d] $5.51\text{-}1 = 5.51 \times 10^{-1}$.

Table 2-8. Conceptual waste-treatment case studies for the model recycle uranium-UF_6 plant[a]

Parameter	Case 1	Case 2	Case 3	Case 4	Case 5
Level of waste treatment	Meets present licensing regulations	Similar to recently completed plant	Near limits of present technology	Uses technology which is not fully developed	Applies technology to large gas flows
Airborne-effluent treatment systems					
Particulates	Porous metal filter, centrifugal separator bag filter, HEPA filter	Case 1 plus additional bag filter	Same as case 2	Case 2 plus HF-resistant HEPA filter, venturi scrubber	Case 4 plus bag filter on ventilation system
Gases	Cold traps for UF_6, condenser for H_2O and $HNO_3(NO_x)$	Same as case 1	Case 1 plus KOH coke-packed tower	Same as case 3	Same as case 3
	Burner for H_2 and H_2S	Same as case 1	Same as case 1	Case 1 plus KOH venturi scrubber	Same as case 4
	Scrubber systems for HF and F_2	Same as case 1	Case 1 plus KOH coke-packed tower	Case 3 plus KOH venturi scrubber	Same as case 4
Liquid-effluent treatment systems					
Nonradioactive	Monitored and released	Same as case 1	Same as case 1	Same as case 1	Same as case 1
Radioactive	Concentrated in evaporators and treated as solid radwaste	Same as case 1	Same as case 1	Same as case 1	Same as case 1
Solid-radwaste treatment systems	All liquids evaporated to dryness, evaporator bottoms and other solids packaged in drums for disposal	Same as case 1	Same as case 1	Same as case 1	Incorporated in cement and packaged in drums for disposal (case 5b)

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[a] From Ref. 7 (Table 1.1).

Table 2-9. Radioactivity and ground surface concentration of long-lived radionuclides released during the 30-year life of the model 1,500-MTU/yr recycle uranium hexafluoride plant

Nuclide	Release during 30-year life of plant[a] (Ci)	Surface concentration[b] (Ci/m^2)
Technetium-99	8.3×10^{-1}	3.4×10^{-11}
Uranium-232	2.1×10^{-2}	8.5×10^{-13}
Uranium-234	1.4	5.7×10^{-11}
Uranium-235	2.3×10^{-2}	9.3×10^{-13}
Uranium-236	5.3×10^{-1}	2.2×10^{-11}
Uranium-238	4.7×10^{-1}	1.9×10^{-11}
Neptunium-237	2.3×10^{-3}	9.3×10^{-14}
Plutonium-238	6.2×10^{-3}	2.5×10^{-13}
Plutonium-239	4.8×10^{-4}	2.0×10^{-14}
Plutonium-240	7.1×10^{-4}	2.9×10^{-14}

[a] Data from Reference 7 for radwaste-treatment case 1, midwestern site. (See Table 2-8). These values are divided by the area within a 55-mile radius (2.46×10^{10} m^2) of the plant to give an assumed deposition rate.

[b] Deposition is assumed to occur uniformly out to a distance of 55 miles.

Table 2-10. Contribution of radionuclides and exposure modes from contaminated ground to the annual whole-body dose[a] received by individuals from model plant shutdown until significant decay of all radionuclides[b]

Nuclide	Whole-body dose per exposure mode (mrem) Contaminated ground	Inhalation	Ingestion	Total
Technetium-99	0	1.2×10^{-11}	1.7×10^{-7}	1.7×10^{-7}
Uranium-232	2.5×10^{-6}	4.5×10^{-8}	3.4×10^{-7}	2.9×10^{-6}
Uranium-234	1.5×10^{-4}	5.5×10^{-7}	4.1×10^{-6}	1.5×10^{-4}
Uranium-235	3.3×10^{-5}	8.2×10^{-9}	6.2×10^{-8}	3.3×10^{-5}
Uranium-236	5.3×10^{-5}	2.0×10^{-7}	1.5×10^{-6}	5.5×10^{-5}
Uranium-238	4.0×10^{-7}	1.6×10^{-7}	1.2×10^{-6}	1.8×10^{-6}
Neptunium-237	1.5×10^{-6}	9.4×10^{-8}	7.2×10^{-9}	1.6×10^{-6}
Plutonium-238	6.5×10^{-7}	2.6×10^{-7}	2.0×10^{-9}	9.1×10^{-7}
Plutonium-239	2.0×10^{-8}	2.3×10^{-8}	1.8×10^{-10}	4.3×10^{-8}
Plutonium-240	6.5×10^{-8}	3.4×10^{-8}	2.7×10^{-10}	9.9×10^{-8}
Total	2.4×10^{-4}	1.4×10^{-6}	7.4×10^{-6}	2.5×10^{-4}

[a] Dose is the average whole-body dose to individuals out to a distance of 55 miles from a model 1,500-MTU/yr recycle uranium hexafluoride plant. Data from Reference 7.

[b] A 30-year lifetime for the plant is assumed.

Table 2-11. Annual organ doses[a] received by individuals from plant shutdown until significant decay of all radionuclides: contribution of long-lived radionuclides deposited on the ground during the operation of the model plant

	Organ dose (mrem) per exposure mode[b]					
	GI tract		Bone		Kidney	
Nuclide	Inhalation	Ingestion	Inhalation	Ingestion	Inhalation	Ingestion
Technetium-99	3.8-12[c]	3.3-5	3.1-11	4.2-7	5.7-10	7.8-6
Uranium-232	2.6-10	7.7-8	6.5-7	4.7-6	7.0-8	5.2-7
Uranium-234	1.5-8	5.2-6	8.8-6	6.6-5	2.1-6	1.5-5
Uranium-235	2.9-10	8.4-8	1.5-7	1.0-6	3.2-8	2.4-7
Uranium-236	5.7-9	1.9-6	3.3-6	1.8-5	7.8-7	5.8-6
Uranium-238	4.9-9	1.3-6	2.7-6	2.0-5	6.1-7	4.5-6
Neptunium-237	2.9-11	8.4-9	2.3-6	1.8-7	7.0-7	5.4-8
Plutonium-238	7.8-11	7.7-9	1.0-5	8.2-8	1.1-6	8.7-9
Plutonium-239	6.2-12	6.2-10	9.6-7	7.6-9	9.9-8	7.7-10
Plutonium-240	9.1-12	9.0-10	1.4-6	1.1-8	1.4-7	1.1-9
Total	2.6-8	4.2-5	3.0-5	1.1-4	5.6-6	3.4-5

[a] The dose is the average individual dose out to a distance of 55 miles from the plant.

[b] Data from Reference 7 for the model 1,500-MTU/yr recycle uranium hexafluoride plant. An operating lifetime of 30 years is assumed.

[c] 3.8-12 x 3.8 x 10^{-12}.

Table 2-12. Annual doses received by the population from plant shutdown until significant decay of all radionuclides: contribution of long-lived radionuclides deposited on the ground during operation of the model plant

Nuclide	Dose[b] (man-rem or man-organ-rem per 3.6×10^6 persons[c])			
	Whole body	GI tract	Bone	Kidney
Technetium-99	6.1×10^{-4}	1.2×10^{-1}	1.5×10^{-3}	2.8×10^{-2}
Uranium-232	1.0×10^{-2}	2.8×10^{-4}	1.9×10^{-2}	2.1×10^{-3}
Uranium-234	5.4×10^{-1}	1.9×10^{-2}	2.7×10^{-1}	6.2×10^{-2}
Uranium-235	1.2×10^{-1}	3.0×10^{-4}	4.1×10^{-3}	9.8×10^{-4}
Uranium-236	2.0×10^{-1}	6.9×10^{-3}	7.8×10^{-2}	2.4×10^{-2}
Uranium-238	6.5×10^{-3}	4.7×10^{-3}	8.2×10^{-2}	1.8×10^{-2}
Neptunium-237	5.8×10^{-3}	3.0×10^{-5}	8.9×10^{-3}	2.7×10^{-3}
Plutonium-238	3.3×10^{-3}	2.8×10^{-5}	3.6×10^{-2}	4.0×10^{-3}
Plutonium-239	1.5×10^{-4}	2.2×10^{-8}	3.5×10^{-3}	3.6×10^{-4}
Plutonium-240	3.6×10^{-4}	3.3×10^{-6}	5.1×10^{-3}	5.1×10^{-4}
Total	8.9×10^{-1}	1.5×10^{-1}	5.1×10^{-1}	1.4×10^{-1}

[a] Dose to the population is the sum of the individual doses out to a distance of 55 miles from the plant.

[b] Data from Reference 7 for the model 1,500-MTU/yr recycle uranium hexafluoride plant. An operating lifetime of 30 years is assumed.

[c] Actual population within a 55-mile radius of the midwestern plant site.

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3. U.S. Atomic Energy Commission, Environmental Survey of the Uranium Fuel Cycle, WASH-1288, April 1974.
4. U.S. Nuclear Regulatory Commission, Final Generic Environmental Statement on the Use of Recycle Plutonium in Mixed Oxide Fuel in Light Water Reactors, NUREG-0002, Vol. 3, August 1976.
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6. U.S. Nuclear Regulatory Commission, Correlation of Radioactive Waste Treatment Costs and the Environmental Impact of Waste Effluents in the Nuclear Fuel Cycle--Conversion of Yellowcake to Uranium Hexafluoride, Part 1, "The Fluorination-Fractionation Process," ORNL/NUREG/TM-7, Oak Ridge National Laboratory, September 1977.
7. Correlation of Radioactive Waste Treatment Costs and the Environmental Impact of Waste Effluents in the Nuclear Fuel Cycle--Conversion of Recycle Uranium to UF_6, ORNL/NUREG/TM-37, Oak Ridge National Laboratory, August 1977.

Chapter 3

ENRICHMENT

3.1 GASEOUS DIFFUSION

3.1.1 GENERAL DESCRIPTION

Natural uranium contains about 0.7% of the fissionable isotope uranium-235. Enrichment of the uranium to 2 to 4% of this isotope is necessary to provide fuel for a nuclear reactor moderated with light water. Currently, the only process used commercially in the United States to produce enriched uranium by isotope separation is based on gaseous diffusion (Ref. 1).

Enrichment by gaseous diffusion depends on two facts:

1. The average velocity of a gas molecule at a given temperature depends on its mass.
2. The rate at which gas molecules escape through a small hole is proportional to their speed.

The gaseous diffusion process uses a porous barrier to provide the holes through which the gas molecules diffuse. For gaseous uranium hexafluoride, the theoretical maximum enrichment in uranium-235 for a single stage is by a factor of 1.0043. The enrichment is increased by using a large number of stages, known as a cascade. The existing plants utilize about 1,200 stages to produce material containing 4% uranium-235 in uranium-238.

The gas flow through the cascade is provided by compressors driven by electric motors. The compression of gas generates heat, which is discharged into the environment. In addition to the enriched uranium hexafluoride product, the gaseous diffusion plants produce uranium hexafluoride depleted in uranium-235 (tails). The tails assay at which an enrichment plant is operated depends on the availability of uranium feed, the enrichment-plant capacity, and the cost of electrical power. After current improvement programs are completed, the existing enrichment plants are expected to operate at a tails assay of 0.2 to 0.3% uranium-235 to sustain the nuclear power plants that have contracted for enrichment services on a long-term basis. The tails are stored at the plants as solid uranium hexafluoride in cylinders for possible future uses.

At present, all of the enrichment services for the U.S. nuclear industry are provided by three government-owned and contractor-operated gaseous diffusion plants. The plants at Oak Ridge, Tennessee, and at Paducah, Kentucky, are operated by the Union Carbide Corporation, Nuclear Division; the plant at Portsmouth, Ohio, is operated by the Goodyear Atomic Corporation.

The Oak Ridge and Portsmouth plants were built on sites originally chosen for their remote location, low population density, and the availability of cheap electrical power. The Paducah plant was built on a government-owned site previously occupied by an ordnance works. Population density in the vicinity of the plants ranges from 30 to 40 persons per square mile, the region within a 50-mile radius of the Oak Ridge plant having the highest population density, primarily because of the proximity of Knoxville, Tennessee. Cooling water is obtained from the Clinch River at Oak Ridge, from the Ohio River at Paducah, and from subterranean wells at Portsmouth. Most of the process heat, however, is discharged to the atmosphere via forced-draft cooling towers. Makeup water is required to replace cooling-tower evaporation, windage, and blowdown losses. (Blowdown is the removal of a portion of the recirculating cooling water to prevent the buildup of dissolved salts in the system.)

Power for the gaseous diffusion plants is drawn from the grids of three utilities: Tennessee Valley Authority (TVA), Ohio Valley Electrical Corporation, and Electric Energy, Inc. These utilities generate power primarily in large coal-fired stations; TVA is rapidly increasing its power production from nuclear-fueled stations.

The existing plants were built between 1943 and 1955 to supply enriched uranium for military use. When this need declined, the output of the three plants was reduced substantially. It reached a low (measured as power demand) of 1,900 MW by 1970. At a production level of 10.5 million separative-work units (MSWU) per year in 1972, the electrical power load was about 3,250 MW. At full capacity, the three-plant complex requires about 6,100 MW of electrical power while operating at a production level of 17.2 MSWU/yr.

The Federal Government is currently modifying and uprating the three gaseous diffusion plants to increase their capacity and supply the nuclear power industry. It is estimated that the program will cost $1.5 billion (1978 dollars) and yield an increase of about 60% in productive capacity. An annual capacity of 27 MSWU is expected, with full reactivation by government fiscal year 1985 (Ref. 2). This capacity increase will result from two currently authorized programs: the Cascade Improvement Program (CIP) and the Cascade Uprating Program (CUP); it will be accompanied by an increase in electrical power requirements from 6,100 MW, the maximum power demand of the original plants, to 7,380 MW.

Once the CIP and CUP are complete, the existing gaseous diffusion plants will be essentially at a practical limit of technology, in both design and operation. The next increment of enrichment capacity in the United States will use the gas centrifuge. This technology is discussed in Section 3.2.

This subsection is concerned with the characteristics and potential impacts of the existing diffusion complex only.

The model plant has a capacity of 8.75 MSWU/yr and requires 2,400 MW of continuous electrical power. This plant is identical with that described as the reference plant in Reference 3.

3.1.2 SAFETY CONSIDERATIONS

Potential accidents at a gaseous diffusion plant and their consequences are discussed in Reference 3, which describes the following accidents and their consequences:

1. Criticality accidents
2. Noncriticality accidents involving radioactive materials
3. Other accidental releases (primarily chemical)
    a. Hydrogen fluoride
    b. Nitric acid
    c. Coolant
    d. Oils
    e. Refrigerants

Certain activities related to safety are now under way in the U.S. Department of Energy (DOE) (Ref. 4). These are related primarily to

1. Evaluation of uranium hexafluoride containment needs in light of increased production rates and higher gas pressures as a result of the cascade improvement and uprating programs
2. Evaluation of the causes and consequences of uranium hexafluoride handling accidents
3. Evaluation of the contamination of interior surfaces of the equipment
4. Evaluation of the risk of criticality resulting from the addition of freezer-sublimer uranium hexafluoride storage equipment being installed to permit faster response to power changes and cascade operating conditions

On the basis of the excellent safety record of the gaseous diffusion plants to date, the risk to the public resulting from their expansion to 27 MSWU/yr and continued operation is very small. Furthermore, the selection of an alternative fuel cycle will have no effect on the model plant operation so long as feed-material specifications remain unaffected by this choice.

3.1.3 ENVIRONMENTAL CONSIDERATIONS

The environmental considerations for an 8.75-MSWU/yr diffusion plant are well documented in Reference 3, Sections 2.3.3 and 3.3. Table 3-1 collects and summarizes information from Reference 3 in a format similar to that of Reference 5. The radiological source terms reflect the return of uranium from spent LWR fuel for recycle (Ref. 3). Table 3-2 provides information on the nonradiological liquid effluents from the plant. It is a reproduction of Table 2.3-42 of Reference 3.

The major environmental impacts of gaseous diffusion plant operation are the heat it rejects to the environment and the 2,400-MWe power demand. Although there appears to be no need for additional gaseous diffusion plants to meet future needs, such expansion appears to be environmentally acceptable (Ref. 6).

3.1.4 LICENSING STATUS AND CONSIDERATIONS

All present gaseous diffusion plants are owned by the Federal Government and are not subject to regulatory review and licensing by the U.S. Nuclear Regulatory Commission (NRC). Private ownership of the plants is not presently envisioned; therefore, Federal nuclear licensing considerations are not applicable to this type of plant. Water quality, air quality, and occupational health and safety regulations, however, must be met.

The DOE has produced a variety of environmental statements and development plans (Refs. 3, 4, 6) that discuss issues of public health and safety. On the basis of these documents and the DOE internal environmental-protection procedures, there is a high degree of confidence that issues of public health and safety will be adequately addressed in new and emerging technologies.

3.1.5 DECOMMISSIONING AND DECONTAMINATION

The impact of the decommissioning of gaseous diffusion plants is best evaluated by defining a specific retirement scenario and analyzing its various steps for impacts on public health and safety. Impacts would occur both at the time of retirement and possibly for several years afterward. The magnitude of the impact depends on the extent of decontamination, the procedure used for decommissioning, the handling of residual radioactivity, and other factors. Section 6.3 of Reference 6, referring to the retirement of plants, states that "sufficient experience in equipment retirement is already available from the ERDA uranium enrichment program. This experience indicates that the decommissioning of large enrichment plants will not introduce any significant new or unknown technical, safety, or environmental problems that differ substantially from those that may occur during operation and maintenance."

Reference 6 goes on to say that the "NRC procedures or more rigorous procedures, will be enforced during the dismantling of the plant . . . . If the plant were to be restored to its original condition, the present worth of the future costs involved is estimated to between 1 and 2% of the original construction cost. Thus, the retirement costs would not affect either the overall program cost-benefit analyses or the cost-benefit analyses for a given enrichment plant." Reference 6 considers mainly the impact of a Portsmouth add-on, but the conclusion is also presented in Section 6.3 of Reference 3 and is applicable to a stand-alone facility as well.

One area not specifically addressed in previous discussions of decommissioning is the fate of tails material held on site and resulting from many years of plant operation. Although radon and other isotopes may build up because of radioactive decay of the storage-cylinder contents, their concentrations will probably be small because of long half-lives of the isotopes at the head of the chain (Ref. 7). It is likely that the major impact resulting from tails storage is most likely to be the requirement for land for long-term storage. If the tails are never to be used, a requirement for disposal will also arise.

In conclusion, the impact of decommissioning an enrichment plant appears to be small.

3.1.6 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

Reference 4 discusses the issues and requirements related to environmental safety and health. Those concerned with gaseous diffusion technology include the following:

1. Meeting the requirements of the Federal Water Pollution Control Act
2. Potential for reduction in Freon releases
3. Improved handling of buried solids
4. Improved protection against accidental releases of uranium hexafluoride
5. Contaminant control, especially if recycle uranium is to be returned in the future for reenrichment
6. Continued attention to cascade criticality possibilities
7. Potential control of in-plant exposure to nickel
8. Continued optimization of tails assay
9. Preferred uses of waste heat
10. Studies of recycle potential of contaminated metals

In addition, there is a need to better define decommissioning criteria for enrichment plants, including the handling of tails material.

Table 3-1. Summary of environmental considerations for an 8.75-MSWU/yr gaseous diffusion plant (GDP)

Natural resource use[a]	
Land, acres	
Temporarily committed	750
Permanently committed	2
Water, 10^6 gal/yr	
Discharged to air	6,234
Discharged to water bodies	1,666
Total discharged	7,900
Fossil fuel	
Electrical energy, 10^3 MW-hr/yr	21,020
Equivalent coal, 10^3 MT/yr	Not available
Coal, 10^3 MT/yr	52.5
Gasoline and diesel fuel, 10^3 gal/yr	180

Effluents		
Chemical, MT/yr		
Gases[b]	GDP	Steam plant
Hydrogen fluoride	2.4	--
Nitrogen oxides	1.1	525
Sulfur dioxide	46.10^{-3}	656
Particulates	0.4	31.5
Carbon monoxide	--	10.5
Hydrocarbons	--	5.3
Liquids		See Table 3-2
Radiological, Ci/yr[c]		
Gases[d]		
Uranium		0.048
Technetium-99		0.54
Other		0.01
Liquids[e]		
Uranium		0.0028
Technetium-99		7.0
Other		0.12
Solids (buried)[f]		
Uranium		0
Technetium-99		62
Other		1.58
Other high-level solids		Not available
Thermal (heat to stack), 10^9 Btu/yr		350[b]

[a] Table 2.2-3 of Reference 3.

[b] Table 2.3-41 of Reference 3.

[c] Based on reactor return of recycle uranium (see discussion in Section 2.3.1.17 of Ref. 3). Without recycle, figures are close to those shown for "uranium" only (see Table IV F-16 of Ref. 1, for example).

[d] Table 2.3-11 of Reference 3.

[e] Table 2.3-40 of Reference 3.

[f] Table 2.3-13 of Reference 3.

Table 3-2. Liquid effluents received by and discharged from the primary holding pond of the gaseous diffusion enrichment plant

Constituent	Decontamination and uranium recovery	Cooling-tower blowdown	Water-treatment backwash	Steam plant	Discharge to receiving water
Flow, liters/day	4,500	16,300,000	303,000	512,000	17,100,000
Leading constituents, mg/liter					
Nitrate[a]	20,300	--	--	--	5.3
Aluminum	136	--	--	--	0.04
Fluorine	250	--	--	--	0.07
Uranium	1.9	--	--	--	0.0005
Nitric acid	19,700	--	--	--	5.2
Aluminum nitrate	1,080	--	--	--	0.28
Tributyl phosphate	38	--	--	--	0.01
Petroleum solvent	114	--	--	--	0.03
Phosphate	--	1.0	--	13.3[b]	0.70
Chlorine	--	0.3	--	(b)	0.285
Chromium	--	0.05	--	--	0.047
Zinc	--	0.5	--	--	0.475
Sulfate	--	(c)	(c)	250	(d)
Chloride	--	(c)	(c)	110	(d)
Suspended solids, mg/liter	(c)	(c)	(c)	140	(d)
Dissolved solids, mg/liter	(c)	(c)	(c)	500	(d)
pH	Neutralized	Neutralized	7	Neutralized	Neutralized
Temperature, °F	--	85	--	(c)	(b)

[a] Total equivalent values; composed of aluminum nitrate and nitric acid.

[b] Based on a concentration of 30 ppm PO_4 in the boiler blowdown. The blowdown is mixed with waste streams from the regeneration of Zeolite units to obtain the total liquid effluent from the steam plant.

[c] Undetermined.

[d] It is anticipated that Federal, State, and local guidelines and regulations for effluents and receiving waters will be met.

3.2 GAS CENTRIFUGE

3.2.1 GENERAL DESCRIPTION

A model centrifuge plant has been described (Ref. 3) as follows:

> A hypothetical gas centrifuge enrichment facility will consist of many centrifuge machines operated in a cascade. For a full-size facility, numerous cascades will be operated in parallel in eight 425-ft-wide-by-650-ft-long process buildings. The process buildings will be equipped with feed stations and product and tails withdrawal facilities to maintain the required uranium hexafluoride flow into and out of the cascade system. A recycle/assembly building will be located onsite to assemble centrifuge machine subassemblies, repair disabled machines, and provide for decontamination and scrap processing. Process heating and space heating requirements will be supplied by a coal-fired steam plant. Miscellaneous support facilities will be located on the centrifuge plant site. Cooling water for the centrifuge units will be provided by a recirculating cooling water system. A routine discharge from this system is required to prevent an unacceptable buildup of dissolved solids. Sanitary water will be supplied and sewage from the centrifuge facility will be treated in a secondary treatment facility. Two holding ponds will be provided to allow settling and equalization of effluent streams and to provide the capability to contain accidental releases. Another pond will be required to settle fly ash and sulfur dioxide scrubber sludge produced at the steam plant and to provide neutralization of coal pile runoff. Approximately 11 acres through the year 2000 is estimated to be required for burial of contaminated scrap.

The model plant has an annual capacity of 8.75 MSWU/yr and requires about 105 MW of electrical power for steady-state operation (Ref. 6).

Detailed plant and process descriptions can be found in References 3 and 6. Section 2.3 of Reference 4 provides a summary of this information and discusses the key environmental, safety, and health issues of centrifuge technology.

The primary sources of data for this section are References 3, 4, and 6, with definition of the model plant based in most part on the second document, which incorporates the first document by reference.

3.2.2 SAFETY CONSIDERATIONS[a]

Potential accidents in feed, withdrawal, and uranium-recovery facilities for a centrifuge plant are the same as those described for the gaseous diffusion plant.

Differences in accident potential and consequences lie primarily in lower in-process inventories and quantities of required oil, chemicals, and chromated recirculating cooling water for the centrifuge plant. Criticality potential is lower because of the lower inventory, whereas the loss of uranium hexafluoride from a centrifuge is improbable as well as inconsequential because of the very low machine inventory. Seismic and wind-loading criteria for the centrifuge process buildings are important mainly for operation rather than environmental protection.

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[a] Data from Reference 6.

Because many vacuum pumps are used, there is considerable oil at the site; however, it is packed in quantities of 2.5 to 45 liters. Thus, the potential for accidental loss is extremely low. Increased use of nitric acid in uranium-recovery operations would increase loss potential somewhat, but the acid would be easily contained, for example, by the use of dikes. Losses of recirculating cooling water due to line breaks would have reduced consequences because smaller lines are used than in the expanded gaseous diffusion plants. In any event, the impact on aquatic life from accidental loss of this chromated water would likely be minimal because of the short exposure times involved.

3.2.3 ENVIRONMENTAL CONSIDERATIONS

There is little difference between the environmental impacts of constructing a gaseous diffusion and a gas-centrifuge plant except that the former requires concurrent construction of large electrical power source(s) and its redundant backup. The major differences during operation are those related to the heat-removal system and the power consumed. Both the heat-removal and power requirements are higher for a gaseous diffusion plant than for a gas-centrifuge plant of similar capacity. The environmental costs are thus correspondingly greater for a gaseous diffusion plant; however, the DOE staff has shown that the environmental effects of the add-on gaseous diffusion plant are acceptable (Ref. 6). The impact of a gas-centrifuge plant, therefore, should also be acceptable.

Tables 3-3 and 3-4 present a summary of environmental considerations and data on liquid effluents, respectively. Note that the impact of the rotor-fabrication plant, while considered in Reference 3, is not considered here.

3.2.4 LICENSING STATUS AND CONSIDERATIONS

The comments concerning the gaseous diffusion plants apply to centrifuge plants as well (see Section 3.1.4).

3.2.5 DECOMMISSIONING AND DECONTAMINATION

Section 6.3 of Reference 6 discusses the retirement of centrifuge enrichment plants. This discussion is essentially the same as that given in Section 6.3 of Reference 3. The discussion previously presented for gaseous diffusion therefore applies to centrifuge plants as well.

3.2.6 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

Reference 4 discusses the issues and requirements related to environment, safety, and health. Those concerned with gas-centrifuge technology include the following:

1. Meeting the requirements of the Federal Water Pollution Control Administration
2. Potential for reduction in Freon releases
3. Improved handling of buried solids
4. Improved protection against accidental releases of uranium hexafluoride
5. Contaminant control, especially if recycle uranium is to be returned in the future for reenrichment
6. Control of solvents used in the manufacturing of centrifuge components
7. Tests to insure the protection of personnel in the event of an essentially instantaneous disintegration of a rotor
8. Improved reduction in noise levels

9. Continued attention to cascade criticality possibilities
10. Evaluation of trichloroethylene substitutes
11. Continued optimization of tails assay
12. Socioeconomic impact analyses of the Portsmouth area (for first centrifuge plant)

In addition, there is a need for better definition of decommissioning criteria applicable to enrichment plants, including criteria for handling of the tails material.

Table 3-3. Summary of environmental considerations for an 8.75-MSWU/yr gas-centrifuge enrichment plant (GCEP)

Natural resource use		
Land, acres		
Temporarily committed		700
Permanently committed		12
Water,[a] 10^6 gal/yr		
Discharged to air		402
Discharged to water bodies		511
Total discharged		913
Fossil fuel		
Electrical energy,[a] 10^3 MW-hr/yr		920
Equivalent coal, 10^3 MT/yr		Not available
Coal, 10^3 MT/yr[b]		52.5
Gasoline and diesel fuel, 10^3 gal/yr[b]		180
Effluents		
Chemicals, MT/yr		
Gases	GCEP[c]	Steam plant[d]
Hydrogen fluoride	0.5	--
Nitrogen oxides	2	525
Sulfur dioxide	0.1	656
Particulates	0.2	31.5
Carbon monoxide	--	10.5
Hydrocarbons	--	5.3
Ethanol	11[d]	--
Liquids		See Table 3-4
Radiological,[e] Ci/yr		
Gases[f]		
Uranium		0.048
Technetium-99		0.54
Other		0.01
Liquids[g]		
Uranium		0.0053
Technetium-99		7.0
Other		0.11
Solids (buried)		0
Other high-level wastes		Not available
Thermal (heat to stack), 10^9 Btu/yr[d]		350

[a] Reference 6, p. 5-9.

[b] Table 2.2-1 of Ref. 3.

[c] Table 5.1-6 of Ref. 6.

[d] Table 2.3-17 of Ref. 3.

[e] Based on reactor returns of recycle uranium. (See discussion in Section 2.3.1.17 of Ref. 3). Without recycle, figures are close to those shown for "uranium" only (see Table IV F-16 of Ref. 1, for example).

[f] Table 2.3-11 of Ref. 3.

[g] Table 2.3-12 of Ref. 3.

Table 3-4. Liquid effluents received by and discharged from the gas centrifuge enrichment plant primary holding pond

	Decontamination and uranium recovery	Cooling-tower blowdown	Water-treatment backwash	Steam plant	Discharge to receiving water
Flow,[a] liters/day	9,350	1,840,000	303,000	512,000	2,880,000
Leading constituent, mg/liter					
Nitrate[b]	41,000	--	--	--	130
Aluminum	3.07	--	--	--	1.0[c]
Fluorine	266	--	--	--	0.9
Uranium	1.53	--	--	--	0.005
Nitric acid	20,260	--	--	--	83.4
Aluminum nitrate	24,200	--	--	--	8
Tributyl phosphate	30.6	--	--	--	0.2
Petroleum solvent	91.7	--	--	--	0.3
Phosphate	--	1.0	--	13.3[d]	2.8
Chlorine	--	0.3	--	(e)	(f)
Chromium	--	0.05	--	--	0.032
Zinc	--	0.5	--	--	0.32
Sulfate	--	(e)	(e)	~250	(f)
Chloride	--	(e)	(e)	~110	(f)
Suspended solids, mg/liter	(e)	(e)	(e)	~140	(f)
Dissolved solids, mg/liter	(e)	(e)	(e)	~900	(f)
pH	Neutralized	Neutralized	7	Neutralized	Neutralized
Temperature, °F	--	85	--	(e)	(f)

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[a] Total flow is rounded off to 760,000 gpd (2,800,000 liters/day) to account for items present in undetermined quantities, such as laboratories, roof and floor drains from process buildings, and miscellaneous items. Concentrations are based on the total dilution of the 760,000 gpd.

[b] These represent total equivalent values and are composed of the aluminum nitrate and nitric acid in the effluent.

[c] Assuming 90% Al^{3+} settled out as hydroxides of aluminum.

[d] Based on a concentration of 30 ppm PO_4 in the boiler blowdown. The blowdown is mixed with waste streams from the regeneration of Zeolite units to obtain the total liquid effluent from the steam plant.

[e] Undetermined.

[f] It is anticipated that Federal, State, and local guidelines and regulations for effluents and receiving waters will be met.

3.3 ADVANCED PROCESSES

3.3.1 GENERAL DESCRIPTION

Two enrichment schemes are under study as potential competitors of the gaseous diffusion and gas-centrifuge processes. One uses selective photoexcitation; the other uses differential diffusion in a jet stream. Photoexcitation processes are being studied that employ the interaction of laser light with uranium in the molecular or atomic state. The jet-diffusion techniques usually employ uranium-containing gas in a lighter gas diluent. The flow expands through a nozzle and is divided by a knife edge. Summary discussions of these processes can be found in various sources (Refs. 3 and 8).

Other processes that have been or are being studied include phase-equilibrium processes such as fractional distillation or gas-liquid absorption, chemical exchange processes such as gas-liquid chemical exchange, ion-exchange chromatography, diffusion processes such as thermal diffusion or sweep diffusion, aerodynamic processes such as the vortex-tube and the Fenn-shock processes, molecular-flow processes, and nuclear-spin processes. None appears likely to become economically competitive with either the gaseous diffusion or the gas-centrifuge processes.

3.3.2 SAFETY CONSIDERATIONS

The technology of advanced isotope-separation processes is not yet far enough advanced to define the accident and safety considerations that must be considered in the design and operation of such facilities. Some preliminary ideas on this matter are given in Reference 8.

3.3.3 ENVIRONMENTAL CONSIDERATIONS

Environmental issues related to advanced isotope-separation processes are not yet well defined. It appears, however, that many of the issues relate to the occupational environment (Ref. 8). Section 3.0 of Reference 8 discusses environmental issues and requirements as presently perceived. This discussion will not be repeated here.

An important potential of laser isotope separation (LIS) is its ability to strip the tails material accumulated from the operations of gaseous diffusion and gas-centrifuge plants. This stripping process may give a product of near-natural enrichment. This development would expand the nation's uranium resources and reduce the environmental impact of mining and milling. Whether such a reduction would be totally offset by the operation of LIS facilities must await further development and definition of the process itself.

3.3.4 LICENSING CONSIDERATIONS

The comments made previously on gaseous diffusion (see Section 3.1.4) apply here as well. In addition, the early developmental status of advanced isotope-separation processes makes it premature to speculate on licensing considerations for such facilities.

3.3.5 DECOMMISSIONING AND DECONTAMINATION

No information is available on this subject.

3.3.6 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

The following discussion is extracted from Section 3.5 of Reference 8:

> Numerous ES&H R&D studies on uranium and its fluorinated compounds have been conducted during the many years of gaseous diffusion plant operations. Consequently, much of the information necessary to resolve the generic issues already exists. Since personnel from the Oak Ridge Gaseous Diffusion Plant (ORGDP) are involved in the AIS development, availability and transfer of this ES&H information should be readily accomplished. This is particularly true of equipment development, evaluation, and design criteria.
>
> With regard to the key I/R, much less information is available and little work is in progress. Limited work is being conducted on magnetic fields and more is planned for the future. However, it is uncertain whether much of this is directly applicable. Although studies may prove that this is not a serious issue for the isotope separation processes, until it is so demonstrated it must be considered at all stages in the design of AIS facilities.
>
> Although possibly insignificant, the question of chronic uranium toxicity has been raised by recommendations for epidemiological studies of past uranium workers. If such studies indicate that the toxicity is more significant than now considered, it could have an impact on progress in all nuclear fuel cycle steps and jeopardize public acceptance of the technology. Similarly ongoing efforts will consider pertinent state-of-knowledge regarding uranium solubility, for example, as the information relates to health effects.
>
> Information on other toxic chemicals is similarly needed. For certain chemicals such as cadmium, selenium, and ozone, there is a large body of existing knowledge and considerable ongoing research related to air pollution and occupational health. This research needs to be reviewed and evaluated. In the case of many of the chemicals associated with lasers (such as dyes), the field has been changing too fast to permit toxicological evaluation. As a minimum, screening toxicological studies are needed at the present time to aid in selecting laser components for development.

Abbreviations:

ES&H - environmental, safety, health, and socioeconomics
R&D - research and development
AIS - advanced isotope separation
I/R - issues and requirements

REFERENCES FOR CHAPTER 3

1. U.S. Nuclear Regulatory Commission, Final Generic Environmental Statement on the Use of Recycle Plutonium in Mixed Oxide Fuel in Light Water Reactors, NUREG-0002, Vol. 3, August 1976.

2. D. E. Saire and E. B. Kiser, "Operation of DOE's Enriched Uranium Production Facilities," AIF International Conference on Uranium Enrichment, January 30, 1978.

3. U.S. Energy Research and Development Administration, Final Environmental Statement, Expansion of U.S. Uranium Enrichment Capacity, WASH-1543, April 1976.

4. U.S. Department of Energy, Environmental Development Plan (EDP), Uranium Enrichment, DOE/EDP-0027, June 1978.

5. U.S. Atomic Energy Commission, Environmental Survey of the Uranium Fuel Cycle, WASH-1248, April 1974.

6. U.S. Energy Research and Development Administration, Final Environmental Statement, Portsmouth Gaseous Diffusion Plant Expansion, ERDA-1549, September 1977, Section 5.1.3.

7. E. E. El-Hinnawi, "Review of the Environmental Impact of Nuclear Energy," IAEA Bulletin, Vol. 20, No. 2.

8. U.S. Department of Energy, Environmental Development Plan (EDP), Advanced Isotope Separation, DOE/EDP-0022, March 1978.

Chapter 4

FUEL FABRICATION

Fuel fabrication includes the manufacturing steps involved with producing finished fuel assemblies ready for use in a nuclear reactor starting with fissile and fertile fuels converted either from a chemical conversion facility or a reprocessing plant, and various pieces of hardware. The processes, typical of those found in both chemical and manufacturing plants, generate scrap for recycle within the plant, as well as liquid effluents and solid wastes.

Regarding the safety and environmental factors in fuel fabrication, the major impacts are related to occupational safety, although there are important differences among alternatives relative to off-site radiological impacts in fuel fabrication. A distinction is made in the discussion which follows between low-radioactivity and high-radioactivity fuels, the latter requiring more careful handling and involving more shielding and remote operations in the plants in order to reduce occupational exposures. High-radioactivity fuels also usually will result in higher off-site radiological effects under both normal and accident situations.

The specific forms of fuel used in the various reactor/fuel-cycle combinations discussed in this volume are shown in Table 4-1 and consist of the following generic types.

Fab-1 Rod-type fuels of low gamma activity are fabricated in facilities using contact operations and contact maintenance.

Fab-2 Rod-type fuels of low gamma activity are fabricated in facilities using remotely controlled operations and contact maintenance.

Fab-3 Rod-type fuels containing uranium-233 or spiked plutonium of high gamma activity are fabricated in heavily shielded facilities using remotely controlled operations and remotely controlled maintenance.

Fab-7 Microsphere or particle-type fuel for high-temperature gas-cooled reactors (HTGR) is fabricated in facilities using contact operations and contact maintenance (fresh fuel) or in facilities using remotely controlled operations and remotely controlled maintenance (recycle fuel).

Table 4-1 also shows the Nonproliferation Alternative Systems Assessment Program (NASAP) fabrication options related to each of the 13 different types of fuels required for the reactor systems. These numerical designations are shown in the mass-flow diagrams of each of the reactor/fuel-cycle combinations discussed in Volumes I through VI of the Preliminary Safety and Environmental Information Document (PSEID).

The remaining identification numbers, Fab-4 through Fab-6, and Fab-8 through Fab-9 are as follows:

Fab-4 Sphere particle-type fuel of low radioactivity uranium or thorium is vibratory-compacted into rod-type fuel using contact operations and contact maintenance.

Fab-5 Sphere particle-type fuel of moderate radioactivity uranium and plutonium is vibratory-compacted into rod-type fuel using remote operations and contact maintenance.

Fab-6 Sphere particle-type fuel of high radioactivity uranium and thorium or uranium and plutonium is vibratory-compacted into rod-type fuel using remote operations and remote maintenance.

Fab-8 After dry processing of spent fuel, standard techniques are used to produce rod-type fuel using remote operations and remote maintenance.

Fab-9 Rod-type fuel containing metallic cylinders of fuel is produced using remote operations and remote maintenance.

These fabrication options are not discussed further in this chapter since they do not appear in other parts (Volumes I through VI) of the PSEID.

4.1 FAB-1: LOW-RADIOACTIVITY ROD-TYPE FUELS

4.1.1 GENERAL DESCRIPTION

Low-radioactivity rod-type fuels consist of fuel rods loaded with pellets of fissile and/or fertile materials that can be manufactured in facilities using contact operations and contact maintenance. The fissile and fertile feed materials are assumed to be fresh thorium oxide powder, depleted, natural, or low-enrichment uranium hexafluoride or purified uranyl nitrate solution (without any uranium-233) recovered during reprocessing.

The basic process parameters for pellet fuel manufacture are discussed in detail in the literature (Refs. 1-5) and are summarized below. The model Fab-1 plant is presumed to process 520 MTHM/year. The plant parameters are presented in Table 4-2.

Fabrication of rod-type low-radioactivity fuel entails (a) feed material conversion and/or preparation, (b) pellet production and fuel rod manufacturing, and (c) bundle assembly as described below.

a. Feed-Material Conversion and/or Preparation

Feed materials for pellet production must be processed for conversion into a homogeneous oxide powder as free-flowing granules suitable for press feed. Uranium hexafluoride is converted by using either the conventional ammonium diuranate process or the more recent dry-conversion process into uranium dioxide powder. Uranyl nitrate is converted by the ammonium diuranate process. Thorium oxide may require calcination to achieve the proper dryness. These oxide powders will require blending for homogeneity and grinding (or some other size-reduction technique) to a specific particle size before precompaction and granulation. A lubricant and a binder may be blended with the granules, depending on the characteristics of the powder.

b. Pellet Production and Fuel Rod Manufacturing

Powder granules are fed into the die cavities of multiple die or rotary presses for compaction into pellets of a specific density and size. When an organic binder is used, the pellets are processed through a low-temperature binder-removal step before they are sintered at high temperature under a reducing atmosphere to produce the desired density. Sintered pellets are finished to final diameter by centerless grinding, dried, and loaded into fuel rods. Internal hardware is inserted, and the rod end cap is seal-welded under an inert (helium) atmosphere at a specified pressure.

c. Bundle Assembly

Fuel and blanket rods are inspected, tested, and certified before assembly into the final fuel bundles. The final fuel bundles are inspected for dimensional parameters and documentation is checked to assure all rod certifications are available and in order.

4.1.2 SAFETY CONSIDERATIONS

No major unresolved safety issues remain in the fabrication of low-radioactivity fuels involving contact operation and maintenance. Safety concerns center around prevention of accidental criticality, explosions, or fire. Worker safety is of greater concern than off-site impacts. Industrial accidents are equally likely in fuel-fabrication facilities as in other manufacturing plants.

Occupational dose commitments and chemical exposures resulting from accidental releases are estimated to be quite low because of employee training in accident procedures and the use of appropriate protective equipment.

4.1.2.1 Criticality

Nuclear criticality safety is a major consideration, and plant equipment is designed and arranged to preclude accidental criticality, unless several independent failures occur simultaneously.

Operations under administrative control require that two independent levels of review be carried out by technically qualified personnel before the start of operations. All personnel involved in the operations receive instructions in the specific procedures and periodic training sessions in criticality safety. No changes are permitted in the approved procedures without the two-level review and approval.

There have been no criticality accidents to date in process operations where undermoderation is a primary method of control--and few accidents in aqueous or moderated systems. The ventilation filters are assumed to remain intact because a criticality is not an explosive process. The decontamination factor of three high-efficiency particulate air (HEPA) filters in series is assumed to be 2×10^7, lower by a factor of 50 than the decontamination factor assumed for normal operating conditions.

A hypothetical criticality accident (the U.S. fuel-cycle industry has never experienced an accidental criticality with low-enrichment uranium) could result in dose commitments of about 50 mrem to the whole body and about 260 mrem to the thyroid of individuals at the site boundary. This body dose commitment is about 40% of the average annual dose an individual receives from natural background radiation.

4.1.2.2 Explosion

Combustible gases may be present at the sintering furnace and in the clean-scrap reduction operation. In addition, flammable solvents are used in the dirty-scrap recycle operation and may be used for cleaning fuel rods and during cleanup and maintenance. These operations are the only ones with a potential for supporting an explosion.

To minimize the potential for explosion, sintering furnaces and the clean-scrap reduction operations use hydrogen diluted with an inert gas to keep the mixture below the explosive limit. A hydrogen explosion in this plant is therefore considered highly unlikely. A hypothetical furnace explosion could result in dose commitments of 1 x 10^{-4} mrem to the lungs of individuals at the site boundary. This is below the normal variation in background dose and represents no measurable radiological impact.

The dirty-scrap process uses a commercial organic solvent, generally kerosene, that is widely used and is not considered to present a major hazard of explosion; however, it is possible that more flammable solvents may be used in cleanup or maintenance operations. The quantity of such solvents will be kept to a minimum and their use controlled. The possibility of a flammable solvent explosion is thus unlikely.

The consequences of an explosion are similar to those of a fire. The amount of uranium and/or thorium reaching or passing through the filters is estimated to be the same as that estimated for the fire (see below) and would have the same relatively minor off-site consequences.

4.1.2.3 Fire

Unlike a criticality excursion or an explosion, a fire is usually not an instantaneous event and very often starts from a small flame source. The design, construction, and operation of fuel-fabrication plants considers in detail the possibility of a fire, and the equipment and procedures for fire prevention. In general, operators have time to react to and extinguish small fires. The process materials, oxides of uranium and thorium, are not themselves flammable.

The final HEPA filters are located some distance from the hoods and gloveboxes. The separation distance should be sufficient to protect the filters against the effects of an explosion or fire. A fire or explosion is assumed to destroy the local filters with uranium oxides reaching the final filters. Assuming that 100 grams of uranium will reach the filters and that each filter will remove 99.9% of the particulates, it is calculated that a total of 0.1 mg of uranium will pass through the filters.

The probability of a major fire in a modern facility is small. In the unlikely event of a fire, however, the calculated total additional dose commitment from the accident would be approximately 10% of the annual dose commitment estimated to accrue from normal operations.

4.1.3 ENVIRONMENTAL CONSIDERATIONS

Table 4-3 summarizes the environmental effects of a model plant producing low-enrichment uranium as UO_2 fuel at an annual throughput of 520 MTHM/year. Compared to the other steps in the nuclear fuel cycle, fabrication of UO_2 fuels represents an insignificant impact (Ref. 2).

The fabrication of thorium, uranium-thorium, or duplex fuel for the light-water breeder reactor (LWBR) is expected to cause environmental impacts similar to those caused by fabrication of UO_2 fuel. There are minor differences as discussed in the following sentences. The former fuels will introduce minor additional amounts of thorium in the solid- and liquid-waste effluents. Although the uranium-235 assay in thorium-uranium fuel pellets would be 10 to 20%, the curie content of uranium in the effluents is expected to be similar to that from low-enrichment UO_2 pellet fuels because the quantity of uranium-235 per kilogram of fabricated fuel is less. Whereas calcium fluoride is a primary solid effluent in fabrication of low-enrichment UO_2 pellet fuel, the quantity generated in the manufacturing of these other fuels would be reduced since less UF_6 would be processed.

The estimated average annual radiological dose commitments from the model UO_2 fabrication plant to the closest theoretical resident are presented in Table 4-4. Doses for other fuel types are not well quantified but would be expected to be similar based on the similarity in source terms and effluents discussed above. It should be noted that even with recycle of uranium, the increase in population dose resulting from UO_2 fabrication operations is very small (Ref. 2).

4.1.4 LICENSING STATUS[a]

The present fuel-fabrication industry for rod-type fuels consists of nine licensed commercial plants. Of these, one plant is currently shut down and being held on standby, and another is being used by contract for the recovery of contaminated scrap generated by others. Two manufacturers produce uranium dioxide powder from uranium hexafluoride for their fuel-assembly-fabrication facilities at other locations. Several fabricators are considering capacity expansion and process modifications, such as changing from the ammonium diuranate process to the dry conversion process. The present annual industrial capacity of 2,700 MTU is adequate for light-water reactor (LWR) needs through year 1979. Because of increasing reactor fuel demands, corporations already licensed in the field have formulated plans to expand capacity in the future. Increasing capacity beyond present licensed limits or the addition of new capacity will require the preparation of environmental and safety analysis reports for new facilities and equipment. No unusual licensing problems are anticipated.

The manufacture of mixed uranium-thorium dioxide pellets, thorium dioxide pellets, and the fuel rods containing these pellets, uses much of the same technology, equipment, and types of facilities as those used for the LWR UO_2 pellet fuel. The addition of thorium to the system does not impose any licensing difficulties. In the past, however, all mixed uranium-thorium dioxide fuels have used high-enrichment uranium-235 as the fissile component, except for the Shippingport LWBR fuel, which uses uranium-233 as the fissile component. Facility licenses for such fuels would have to include the handling of high-enrichment fuel. The use of medium-enrichment uranium in place of high-enrichment fuel for nonproliferation reasons would not present unresolvable licensing problems, although criticality control would be more important.

4.1.5 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

Fabrication of rod-type fuel assemblies in low-gamma-activity contact facilities has a long history of development and demonstration. Current work is limited to upgrading fuel performance, improvement of reliability and fuel life, and modification of the basic processes to reduce the wastes generated during production and to approach zero-release conditions. Several facilities are involved in the demonstration under reactor conditions of fuels made with pellets produced by the dry conversion of uranium hexafluoride. The industry has established process parameters for producing fuels with satisfactory performance. An exception is the "duplex" conceptual fuel for LWBR prebreeder fuel cycles. This duplex fuel, consisting of a uranium dioxide/calcium

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[a] See Reference 2 for more detailed information.

oxide/zirconium dioxide annular pellet with a thorium dioxide center, has had significant process development and is currently undergoing in-reactor testing.

4.1.6 DECOMMISSIONING AND DECONTAMINATION[a]

Fuel-cycle facilities become contaminated in the course of fuel-fabrication and waste-treatment operations. On retirement, these facilities become a waste that requires managing by decommissioning. Of the various alternatives available, two basic decommissioning modes have been considered: (1) immediate dismantling, and (2) safe storage with deferred dismantling.

If immediate dismantling is selected as the mode of decommissioning, at shutdown all radioactive contamination above regulatory limits is removed from the facility to an approved disposal site. Depending on further uses of the fuel-cycle facility site, noncontaminated portions of the facility remaining after dismantling may be demolished and removed or converted to alternative uses.

If safe storage with deferred dismantling is selected, preparations are made to leave the facility in place for an extended time by installing temporary physical barriers. Continuing surveillance and maintenance are required.

Regulatory criteria for decommissioning and decontamination of Fab-1 type plants are being developed. Although one facility is shut down and decommissioning studies are under way, the only facilities for LWR fuel-pellet manufacture that have been decontaminated and decommissioned were pilot facilities. The Westinghouse Electric Corporation and the General Electric Company successfully decontaminated and converted for other uses their production lines at Cheswick and San Jose, respectively, when their commercial plants at Columbia, South Carolina, and Wilmington, North Carolina, were placed in operation.

A major problem in decommissioning is expected to be the ultimate disposal of the calcium fluoride waste sludge in storage ponds. Plant building and equipment can be decontaminated to a level that permits general usage of the facility.

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[a] From Reference 6.

Table 4-1. Fuel forms used in various reactor/fuel-cycle combinations

Fabrication option	Fuel form	Vol. I-LWR (figure nos.)					Vol. II-HWR	Vol. III-LWBR								Vol. IV-HTGR				Vol. V-GCFR	Vol. VI-LMFBR						
		2-1	3-1	4-1	5-1	6-1	2-1	2-6	2-7	3-5	3-6	4-2	4-3	5-4	5-5	2-2	3-1	4-1	5-1	2-3	2-1	3-1	4-1	5-1	6-1	7-1	8-1
	Rod fuel																										
Fab-1[a]	1 U(5) O$_2$	X	X	X			X														X	X	X	X			
	2 U(5) O$_2$-CaO ZrO$_2$, ThO$_2$							X		X				X													
	3 Th O$_2$							X		X			X	X	X					X				X	X	X	X
	4 (U(5)-Th) O$_2$											X															
Fab-2[b]	5 U(5)-Pu O$_2$																			X	X	X	X	X	X		
Fab-3[c]	6 Pu-Th O$_2$																									X	
	7 U(3) O$_2$																										X
	8 U(3)-Th O$_2$				X				X		X		X		X												
	9 U(5) Pu O$_2$ (spiked)			X																							
	10 Pu-Th O$_2$ (spiked)					X																					
	HTGR fuel																										
Fab-7[a,b,c]	11 U(5) Th OC[a]															X		X									
	12 Th O$_2$[b]															X	X	X	X								
	13 U(3) O$_2$[c]																X	X	X								

Abbreviations:
LWR = light-water reactor
HWR = heavy-water reactor
LWBR = light-water breeder reactor
HTGR = high-temperature gas-cooled reactor
GCFR = gas-cooled fast-breeder reactor
LMFBR = liquid-metal fast-breeder reactor

[a] Low gamma activity; contact operation; contact maintenance.
[b] Low gamma activity; contains plutonium; remote operation; contact maintenance.
[c] High gamma activity; remote operation; remote maintenance.

Table 4-2. Fab-1: Rod fuel fabrication model plant

Fuel materials	UO_2, ThO_2, $(U\text{-}Th)O_2$, $UO_2\text{-}CaO\text{-}ZrO_2\text{-}Tho_2$
Mode of operation	Contact operation, contact maintenance
Production capacity	520 MTHM/yr
Design capacity	
3-rod production lines	240 MTHM/yr each line
Fuel assembly line	730 MTHM/yr
Plant efficiency	72%
Operating schedule	24 hr/day, 7 days/wk
Principle of operation	Toll processing of source and special nuclear material with sufficient feed provided by customer; other materials provided by fabricator
Feed materials	UF_6, UNH, ThO_2
Waste treatment	Low-level wastes prepared and packaged for shallow land burial as solids
Scrap recycle	Clean and dirty scrap recycled in the fuel-fabrication plant

Table 4-3. Environmental effects of the Fab-1 model fuel-fabrication plant[a] with a 520-MTHM/yr plant capacity[b]

Land use, acres	
Site size[c]	100
Plant and services	20
Temporarily committed area	3
Actually disturbed land	0.5
Building (two-story, windowless structure), ft^2	100,000
Water use, gpd	
Process	16,000
Cooling water	284,000
Power	
Electric, MWe	4
Gas, scf	2.1×10^6
Chemical effluents	
Gases--fluoride, $\mu g/m^3$	0.1[d]
Liquids	
Volume, gpd	300,000
Mass, MT	
Fluoride	61.5
Nitrogen as nitrates	79.5
Nitrogen as ammonia	126
Concentration, mg/l	
Fluoride	200
Nitrogen as nitrates	280
Nitrogen as ammonia	420
Solids	
Volume, cubic yards	286
Content--calcium fluoride, MT/yr	300
Radiological effluents	
Gases	
Average annual uranium content, $\mu Ci/cm^3$	2.2×10^{-12}
Total annual release, Ci	0.003
Liquids	
Uranium content, mCi	300
Thorium-234 content, mCi	150
Protactinium-234 content, mCi	150
Uranium concentration, $\mu Ci/ml$	1×10^{-6}
Thorium-234 concentration, $\mu Ci/ml$	5×10^{-7}

See footnotes at end of table.

Table 4-3. Environmental effects of the Fab-1 model fuel-fabrication plant[a] with a 520-MTHM/yr plant capacity[b] (continued)

Protactinium-234 concentration μCi/ml	5×10^{-7}
Solids	
Uranium content in CaF_2, Ci	3.45
Uranium concentration in CaF_2, μCi/g	0.01
Uranium content in waste, mCi	375
Thermal releases (waste heat), Btu/yr	13.5×10^{10}

[a] Data from Reference 3.
[b] Scaled from data for a 900-MTHM/yr plant.
[c] Data from Reference 2.
[d] At the site boundary.

Table 4-4. Estimated annual average dose commitments to the closest theoretical resident[a]

	Dose commitment, mrem/yr[b]			
	No U or Pu recycle		With U recycle	
Organ	Air pathway	Water pathway	Air pathway	Water pathway
Whole body	0.04	0.09	0.04	0.9
GI tract	0.033	0.15	0.035	0.16
Bone	0.45	1.4	0.45	1.5
Liver	0.012	2.8×10^{-6}	0.012	2.8×10^{-6}
Kidney	0.25	0.34	0.26	0.35
Thyroid	0.011	1.6×10^{-6}	0.011	1.6×10^{-6}
Lung	0.34	1.6×10^{-6}	0.35	1.6×10^{-6}
Skin	0.011	---	0.011	---

[a] Data from Reference 2, Table IV-F-19, normalized from 1,500 MTHM/yr throughput.

[b] Based on effluents from UO_2 fabrication plants processing 520 MTU/yr for each option. $\chi/q = 5.4 \times 10^{-6}$ sec/m^3 at 500 meters from ground-level release.

4.2 FAB-2: LOW-RADIOACTIVITY ROD-TYPE FUELS CONTAINING PLUTONIUM

4.2.1 GENERAL DESCRIPTION

Low-radioactivity rod-type fuels containing plutonium consisting of fuel rods loaded with pellets or packed spheres of recycle plutonium oxide or carbide are assumed to be manufactured in facilities using remote operation in lightly shielded gloveboxes and contact maintenance of equipment. Fissile feed materials received from reprocessing would be plutonium-uranium oxide powder.

An alternative form of fuel material would be uranium-plutonium in the form of either oxide or carbide dried gel microspheres. Fertile feed material is depleted UF_6 for use as a diluent for the fissile material. Recycle plutonium with a high plutonium-240 content will require some gamma shielding and neutron shielding to minimize occupational exposure.

The basic parameters for both pellet and Sphere-Pac fuel manufacture are discussed in detail in the literature (Refs. 1-5) and are summarized below. The model recycle-fuel fabrication plant for low gamma activity fuel is assumed to process 480 MTHM/year. The plant parameters are presented in Table 4-5.

4.2.1.1 Feed Material Conversion and/or Preparation

Depleted uranium hexafluoride is converted into oxide powder as described in Section 4.1.1a. Mixed U-Pu oxide powder is blended with the depleted UO_2 to the desired fissile assay. The blended powder may be comminuted to a desired particle size before precompaction and granulation. A lubricant and a binder may be blended with the granules, depending on the characteristics of the powder.

For the alternative fuel form, dried gel microspheres of mixed oxide from reprocessing are calcined and sintered to density in a reducing atmosphere. Sintered oxide or carbide microspheres are separated into particle-size ranges and blended in the proper ratio of sizes to assemble batches for gravity loading into rods.

4.2.1.2 Pellet Production and Fuel Rod Manufacturing

Powder granules are fed into the die cavities of multiple die or rotary presses for compaction into pellets of a specific density or size. When an organic binder is used, the pellets are processed through a low-temperature binder-removal step before they are sintered to high density. Sintered pellets are finished to final diameter by centerless grinding, dried, and inspected in preparation for rod loading.

A major process variation in the manufacture of pellets is the production of U-Pu carbide pellets. The mixed U-Pu oxide powder received from the reprocessing plant is blended with graphite and pressed into briquets. Following carbothermic reduction, the carbides are comminuted to the desired particle size under inert atmosphere to prevent reaction with moisture and oxygen. Carbide pellets are manufactured in the same manner as the oxide pellets described above, with the exception that lubricants, binders, and sintering aids are in the form of dry powders. All pellet production and grinding are conducted in an inert atmosphere.

Sintered pellets of oxide or carbide are loaded into the fuel rods in weighed and measured stack lengths. Carbide pellets are handled and loaded in an inert atmosphere.

Internal hardware is inserted and the rod end cap is seal-welded under an inert helium atmosphere at a specified pressure. An alternative heat-transfer agent considered for use in liquid-metal fast-breeder reactors (LMFBRs) is sodium to fill the annulus between the pellets and the cladding. Sodium loading would also be conducted in an inert atmosphere.

For the alternative fuel form, weighed batches of sized microsphere blends are loaded by gravity into a gently vibrating, vertically oriented fuel rod. The rod is compacted by vibration before internal hardware is inserted and rod end cap is seal-welded under an inert atmosphere of helium at a specified pressure.

4.2.1.3 Bundle Assembly

Certified fuel rods are decontaminated and assembled under contact operating conditions. Because of the neutron radiation, hydrogenous shielding may be necessary to reduce the occupational exposure.

4.2.1.4 Preirradiation

Preirradiation refers to the exposure of fabricated fuel assemblies to a neutron flux of adequate strength and duration to generate activation products in sufficient quantity to render the fuel containing recycle plutonium too radioactive to handle without significant exposure to a terrorist attempting diversion. The level of radiation that would be significant is not yet determined and depends on the diversion scenario. Appendix A provides some discussion of this subject. A level of 10 rads/hr at 1 meter (equivalent to 10 rem/hr) for a fuel assembly 2 years after irradiation would require a burnup in the preirradiation facility of 40 MTD/MT of fuel. The preirradiation will require handling and transporting the fuel under shielded conditions to minimize occupational exposure and to meet the transport limits of 200 mrem/hr at contact and 10 man-rem/hr at 6 feet from the container. One type of preirradiation facility could be a water-cooled, pool-type reactor to irradiate the fuel. It has been estimated (Ref. 7) that it will require about 6 to 7 days of exposure in a 100-MWT reactor to achieve the 40 MWD/MT burnup.

Such a facility would require many of the safety systems required for a power reactor. A development program will be needed to determine operating parameters for the preirradiation step and to provide design criteria for equipment needed for materials handling of the "hot" new fuel. The spent-fuel pool of the reactor will provide temporary storage of the irradiated assemblies. It will be necessary to design special shipping containers, receiving stations at the power reactors, remotely operated test equipment to evaluate the as-received fuel, and shielded new-fuel storage facilities if the spent-fuel pool capacity at the power reactor is limited.

4.2.2 SAFETY CONSIDERATIONS[a]

Mixed-oxide fuel-fabrication plants processing uranium-plutonium dioxides or carbides are required by 10 CFR 70.22(f) to be designed, constructed, tested, and operated under rigid quality-assurance programs. Quality assurance consists of all the planned and systematic actions necessary to provide adequate confidence that structures, systems, components, and operations will perform satisfactorily in service (10 CFR 50, Appendix B).

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[a] Excludes the reactor used for preirradiation, if any.

At plants fabricating mixed-oxide fuel, all operations that involve the handling of plutonium, except when it is contained in shipping containers or sealed fuel rods, are carried out inside shielded process enclosures (gloveboxes). These enclosures provide confinement of plutonium in the event of equipment failure. The process building, essential equipment, and supporting systems are designed to withstand impacts resulting from natural phenomena such as tornadoes, earthquakes, and floods.

During the life of a mixed-oxide fuel-fabrication plant, some equipment (or accessory) failures may occur. Monitors are installed to detect such failure or process upset conditions that can cause safety-related damage. Some correctable action is automatically provided. The ventilation system is designed to function during and after accidents and to pass all plant ventilation air through HEPA filters before releasing it to the atmosphere. In summary, a plant fabricating mixed-oxide fuel is (a) designed, constructed, tested, and operated according to a quality-assurance plan; (b) designed to cope with accidents; and (c) designed to minimize the off-site consequences of potential accidents.

Some incidents, such as punctures or tears in gloves or other glovebox malfunctions, are expected to occur during normal operations. More serious accidents, such as glovebox window breakage, will occur far less often, and although their off-site consequences are judged to be insignificant, such accidents are included in the estimate of airborne effluents resulting from normal operations. Upper limit accidents that may occur include a criticality incident, an explosion, or a fire.

4.2.2.1 Criticality

The considerations for criticality for a Fab-2 facility are the same as for a Fab-1 facility, as described in Section 4.1.2.1. In addition, calculations show that the maximum off-site individual dose commitment results from the absorption of fission-product iodine in the thyroid and amounts to 360 mrem, a dose comparable to that received by the closest theoretical resident in a criticality accident at a uranium dioxide fuel-fabrication plant. The slightly different fission-product yield and the presence of small amounts of plutonium particulates do not make the effects of a plutonium dioxide criticality accident significantly different from those of a uranium dioxide criticality accident.

4.2.2.2 Explosion

The enclosure of processing operations in gloveboxes and the extensive ventilation systems used in a Fab-2 facility require even greater care to assure that combustible and potentially explosive materials are closely controlled to ensure inert or nonexplosive atmospheres. In the event of an explosion, the glovebox windows and gloves are most likely to rupture but the products of the explosion would be contained within the building under the control of the building heating, ventilation, and air-conditioning (HVAC) system resulting in minor off-site consequences. The amount of plutonium reaching or passing through the final HEPA filters is estimated to be the same as for a fire (see below).

4.2.2.3 Fire

Detailed consideration is given in the design, construction, and operation of Fab-2 type plants to the possibility of a fire, and to the equipment and procedures for fire prevention. In general, operators have time to react to and extinguish small fires. The process materials, oxides of uranium and plutonium, are not themselves flammable. The final filters are protected against fire by water-spray systems installed in the duct some distance upstream of the final filters. Mist deflectors or collectors are installed between the water-spray system and the filters to remove large drops of water. The water from the sprays collects in the bottom of the ducts and flows to a fire-water collection tank. This tank is either a safe-geometry tank or a fixed-poison-controlled tank to preclude the possibility of a criticality accident as the result of a fire.

The final HEPA filters are located some distance from the gloveboxes. The separation distance and the water-spray system should be sufficient to protect the filters against the effects of an explosion or fire. A fire or explosion is assumed to destroy the local filters on the gloveboxes, with plutonium and uranium oxides reaching the final three filters. Assuming that 100 grams of plutonium will reach the three filters and that each filter will remove 99.9% of the particulates, it is calculated that a total of 0.1 mg of plutonium will pass through the filters.

The probability of a major fire in a modern plutonium facility is small. In the unlikely event of a fire, however, the calculated total additional dose commitment from the accident would be approximately 10% of the annual dose commitment estimated to accrue from normal operations.

4.2.3 ENVIRONMENTAL CONSIDERATIONS

The Fab-2 type mixed oxide processing facility will not have a high impact on the environs, nor will it interfere with the use of adjacent land and streams. Calculations based on conservative assumptions show that a Fab-2 facility will release about 520 mg of mixed oxides per year (12 mCiα) to the atmosphere. The annual release from waterborne effluents is assumed to be 97 mg of oxides containing 4.1 mg of plutonium (2.1 mCiα).

The maximum annual deposition of plutonium occurring on the plant property about 100 meters from the release point will be on the order of 8×10^{-9} Ci/m^2. (The atmospheric diffusion characteristics are discussed in Ref. 2.) The maximum annual deposition at the site boundary, 500 meters from the plant, will be about 6.7×10^{-10} Ci/m^2. Assuming an annual airborne release of 20 mg and no downward migration of plutonium in the soil, the maximum surface accumulation for 20 years of plant operation will be less than 5×10^{-7} Ci/m^2 in the top millimeter of soil (proposed as standard for ground contamination, Ref. 8). Therefore, plant sites used for mixed-oxide-fuel production are expected to be returned to unrestricted use after plant shutdown.

Annual average daily chemical releases consist of about 1.3 grams of hydrogen fluoride, 1,600 grams of nitrogen oxides, and 215 grams of ammonia. Applying an atmospheric diffusion factor at the plant boundary of 5.4×10^6 sec/m^3 results in the following average concentrations at the plant boundary:

Effluent	Concentration (μg/m^3)
Hydrogen fluoride	$.8 \times 10^{-5}$
Nitrogen oxides	9.9×10^{-2}
Ammonia	1.3×10^{-2}

The off-site concentration of hydrogen fluoride is about 0.016% of the most restrictive state standard of 0.5 μgm/m^3 (Ref. 9). The concentration of nitrogen oxides is about 0.099% of the U.S. Environmental Protection Agency (EPA) standard of 100

μ gm/m^3, and the concentration of ammonia is about one-millionth of the occupational limit established by the American Conference of Government and Industrial Hygienists.

Table 4-6 summarizes the environmental effects of the model Fab-2 plant; Table 4-7 presents the estimated average annual dose commitment to the nearest theoretical resident.

4.2.4 LICENSING STATUS[a]

The Exxon Nuclear Corporation currently has a licensed 1-MTM/day mixed-oxide (uranium-plutonium) fuel-pellet plant on standby. The Westinghouse Electric Corporation, the Kerr-McGee Corporation, and Babcock & Wilcox (Lynchburg facility) have licensed development facilities that have been working on the production of fuel rods for the Fast Flux Test Facility. The Kerr-McGee facility is now on standby, and Kerr-McGee is considering decommissioning the plant. Westinghouse has filed an application to construct and operate a commercial plant in South Carolina. The General Electric Company has produced developmental quantities of mixed-oxide fuel at its Vallecitos laboratory. Fuel has also been produced at the Erwin, Tennessee, facilities of Nuclear Fuel Services, Inc. (NFS); the latter were licensed for this purpose. Nuclear Fuel Services, Inc., produced the 20% fissile plutonium oxide pellets in stainless-steel fuel rods used in the SEFOR reactor experiment constructed and tested by General Electric in Arkansas. The NFS facilities are also on standby. The suspension of the GESMO[b] hearings resulted in NRC stopping all licensing case review of the Westinghouse application for the mixed-oxide production plant.

No facility for producing mixed plutonium-uranium carbide fuel pellets has been licensed. However, such a facility would be similar in process, equipment, and environmental impacts to a mixed uranium-plutonium dioxide plant, and its licensing is not expected to be more difficult than that of a mixed-oxide plant.

A facility for producing fuel of oxides or carbides of mixed uranium-plutonium by the Sphere-Pac process has not been licensed in the United States, nor have an application for license, a safety analysis report, or an environmental report been filed with the U.S. Nuclear Regulatory Commission (NRC). Such filing is not probable until current restrictions on the use of plutonium are lifted and reprocessing facilities are authorized for operation. A Sphere-Pac facility is similar to mixed-oxide pellet-fuel plants and similar licensing requirements are to be expected.

4.2.5 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

Mixed-oxide and carbide fabrication facilities for producing pellet fuels containing plutonium have been shown to be technically feasible in pilot facilities, and mixed-oxide fuels have been and are under irradiation in LWRs. Table 4-8 presents a brief summary of development requirements for a mixed-oxide plant. Although plutonium recycle has been shown to be technically feasible, the mixed-oxide fuel-fabrication industry is practically nonexistent. Many of the fabrication operations cannot be performed with

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[a] See Reference 2 for more detailed information. Excludes the reactor used for preirradiation, if any.

[b] Final Generic Environmental Statement on the Use of Recycle Plutonium in Mixed Oxide Fuel in Light Water Reactors.

equipment or procedures developed for the typical enriched-uranium fuel-fabrication plant. Personnel dose limits and plutonium safeguards requirements necessitate the use of highly mechanized equipment for hands-off operation. These requirements make even current well-developed equipment (hydraulic presses, sintering furnaces, etc.) inadequate for plutonium fuel fabrication without extensive modification.

The status of the development of microsphere and Sphere-Pac technology is far behind pellet technology. The microsphere and Sphere-Pac concepts originated in the United States almost 20 years ago (as sol-gel), and active development was pursued until 1972. At that time, the U.S. fast-reactor program concentrated on pellet fuel. However, several countries--including the Netherlands, the United Kingdom, the Federal Republic of Germany, and Switzerland--have continued microsphere-fuel development and have made significant contributions. The U.S. Gel-Sphere-Pac work was reinitiated in June 1977.

The status of development for the two major Gel-Sphere-Pac areas (gel-sphere preparation and Sphere-Pac loading) is roughly the same. Gel-sphere preparation has received more attention for a longer period than Sphere-Pac loading, but it is significantly more complex. Therefore, their remaining overall development times are estimated to be about equal.

In general, cold laboratory work provides the basis for both hot laboratory and cold engineering work. The latter two together provide the basis for both hot engineering and cold prototype work, which in turn provides a solid basis for a commercial-scale facility. Fuel samples for irradiation testing would normally be produced during all stages of development. In the above sequence, Sphere-Pac development is well into the cold laboratory stage. Work is beginning on both the hot laboratory and the cold engineering phases.

Some irradiation testing has been done, both in the United States and in Europe, with test rods of various lengths, with generally favorable results. Additional tests are under way in Europe and in preparation in the United States.

From a functional viewpoint, Sphere-Pac development may be divided into the following areas: calcination and sintering, sphere characterization, fuel-rod loading, fuel-rod inspection, and scrap recycle.

Calcining and sintering are being done successfully on a batch basis, yielding a product with more than 98% of theoretical density. However, considerable development is still required in order to understand and optimize these processes for the heavy-metal compositions of interest. Scaleup will require equipment for continuous operation and/or larger batches while providing the necessary atmosphere control, residence time, and uniformity.

Considerable technology has been developed for microsphere characterization as part of the HTGR program. Contact or glovebox techniques have been developed for the determination of particle density, size, shape, chemistry, crushing strength, and microstructure. However, a need still exists for techniques that can be applied to the characterization of microspheres and more rapid methods of chemical analysis.

The Sphere-Pac process for loading a fuel rod involves vibratory packing of carefully sized spheres of the proper size ratio. Considerable technology has been developed for the identification of proper sizes, size ratios and blending ratios, and loading sequences to produce maximum smear densities and minimum loading times.

Sphere-Pac loading of commercial-length rods remains to be demonstrated. Simultaneous loading of all three size fractions shows promise of overcoming the problem of excessively long loading time for long rods. Much of the particle dispensing and blending technology developed for HTGR fuels is applicable.

Significant development is required for economic inspection of fuel rods with acceptable precision, accuracy, and speed. Most of the development is required for semi-remote inspection, regardless of whether the fuel rod is fabricated from pellet or Gel-Sphere-Pac fuel.

Compared to pellets, scrap recycle in Gel-Sphere-Pac fuel material is a much smaller problem since microsphere dimensions are not as critical as pellet dimensions. Any defective microspheres can be recycled after drying, while dissolution is still relatively easy, before sintering to density.

Very little effort to date has been directed toward the planning and analysis of an integrated commercial refabrication plant based on Gel-Sphere-Pac technology. Concepts for Gel-Sphere-Pac processes and equipment are rapidly progressing to the point where meaningful evaluation can and should be performed.

A thorough irradiation test program is needed since performance is the crucial item in the final acceptance of Gel-Sphere-Pac fuel from both commercial and licensing aspects. The technical feasibility of the Gel-Sphere-Pac process for fabricating fuels is certain for all fuel types currently under consideration. However, as described, commercial application is dependent on considerable development to bring the various alternative processes to a suitable reliability status at commercial-scale capacity. Furthermore, equipment appropriate to a given reactor-fuel-element design must be designed and tested to establish commercial operations.

4.2.6 DECOMMISSIONING AND DECONTAMINATION[a]

The discussion in Section 4.1.6 for Fab-1 facilities is applicable to Fab-2 facilities. The major problem in decommissioning a Fab-2 facility would be the processing and ultimate disposal of transuranic (TRU) wastes in a geologic repository. The limited amount of depleted UF_6 used would generate some calcium fluoride waste sludge which would require removal from the storage ponds, packaging, and shipment to a low-level waste, shallow land burial site.

Much experience has been gained in the United States with the decontamination and dismantling of pilot plutonium facilities and thorium-processing facilities. The techniques used are readily adaptable to mixed-oxide fuel-fabrication plants. The experience gained in decontaminating the ORNL Metal Recovery Plant and dismantling the Savannah River and Hanford chemical processing plant cells may be applicable. Facilities that previously processed uranium/plutonium dioxide pellets have been decontaminated and converted to other nuclear facility uses, and some are in the process of being decommissioned.

Regulatory criteria for decommissioning and decontamination of a Fab-2 type mixed-oxide fuel-fabrication facility have not been developed.

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[a] From Reference 6.

Table 4-5. Fab-2: Rod fuel fabrication model plant

Fuel materials	(U-Pu) oxide or carbide
Mode of operation	Remote operation, contact maintenance
Production capacity	480 MTHM/yr
Design capacity	
3-rod production lines	240 MTHM/yr, each line
Fuel assembly lines	730 MTHM/yr
Plant efficiency	67%
Operating schedule	24 hr/day, 7 days/wk
Principle of operation	Toll processing of source and special nuclear materials with sufficient feed provided by the customer; other materials provided by fabricator
Feed materials	Mixed oxides of uranium and plutonium; depleted UF_6. Alternative is gel microspheres of U-Pu oxide or carbide
Waste treatment	Low-level wastes prepared and packaged for shallow land disposal; TRU wastes solidified and packaged for geologic disposal

Table 4-6. Environmental effects of the Fab-2 model fuel-fabrication plant[a] at a 480-MTHM/yr plant capacity[b]

Land use, acres	
Site size[c]	100
Plant and services	20
Temporarily committed area	3
Actually disturbed land	0.5
Building (two-story, windowless structure), ft^2	~100,000
Water use, gpd	
Process	15,000
Cooling water	260,000
Power	
Electric, MWe	3.7
Gas, scf	19×10^6
Chemical effluents	
Gases--fluoride, $\mu g/m^3$	<0.1[d]
Liquids	
Volume, gpd	275,000
Mass, MT	
Fluoride	57.4
Nitrogen, as nitrates	74.2
Nitrogen, as ammonia	117.6
Concentration, mg/l	
Fluoride	--
Nitrogen, as nitrates	--
Nitrogen, as ammonia	--
Solids	
Volume, cubic yards	154
Content--calcium fluoride, MT/yr	364
Radiological effluents	
Gases	
Average annual uranium content, $\mu Ci/cm^3$	2.2×10^{-12}
Total annual release, Ci	0.0027
Plutonium, mCi (*a*)	12
Liquids	
Uranium content, mCi	280
Thorium-234 content, mCi	140
Protactinium-234 content, mCi	140
Plutonium content, mCi (*a*)	2.4 (4.1 mg/yr)
Uranium concentration, $\mu Ci/ml$	1×10^{-6}
Thorium-234 concentration, $\mu Ci/ml$	5×10^{-7}
Protactinium-234 concentration, $\mu Ci/ml$	5×10^{7}

See footnotes at end of table.

Table 4-6. Environmental effects of the Fab-2 model fuel-fabrication plant[a] at a 480-MTHM/yr plant capacity[b] (continued)

Solids	
Uranium content in CaF_2, Ci	3.22
Uranium concentration in CaF_2, μCi/g	0.01
Uranium content in waste, mCi	350
Thermal releases (waste heat), Btu/yr	12.6×10^{10}

[a] Data from Reference 3.

[b] Scaled from data for a 900-MTHM/yr plant.

[c] Data from Reference 2.

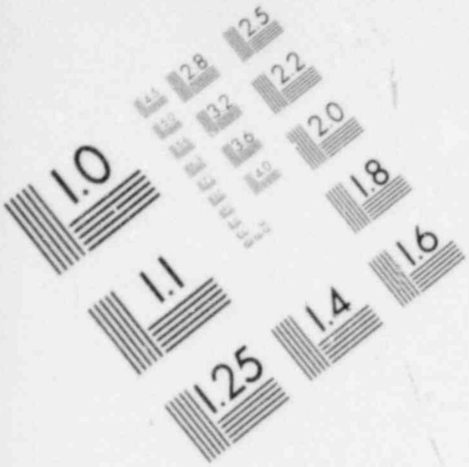
[d] At the site boundary.

Table 4-7. Estimated annual average dose commitment to the closest theoretical resident[a]

Organ	Airborne effluents Resident at 500 m (mrem)	Waterborne effluents Resident at 500 m (mrem)
Whole body	4.9	1.5×10^{-4}
Bone	228	5.9×10^{-3}
Gastrointestinal tract	0.2	5.9×10^{-4}
Lung	5.2	--
Liver	23.6	6.9×10^{-4}
Kidney	20.3	6.9×10^{-4}
Skin	0.09	6.9×10^{-4}

Note: Plutonium and americium are assumed to be insoluble in deriving lung dose and soluble in deriving all other doses.

[a] Scaled to 480 MTHM/yr throughput; data from Reference 2, Tables IV-D-10 and IV-D-11.



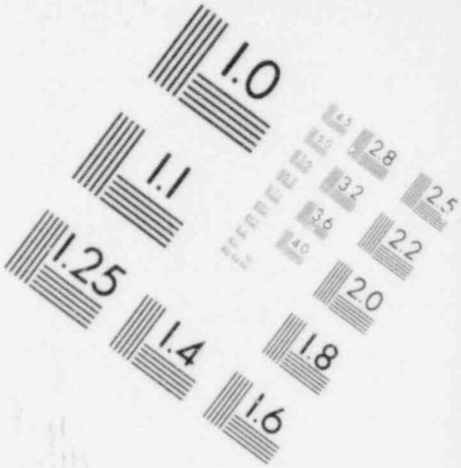
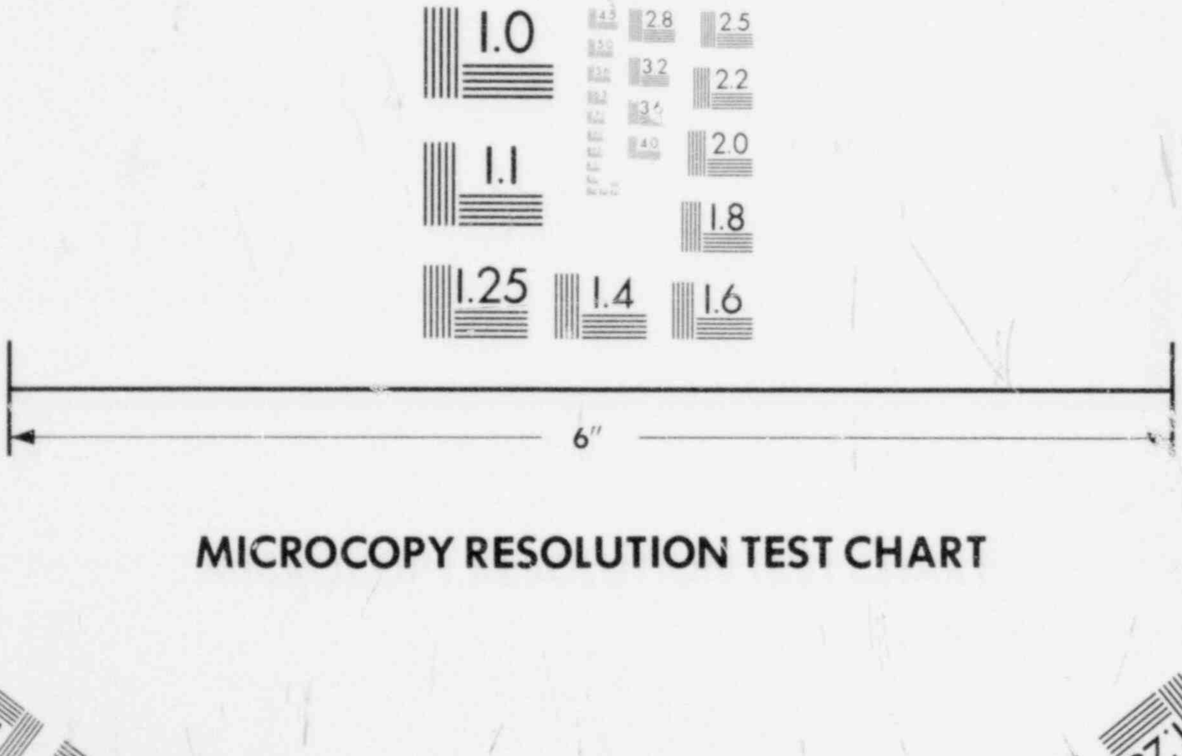
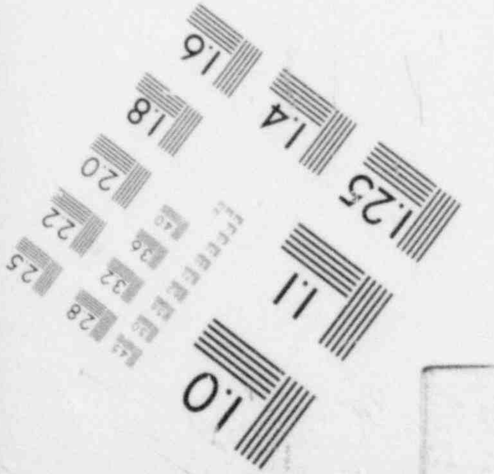
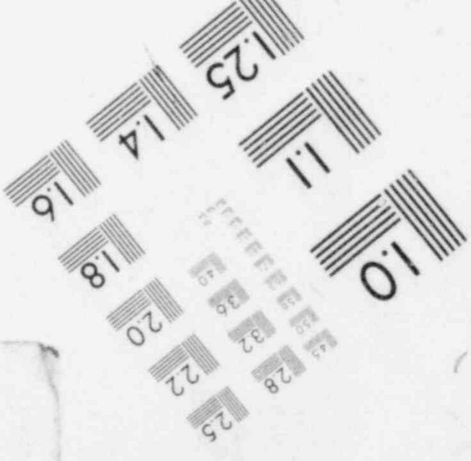


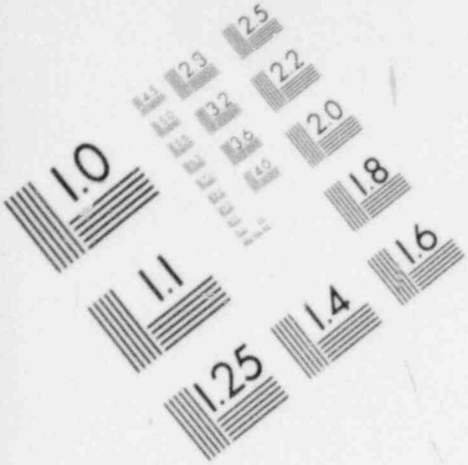
IMAGE EVALUATION
TEST TARGET (MT-3)



MICROCOPY RESOLUTION TEST CHART







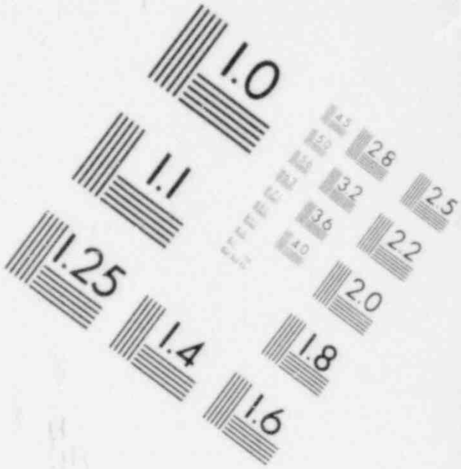
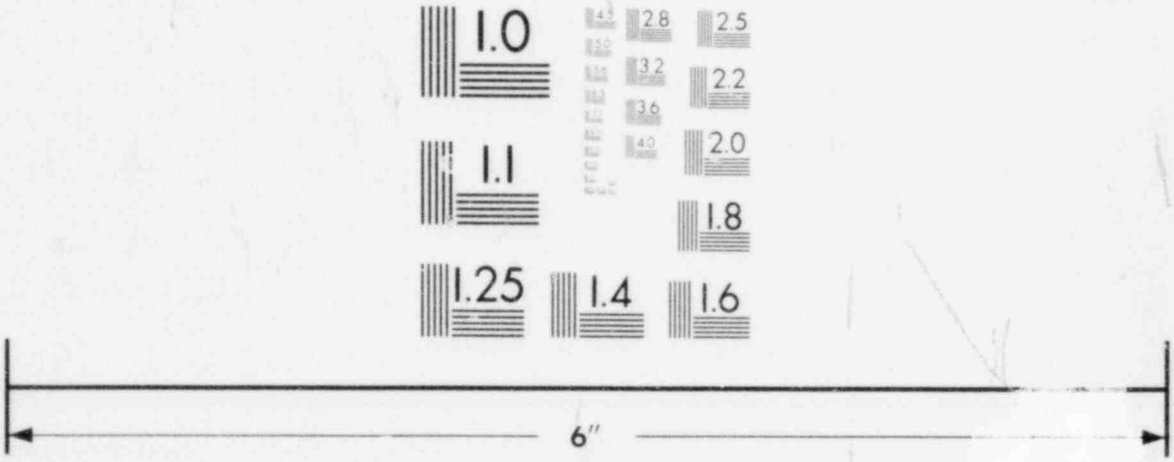
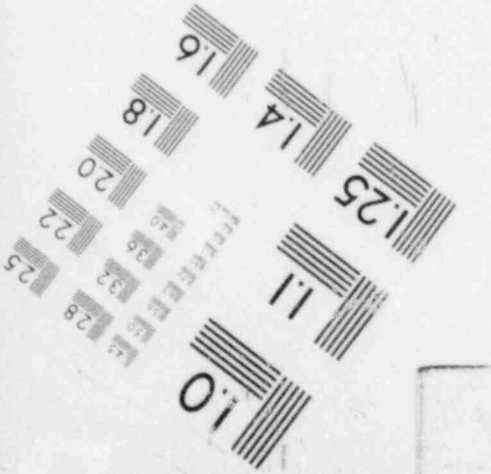


IMAGE EVALUATION
TEST TARGET (MT-3)



MICROCOPY RESOLUTION TEST CHART



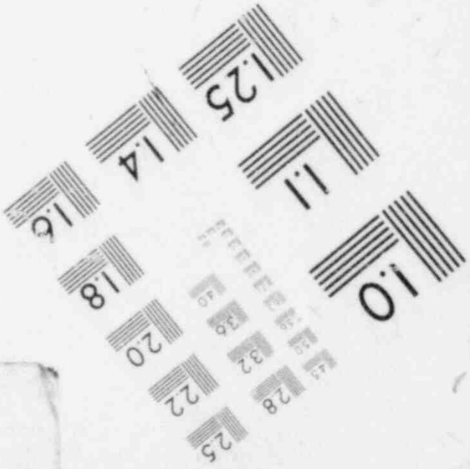


Table 4-8. Development requirements for automated fuel-fabrication plants[a]

Unit operation	Current status	Mixed-oxide requirements	Development status
Mixed-oxide receiving	Manual transfer and accountability	Mechanized transfer, accountability, and reseal	Cold laboratory
Feed-powder modification	Manual loading and unloading	Dust control, improved accountability	Cold laboratory
Slug and pellet compaction	Automated pressing	Dust control, improved accountability, transfer systems interfaces	Hot engineering
Sintering	Batch and continuous furnaces	Mechanized loading and unloading, accountability interface	Hot laboratory
Grinding and measuring	High dust	Dust control, automatic transfer, accountability interface	Hot laboratory
Rod fabrication	Manual load, decontamination, and welding	Mechanized loading, decontamination, and welding	Cold laboratory
Nondestructive assay	Manual transfer to stations	Mechanized transfer and examination	Cold laboratory
Bundle assembly	Manual assembly and inspection	Mechanized assembly, bundle redesign	Cold laboratory
Overall transfer system	Manual transfer	Mechanized transfer with accountability stations	Cold laboratory

[a] From Reference 4.

4.3 FAB-3: HIGH-RADIOACTIVITY ROD-TYPE FUELS

4.3.1 GENERAL DESCRIPTION

High-radioactivity rod-type fuels consist of fuel rods loaded with pellets or packed spheres of highly radioactive fissile or fertile materials which are assumed to be manufactured in facilities using remote operation and remote maintenance (hot cells). Feed materials from reprocessing would be recycled thorium nitrate solution, and/or fissile and fertile oxide or carbide powders or gel microspheres. The radioactivity of the feed materials is due to the gamma or neutron activity of uranium-233, high-burnup plutonium, or the cobalt-60 spikant. Make-up feed materials include depleted UF_6 and fresh thorium oxide.

The basic parameters for both pellet and Sphere-Pac fuel manufacture are summarized below. The model Fab-3 facility for processing high gamma activity fuel is assumed to process 480 MTHM/year. The plant parameters are presented in Table 4-9.

4.3.1.1 Feed Material Conversion and/or Preparation

Depleted UF_6 will be converted as described in Section 4.1.1a. New thorium oxide will be calcined to dryness and comminuted to a specific particle size. Facilities for these purposes will utilize contact operation and contact maintenance.

The depleted UO_2 powder is blended remotely, in a hot cell, with spiked recycle PuO_2 to the desired fissile assay. The blended powder may also be comminuted to a specific particle size before precompaction, granulation, and the addition of lubricant and binder. Granules are cold pressed, sintered, and ground to produce pellets for rod loading.

The fresh ThO_2 will be blended remotely with spiked PuO_2 powder or recycle uranium-233 oxide powder. Pellets are produced in the same manner as described for the Fab-2 facility.

Mixed oxides of $(Pu-Th)O_2$ or $(U-233, Th)O_2$ require only remote precompaction and granulation to prepare them as press feed. Dried gel microspheres of these materials are remotely calcined and sintered to density in a reducing atmosphere. Sintered oxide or carbide microspheres are separated into particle-size ranges and blended in the proper ratio of sizes to assemble batches for gravity loading into rods.

4.3.1.2 Pellet Production and Fuel Rod Manufacturing

A major process variation in the manufacture of pellets is the production of U-Pu or Pu-Th carbide pellets. The mixed-oxide powder is remotely blended with graphite and pressed into briquets. Following carbothermic reduction, the carbides are comminuted to the desired particle size under inert atmosphere to prevent reaction with oxygen or moisture. Carbide pellets are manufactured in the same manner as described for oxide pellets with the exception that lubricants, binders, and sintering aids are in the form of dry powders.

4.3.1.3 Bundle Assembly

Rod loading and bundle assembly are conducted as described in Section 4.2.1.3, except that all operations will be done remotely in hot cell facilities because of the radioactivity of the fissile and fertile feed materials. Rods are decontaminated before bundle assembly. All assembly operations and final inspection must be remotely conducted behind heavy shield walls.

4.3.2 SAFETY CONSIDERATIONS

The Fab-3 safety considerations are similar to those Fab-2 described in Section 4.2.2. Operations in hot cells, rather than the restricted volume of gloveboxes, tend to minimize the possibility of an explosion. However, if an explosion should occur, the hot cell integrity would be maintained and pressure relief will be through the ventilation system. The system would be designed to assure that the final HEPA filters would remain intact and the amount of plutonium, uranium-233, thorium, or cobalt-60 released would be about the same as for a fire.

The criticality and fire considerations would be the same as described in Section 4.2.2.

4.3.3 ENVIRONMENTAL CONSIDERATIONS

The Fab-3 hot-cell facility design will result in equal or less impact to the environment than the Fab-2 facility of the same throughput capacity. The specific impacts will differ slightly from those shown in Table 4-7 for the Fab-2 facility because of the different products under consideration.

Processing of spiked (Pu-U) oxide is expected to result in about 480 mg of mixed oxide released annually to the atmosphere (12 mCi) and 4.1 mg Pu to the water courses (2.1 mCi) as from a Fab-2 facility discussed above. However, these expected releases will also contain gamma activity from the spikant (or fission products) associated with these quantities. The chemical effluents and uranium releases are expected to be about the same as shown in Table 4-6 for the Fab-2 facility since the depleted uranium feed material will be in the UF_6 form.

Processing uranium-233 will result in different effluents since the feed uranium-233 will probably be as nitrate or oxide resulting in no calcium fluoride or hydrogen fluoride in the effluents. Also, there will be no plutonium released. Where (U-233-Th) oxide is processed, there will be more thorium in the effluents than shown in Table 4-6. The protactinium is not expected to increase since the time involved from reactor discharge to refabrication will result in essentially complete decay to uranium-233.

Processing (Pu-Th) oxides or carbides will result in different effluents since no uranium is present. There will be no calcium fluoride, hydrogen fluoride, or uranium in the effluents, but there will be increased quantities of plutonium and radioactive recycle thorium.

The estimated average annual dose commitment to the closest theoretical resident from a Fab-3 operation is estimated to be as presented in Table 4-10.

4.3.4 LICENSING STATUS[a]

Licensing requirements for a Fab-3 facility are undefined at the present time but will be similar to those encountered in licensing a mixed-oxide fuel facility (Fab-2) or a spent-fuel reprocessing plant.

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[a] See Reference 2 for more detailed information.

4.3.5 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

The major needs for fabrication of fuels based on uranium-233, recycle thorium, high burnup plutonium, and spiked plutonium are related to the requirements for remotely controlled operations and maintenance. Development work is needed to establish reliable materials-handling systems with unit operations interfaces. Surge-type inventory is necessary throughout the plant, and technology for placing and extracting material of various forms from inventory storage is required. Furthermore, it will be necessary to develop biological shielding techniques permitting personnel to operate and to perform remote equipment maintenance. The higher radioactivity of mixed uranium-233/thorium dioxide, recycle thorium, and spiked plutonium dioxide complicates analytical measurements for both process and safeguards needs. New methods of sampling and analysis will have to be developed for both process and safeguards needs. New methods will have to be developed and demonstrated for nondestructive assay and mechanical inspection.

There are uncertainties associated with the engineering of a remotely operated and maintained pellet-fabrication plant. More development of remotely controlled pellet-production processes and particulate processes (e.g., the Sphere-Pac process or vibrating compaction) will be required before a favored approach can be determined. Section 4.2.5 discusses research, development, and demonstration requirements for Sphere-Pac fuel.

The need for remote-handling techniques and equipment design would also have significant impacts in three important support areas: process control/sample analysis system, scrap-recovery systems, and offgas systems.

The sample-analysis techniques would be in large measure identical to those used for low-radioactivity uranium-plutonium fuels, and techniques have been or are being developed and defined to provide equivalent accuracy. However, the penetrating gamma radioactivity in the samples would require that all equipment be designed for remote operation. A suitable facility would also have to include shielded analytical cells for receiving the samples and performing some of the analyses.

The scrap-recovery system would be similar in structure to that for low-radioactivity uranium-plutonium fuels. The need for shielded remote operation would demand increased development.

Finally, the off-gas system from all areas dealing with exposed fuel materials would have to incorporate either radon traps or sufficient delay to ensure the decay of the short-lived radon in the uranium-232 decay chain before the usual filtration for particulate contaminants. The design of the plant would have to allow remote handling of these filters to prevent overexposure of maintenance personnel.

The development needs for mixed plutonium-thorium dioxide and carbide fuels include the study of the irradiation behavior of integral pins and the power-to-melt investigations. The chemistry of the ThO_2-PuO_2 fuel system is expected to be similar to that of UO_2-PuO_2. The effects on cladding wastage may be the most serious; cesium migration and the detrimental effects of chemical reactions may actually be reduced.

The development effort for the LWBR fuel indicates that the basic reference processes are adaptable to any thorium-based fuel system. Process development work is under way to evaluate possible benefits of an increased sintering temperature. The development requirements for mixed uranium-plutonium dioxide fuels outlined in

Table 4-8 also apply to mixed plutonium-thorium dioxide pellet fuels. However, the development status in most cases has not reached the cold-laboratory stages.

To fabricate rod-type fuel containing a spikant or residual fission products will require continued and expanded effort to develop automatic systems for remote pellet production, rod-loading, and bundle assembly, as well as final inspection techniques. The Gel-Sphere-Pac technology now under initial development may be more adaptable to rod-fuel remote manufacture, but development is needed to ascertain the effects of the various spikants or fission products on the chemical and mechanical characteristics of the feed material. There is no evidence that conventional fabrication processes, even if automated and remotely handled, can be used with spiked fuel and recycle thorium.

4.3.6 DECOMMISSIONING AND DECONTAMINATION

The discussions in Section 4.2.6 for Fab-2 facilities are applicable to Fab-3 facilities.

Table 4-9. Fab-3: Rod fuel fabrication model plant

Fuel materials	(U(5), Pu)O_2 w/Co-60, U(3) O_2, (U(3)-Th)O_2, (Pu-Th)O_2 w/Co-60, (Pu-Th)O_2 w/recycle Th
Mode of operation	Remote operation, contact maintenance
Production capacity	480 MTHM/yr
Design capacity	
3-rod production lines	240 MTHM/yr, each line
Fuel assembly line	730 MTHM/yr
Plant efficiency	67%
Operating schedule	24 hr/day, 7 days/wk
Principle of operation	Toll processing; sufficient fissile and fertile material provided by customer; materials other than heavy metals provided by fabricator
Feed materials	Depleted UF_6, new ThO_2, recycle thorium nitrate, calcined oxide powders, gel microspheres, carbide powders, carbide microspheres
Waste treatment	Wastes prepared and packaged as immobile solids
Scrap recycle	Clean and dirty scrap recycled in fuel-fabrication plant

Table 4-10. Estimated annual average dose commitments to the closest theoretical resident[a]

Organ	Resident at 500 m (mrem)
Lung	5.33×10^{-1}
Bone	7.3×10^{-3}
Whole body	1.0×10^{-3}
Gastrointestinal	--
Epithelium of bronchii	2.5×10^{-1}

[a] From Reference 10.

4.4 FAB-7: HTGR FUEL

4.4.1 GENERAL DESCRIPTION

The high-temperature gas-cooled reactor (HTGR) fuel consists of graphite blocks loaded with rod segments composed of fissile and fertile spherical particles that are coated with pyrocarbon with or without a silicon carbide barrier coat. Fresh-fuel fissile particles are uranium-235 dioxide spheres or uranium-235/thorium dioxide spheres. Recycle fissile particles are uranium-233 dioxide spheres, and all fertile particles are thorium dioxide spheres. Fresh-fuel fissile particles and all fertile particles can be processed in a contact-operation hooded facility; recycle fissile particles require a heavily shielded facility with remotely controlled operations and remotely controlled maintenance.

The basic process for the manufacture of HTGR fuel is described in detail in the literature (Ref. 11) and is summarized below. The model fresh-fuel fabrication plant for HTGR fuels is assumed to process 260 MTHM/yr; the model plant for recycle HTGR fuel is assumed to process 240 MTHM/yr.

4.4.1.1 Feed Material Conversion and/or Preparation

The HTGR fissile feed material for fresh (make-up) fuel is enriched uranium hexafluoride, which is converted to uranyl nitrate solution. The feed material for fertile fuel is received as thorium oxide or hydroxide and converted to thorium nitrate solution. These metal nitrate solutions are the feed for the microsphere production process. Recycle fissile fuel feed material is assumed to be received from the reprocessing plant in the form of dried gel microspheres.

4.4.1.2 Microsphere Production

The basic processes for microsphere formation are the weak-acid resin (WAR) process, the internal gelation process, and the external gelation process. The WAR process is restricted to highly enriched uranium ions loaded on presized ion-exchange-resin microspheres. The loaded resin is converted in a fluidized bed to low-density kernels containing uranium oxycarbide and some excess carbon. The kernels are sintered to high density before being coated with silicon carbide and pyrocarbon. The gelation processes are used to prepare solid-gel spherical particles directly from an aqueous solution of metal nitrates to which a gelling agent has been added. The gel particles are washed and heated to form the high-density metal oxide or mixed oxide/carbide fissile kernel. Fertile kernels of thorium dioxide are typically prepared by the Sol-Gel process, in which thorium nitrate solution is denitrated with steam to yield an aqueous sol of suspended thorium dioxide. The sol is transformed into microspheres by injection through a vibrating nozzle into a drying or gelling solvent. The gel microspheres are heated to produce dense fertile kernels.

The recycle fissile microspheres received from the reprocessor are calcined and sintered to dense kernels.

After inspection, acceptable fissile kernels are coated in a fluidized bed by the thermal decomposition of appropriate hydrocarbon and silane gases to produce an inner low-density carbon buffer layer, a high-density isotropic carbon inner layer, a high-density silicon carbide layer, and an outer layer of high-density carbon. Fertile particles typically have only a two-layer coating consisting of a low-density carbon buffer with a thick outer layer of high-density isotropic carbon.

Controlled quantities of fissile and fertile coated spheres, together with inert graphite particles, are blended and dispensed into a die to form a packed bed. The bed is infiltrated with a carbonaceous binder. This fuel stick or rod is ejected from the mold and inspected before fuel-element assembly.

4.4.1.3 Fuel-Block Assembly

The as-pressed fuel rods are assembled into fuel columns and inserted into the machined graphite fuel blocks together with appropriate burnable-poison disks. Each fuel column is capped at both ends with a graphite plug. The loaded block is heated in a high-temperature furnace in a controlled time-and-temperature cycle to remove the volatile components in the binder and plug cement and to convert the residue to carbon.

4.4.2 SAFETY CONSIDERATIONS

An HTGR fuel-fabrication facility is assumed to consist of two major processing lines; one to fabricate make-up fuel of new uranium-235 and thorium and the other to fabricate recycled uranium-233, uranium-235, spiked uranium-233, and spiked uranium-235. The plants are required by 10 CFR 70.22(f) to be designed, constructed, tested, and operated under rigid quality-assurance programs.

The fuels containing uranium-235 and thorium are fabricated in a low-gamma-activity, contact-operated and contact-maintained facility. Safety measures are the same as those discussed in Section 4.1.2. The hazard of fire is somewhat greater because of the use of hexanol or other alcohols for sphere drying and gelation and the use of highly flammable hydrocarbon gases for the deposition of pyrocarbon coatings. Facility design must incorporate additional protection for this hazard. In addition, medium-enrichment uranium hexafluoride is the feed material for make-up fuel, and care must be exercised to ensure positive criticality control.

The fissile fuel containing recycle uranium-233 and spiked uranium-235 must be fabricated in a shielded facility using remotely controlled operations and remotely controlled maintenance. The safety considerations discussed in Section 4.3.2 for plutonium fuels will apply.

4.4.3 ENVIRONMENTAL CONSIDERATIONS

The Fab-7 type plant for HTGR fuel manufacture is not expected to have a high impact on the environment. The contact facility for initial core and make-up fuel manufacture will have about the same impact as a Fab-1 facility. The remote facility for recycle fuel will have about the same impact as a Fab-3 type facility. The environmental effects of the contact facility are presented in Table 4-11. The airborne releases are small and the estimated average annual dose commitments to the closest resident are presented in Table 4-12. Data are not available for a remote facility; however, the land use, building size, water use, etc., are expected to be about the same as for the contact facility. Special effluent controls to limit the release of uranium-232 daughters, minute amounts of recycle thorium, and spikant materials may reduce the effuent volumes. Thermal releases are expected to be about the same as for a contact facility; however, additional processes to immobilize the highly radioactive wastes are expected to add to the effluents and thermal releases. The overall environmental impact will be very small compared to other parts of the nuclear fuel cycle such as mining, milling, enrichment, and reprocessing.

4.4.4 LICENSING STATUS[a]

The General Atomic Company has a licensed HTGR fresh-fuel pilot manufacturing plant at San Diego, California. Facilities can probably be designed to meet future regulatory requirements; the principal constraints will be cost increases and delays engendered by requirements to backfit facilities to meet changing regulations. Future facilities will be subject to the same extensive safety-analysis requirements for licensing as existing facilities.

From a safety and licensing standpoint, the fabrication of low- and medium-enrichment fuel-cycle alternatives for the HTGR present no basic differences from the reference high-enrichment uranium cycle.

It should be noted that recycled uranium-233 has significant gamma activity and radon emissions (Refs. 12 and 13). Most operations will need to be remotely controlled and highly automated. Licensing requirements will be similar to those encountered in licensing a mixed-oxide fuel facility or a spent-fuel reprocessing plant.

4.4.5 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

HTGR fuel (Ref. 11) consists of graphite and silicon carbide coated microspheres in a graphite matrix. The fuel fabrication and fuel refabrication are uniquely different from the corresponding portions of the liquid-metal or water-cooled reactor fuel cycle. The microspheric formulation developed for the HTGR program may also be applicable to the preparation of Sphere-Pac metal-clad fuel elements.

There has been extensive development of the HTGR fuel cycle, but it has not reached full-scale commercial production. The development of LWR fuels has, of course, advanced to commercial operation, and experience in this field will be valuable in the commercialization of HTGR fuels.

Development of fuel for HTGRs was started in the late 1950s and is continuing. Early in the development, it was decided that the most economic fuel system consisted of highly enriched (93%) uranium, with thorium for forming the bred fissile isotope uranium-233 and graphite to moderate the neutrons.

The qualification of recycle fuel for reactor use 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 recover and recycle refabrication plant reject (scrap) material and waste appears necessary to reduce the release of plutonium.

There has been extensive development both in private industry and in government laboratories for the recycle of uranium-233 recovered from spent HTGR fuels. Fuel refabrication processes and equipment have been studied in engineering-scale operations, with emphasis on the requirement that fuel be fabricated in shielded and remotely operated facilities because of the activity in the recycle fuel from uranium-232 daughter

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[a] See Reference 11.

products. In general, cold development of processes and equipment for fabrication/refabrication has taken into consideration scrap recycle, effluent control, and fuel-assay requirements, and fuel-assay methods have been tested. Additional research and development efforts are needed, however, particularly in the scaleup of prototype refabrication equipment, the recycle of scrap material, the control of effluents, the assay of fuel-containing materials, and the qualification of recycle fuel. Finally, hot demonstration is required.

Microspheres presently in HTGR fuel loadings were prepared by sol-gel techniques or uranium loading of weak-acid ion-exchange resins. Additional small-engineering-scale studies indicate that an internal gelation method may be more adaptable to the preparation of the wide range of microspheres being considered and requiring remote operation. This and possibly other fabrication techniques should be investigated before the decision is made to commercialize the fuel. These techniques must be adaptable to the fabrication of uranium-thorium and plutonium-thorium particles as well as the pure uranium, thorium, and plutonium particles.

The production of recycle fuel is in an early stage of development. The chemical and radiation effects of a spikant or retained fission products on the coated particles and the remote manufacture of HTGR coated fuel particles has not been demonstrated. There is inadequate experience that conventional (conceptual) spherical particle manufacture can be used with recycle or spiked fissile materials.

4.4.6 DECOMMISSIONING AND DECONTAMINATION

Separate facilities are expected to be used for fresh and recycle fuel. The fresh-fuel fabrication facility would have low gamma activity and would be relatively easy to decontaminate. Decommissioning a recycle-fuel fabrication facility contaminated with uranium-233 and possibly plutonium would result in quantities of highly radioactive TRU wastes. Much U.S. experience has been gained with the decontamination and dismantling of contact type pilot plutonium, uranium-233, and thorium (fresh only) processing facilities. The technique would be applicable to the Fab-7 remote facility. Also, experience gained in decontaminating the ORNL metal recovery plant and dismantling the Savannah River and Hanford chemical processing cells may be applicable to the Fab-7 recycle fuel plant. Regulatory criteria for decontaminating and decommissioning a Fab-7 facility processing uranium-233, plutonium, and recycle thorium have not been developed.

Table 4-11. Environmental effects of the Fab-7 model initial core and make-up fuel-fabrication plant[a]

Land use	
Plant and services, acres	150
Buildings	
Multiple, ft^2	225,000
Water use	
Process, gpd	28,000
Cooling, gpd	248,000
Power	
Electric MWe	20
Effluents, radioactive	
To atmosphere	
Thorium, g/day	1.181
Uranium, g/day	0.570
Thoron (Rn-220), μCi/day	1.68×10^5
To sewer	
Thorium, g/day	0.615
Uranium, g/day	0.0454
Thoron, μCi/day	0.72
Chemical effluents	
Gaseous, quantity/day	
Alumina, g	0.00084
Aluminum nitrate, g	5.15
Ammonia, ft^3	303
Ammonium hydroxide, g	719
Argon, ft^3	1.28×10^5
Ash, g	16.8
Carbon, g	155.9
Carbon dioxide, ft^3	7.94×10^4
Carbon monoxide, ft^3	236.3
Ethyl cellulose, g	2.8×10^{-4}
Hydrogen, ft^3	2.32×10^4
Hydrogen chloride, ft^3	31.8
Hydrogen fluoride, ft^3	16.6
Kerosene, ft^3	96.4
Lint, g	5.6
Nitric acid, ft^3	6.31×10^5
Liquid waste discharges[b]	
UC_2, ppm	1.37×10^{-4}
ThO_2, ppm	1.92×10^{-3}
Rn-220, μCi/l	1.97
Solids, ppm	158
Additives	47[c]

See footnotes at end of table

Table 4-11. Environmental effects of the Fab-7 model initial core and make-up fuel-fabrication plant (continued)

Thermal release	
Waste heat, BTU/min	734×10^3
Nitrogen, ft^2	6.31×10^5
Nitrogen dioxide, ft^3	47
Nitrogen oxide, ft^3	1.7
Oxalic acid, g	10.1
Phosphoric acid, g	10.1
Pitch, g	254
Potassium fluoride, g	61.6
Silicon carbide, g	0.85
Silicon dioxide, g	0.84
Sodium chloride, kg	1.81
Sodium fluoride, g	0.001
Sodium hydroxide, kg	0.172
Sulfur dioxide, ft^3	21.8
Sulfur trioxide, ft^3	1.7
Sulfuric acid, g	20.16
Tributyl phosphate, ft^3	0.6
Trichlorethylene, liters	2.5
Vermiculite dust, g	0.08
Water, gal	1.02×10^5

[a] From Reference 10.

[b] Depends on inhibitor.

[c] Daily effluent concentration, drinking water standard = 5 ppm uranylion, 500 ppm solids.

Table 4-12. Estimated annual average dose commitments to the closest theoretical resident[a]

Organ	Resident at 500 m (mrem)
Lung	5.33×10^{-1}
Bone	7.3×10^{-3}
Whole body	1.0×10^{-3}
Gastrointestinal	--
Epithelium of bronchii	2.5×10^{-1}

[a] From Reference 10.

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Chapter 5

REPROCESSING

Chemical reprocessing plants would include: (1) temporary spent-fuel storage at the reprocessing plant; (2) head-end treatment to separate mechanically and chemically the fuel material from other materials (e.g., hardware); (3) a selective extraction to separate uranium, plutonium, and/or thorium from fission-product wastes; (4) interim storage of recovered products; (5) conversion of recovered products from a liquid to a solid form; and (6) pretreatment of the waste to convert fission-product solution to a solid form for ultimate disposal in a waste repository. The discussions that follow cover all the above six items except for item (4), interim storage of recovered products, which is covered separately in Chapter 6. All products and wastes are required to be converted to solids on site prior to shipment.

The reprocessing options for the various reactor/fuel cycles are shown in Table 5-1 and as discussed below, include the following generic types of processes.

Purex 1	Rod-type fuel is reprocessed to recover separated streams of fission products, purified uranium, and purified plutonium, including the uranium-235 bearing portion of light-water breeder reactor (LWBR) duplex fuel.
Purex 2	Rod-type fuel is reprocessed to recover a stream of fission products, a stream of uranium, and/or a stream of plutonium-uranium (coprocessing).
Purex 5	Rod-type fuel is reprocessed as in Purex 2 but a spikant (such as cobalt-60) is added to the plutonium-uranium stream.
Thorex 1	Thorium-bearing spent fuel (rod-type and high-temperature gas-cooled reactor (HTGR)) is reprocessed resulting in a stream of fission-product wastes, a separated stream of uranium-233, and a separated stream of thorium. Small amounts of plutonium may be sent to waste or recovered as a purified stream.
Thorex 3	Uranium, thorium, and plutonium are recovered as separated, purified streams suitable for fuel materials for fabrication. The plutonium stream may be spiked with cobalt-60 before or during conversion to the oxide. The thorium and plutonium may be blended to dilute the plutonium. Fission products are separated as high-level wastes.

Not included in this chapter, because they were not part of the fuel-cycle system for reactors described in Volumes I through VI of the Preliminary Safety and Environmental Information Document (PSEID) are the following:

Purex 3	Same as Purex 2 (coprocessing) but with retention of part of the fission-product wastes in the plutonium-uranium stream.
Purex 4	Same as Purex 2 (coprocessing) but with preirradiation of the excess plutonium product not needed for immediate fuel refabrication. A stream of fission products is also produced for disposal as waste.

Thorex 2	Uranium-233 and thorium are recovered as a single stream but contain some fission products; a second stream of fission products is also produced for disposal as waste.
Thorex 4	Same as Thorex 3, except that the plutonium is sent to waste rather than recovered in a purified form for refabrication.
Thorex 5	Uranium, thorium, and plutonium are recovered in separated streams, but the uranium-233 is denatured in process at the head end by addition of depleted uranium, and the uranium-235 is sent to waste.
Thorex 6	Same as Thorex 1, except that (1) larger quantities of plutonium must be handled, (2) the uranium-238 bearing fuel is intimately mixed with uranium-233 fuel in such a manner that the recovered uranium-233 is denatured in situ when it is recovered in the purified uranium stream, and (3) purified uranium-235, if any, is sent to waste.
Thorex 7	Same as Thorex 3, except that plutonium is spiked after purification, and the in situ denatured uranium-233 recovered is diluted with uranium-235 and uranium-238.

5.1 PUREX 1 REPROCESSING: PARTITIONED URANIUM AND PLUTONIUM

5.1.1 GENERAL DESCRIPTION[a]

A Purex 1 fuel-reprocessing plant recovers uranium and plutonium from spent-fuel assemblies by mechanical head-end treatment and chemical treatment (Purex process). The fuels to be reprocessed by this option are shown on Table 5-1. The recovered plutonium and uranium products of purified nitrate solutions are converted to uranium hexafluoride or oxide powders for subsequent use. The separated fission products are recovered in the form of an acidic nitrate solution and later solidified and packaged for temporary storage at the site prior to disposal in a geologic repository (Figure 5-1). The model reprocessing plant is assumed to process 1,500 MTHM/yr. The model plant parameters are presented in Table 5-2.

5.1.1.1 Fuel Receiving and Storage

The spent-fuel assemblies arrive at the plant in shielded casks. The cask and carrier are monitored for external contamination and washed to remove external dirt. The cask is removed from the carrier, and the condition of the fuel and the cask is determined.

The cask is moved by the cask-handling crane to the cask-unloading area. The cask is opened and the fuel is removed. The identity of each fuel element is established before the element is placed in storage canisters and moved to the fuel storage area for retention until reprocessing.

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[a] See Reference 1 for a more detailed discussion.

5.1.1.2 Head-End Treatment

a. Rod-Type Fuel

Fuel assemblies are transferred from the fuel-storage area to the feed mechanism of the mechanical shear. They are chopped into small segments (from about 2 to 5 inches long) to expose the fuel. The fuel segments fall into the dissolver, which contains hot nitric acid (3 to 8 M) and gadolinium nitrate to dissolve virtually all uranium and plutonium and most of the fission products. Gases generated during dissolution are channeled to the off-gas treatment system to remove particulates, radioiodine, and nitrogen oxides. The dissolver solution is transferred to tanks for accountability sampling and final acid adjustment before being fed to the solvent-extraction process. The undissolved cladding hulls are rinsed, monitored for fissile material, packaged, and transferred to the solid-waste storage area.

b. Light-Water Breeder Reactor (LWBR) Rod-Type Fuel (Duplex Pellets)

Fuel assemblies consisting of rods loaded with duplex pellets (UO_2-CaO-ZrO_2 annulus, ThO_2 core) are transferred to the head-end processing area and disassembled. The cladding from each rod is assumed to be stripped to expose the pellets. The pellets are charged into the dissolver where the uranium tertiary compound is preferentially dissolved in hot nitric acid. The dissolver and rinse solution is transferred to accountability tanks and acid adjustment tanks. The solution is processed as described in a, above for dissolved light-water reactor (LWR) rod-type fuel. Dilution control is used as needed in processing high-burnup LWBR fuel with a high fissile content.

The residual ThO_2 core pellets and spacer pellets containing the bred uranium-233 are transferred to the Thorex-processing area and processed by the Thorex-1 flowsheet, discussed in Section 5.4.

c. Liquid-Metal Fast-Breeder Reactor (LMFBR) Rod-Type Fuel (Uranium-233 Oxide or Carbide)

The LMFBR fuel rods may contain pellets or Gel-Sphere-Pac oxides or carbides of recycle uranium-233. The fuel rods will have been sealed with either helium or sodium heat transfer media in the rods. The head-end treatment for helium-bonded rods containing oxide pellets is the same as for LWR rod-type fuel discussed in a, above. Sodium-bonded rods containing either oxides or carbides will require chopping in an inert atmosphere containing wet steam to remove the sodium. Rods containing carbide pellets or carbide spheres will also require a voloxidation step to convert the carbides to oxides before dissolution.

5.1.1.3 Product Separation and Purification

After acid adjustment, the dissover solution is filtered or centrifuged and sent to solvent extraction, where it passes countercurrently against an organic solution of tributyl phosphate (TBP) in normal paraffin hydrocarbon diluent (primarily dodecane). The organic solution preferentially extracts the plutonium and uranium, leaving about 99% of the fission products in the aqueous phase. The organic solution is scrubbed with a concentrated nitric acid solution that removes (back-extracts) about 96% of the fission products that were extracted into the organic solution. The combined aqueous waste stream contains about 99.8% (or more) of the fission products and is sent to the high-level-waste concentrator.

The scrubbed organic solution is partitioned to separate the plutonium from the uranium. The aqueous stream containing the plutonium flows to the plutonium solvent-extraction purification cycles. After the final plutonium cycle, the plutonium nitrate solution is washed with pure hydrocarbon diluent to remove traces of TBP and is evaporated in a concentrator. The plutonium product solution is analyzed and stored in geometrically favorable tanks prior to later conversion to plutonium oxide.

The uranium is stripped from the organic solution and flows to the uranium solvent-extraction purification cycles. After the final cycle, the uranyl nitrate solution is concentrated by evaporation. The concentrated solution is passed through silica-gel beds to remove residual traces of fission products. The uranyl nitrate product solution is analyzed and stored as a nitrate solution for possible future conversion to uranium hexafluoride or for recycle to fabrication as either nitrate solution or oxide powder.

5.1.1.4 Product Conversion

The products from a Purex 1 reprocessing plant are uranyl nitrate solution and plutonium nitrate solution. The uranyl nitrate may be low-assay uranium-235 ($<20\%$), high-assay uranium-235 ($>20\%$), or denatured uranium-233 ($<12\%$ U(3)).

The low-assay uranium-235 to be re-enriched or stored will be converted to uranium hexafluoride (UF_6). The UF_6 will be packaged in steel cylinders, which would probably be identical to those used to store unirradiated uranium, their capacity varying with the uranium-235 assay. The low-assay uranium-235 to be shipped to fabrication is assumed to remain as uranyl nitrate solution. The high-assay uranium-235 is assumed to be converted to UO_2 and a spikant (probably cobalt-60) added to provide proliferation-resistant material.

The denatured uranium-233 probably will be converted to oxide and shipped to fabrication in shielded containers or placed in a secure shielded storage facility.

The plutonium nitrate may be converted to PuO_2, placed in secure storage, or blended with a spikant, probably colbalt-60, before storage or shipment to a fabricator. Low-assay uranium-235 (depleted UF_6, natural uranium or recovered uranyl nitrate) may be blended with unspiked plutonium and converted to oxide to provide a fissile assay below 20%. The diluted $(Pu\text{-}U)O_2$ would then be packaged and stored in a secure facility.

5.1.1.5 Waste Pretreatment

The radioactive aqueous waste streams from all the solvent-extraction cycles are concentrated in the high- or low-level waste concentrators, depending on the radioactivity content. The acidic concentrator bottoms are stored in a cooled stainless-steel waste tank. The concentrator overheads are passed through a distillation column to recover the nitric acid. The distillation-column overheads (essentially water) are recycled as process water or sampled and released to the stack from a vaporizer. The recovered acid is used in the process, where the residual radioactivity can be tolerated.

The stored high-level liquid wastes (HLLW) may be evaporated to a high concentration to be used as feed to a continuous melter. Glass frit is added and a continuous pour of molten glass is cast into canisters or molds. An alternative would be the evaporation of the HLLW to dryness followed by calcination and either pot or continuous melting with a glass frit. Other processes under consideration and development are the "cermet" and "synrock" processes. In any event, the immobilized HLLW will be stored under water or in a shielded vault until shipment to a geologic repository.

Other wastes requiring treatment are--

- cladding hulls and fuel hardware
- intermediate-level transuranic waste
- low-level liquid waste
- low-level solid waste

Cladding hulls and fuel hardware are packaged and seal-welded in stainless-steel drums for shipment in shielded overpacks to a geologic repository. Intermediate-level transuranic wastes are solidified or compacted for shipment to a geologic repository. Low-level liquid wastes are solidified by evaporation and fixed in concrete, urea formaldehyde or bitumen, and packaged. Low-level solids are incinerated or compacted. Low-level wastes are packaged in drums for shipment to a shallow land burial site.

The conversion of depleted UF_6 to UO_2 may result in some calcium fluoride requiring packaging and disposal in a shallow land burial site with other low-level wastes.

5.1.2 SAFETY CONSIDERATIONS

5.1.2.1 Occupational Exposure (Ref. 2)

The confinement of radioactive material and shielding are the primary means of protecting workers from radiation. To ensure adequacy, shielding design for existing plants is conservatively based on high-burnup plutonium recycle fuel and potentially abnormal process conditions.

To determine occupational exposure, plant-operation-worker and laboratory-worker exposures were estimated on the basis of the exposure conditions set by design bases, the personnel needed to carry out various operations, and the types of activity and associated occupancy time in various work zones. On the assumption that the average exposure of maintenance personnel and other service personnel, except administrative, would equal the average exposure of operating personnel, the estimated annual-average occupational exposures for a model reprocessing plant are as shown in Table 5-3. The estimates are scaled from those given in Reference 2 for a 2,000-MTHM/yr plant. The occupational-exposure estimates are based on normal operations, including relatively minor incidents. They do not include exposures that might result from steps that would be taken to decontaminate an area in the event of a major accident, nor do they include exposures that might be associated with the decontamination of facilities and equipment related to the repair, modification, or replacement of process components or systems after unexpected equipment failures.

5.1.2.2 Accidents (Ref. 2)

The most severe accidents that may occur at separations or plutonium dioxide conversion plants include the following:

1. Criticality
2. Explosion in the high-level-waste concentrator or calciner
3. Explosion in the plutonium-product concentrator
4. Fire

The dose commitments from the hypothetical accidents that might occur in the fuel-separations section and product-conversion section of the plant have been calculated for the maximally exposed individual (Table 5-4). The maximum individual bone dose from the hypothetical plutonium-concentrator explosion is 11 mrem. The maximum bone-dose commitment to an off-site individual is about 19% of the annual dose from natural background radiation. The amount of plutonium released would be equivalent to about 0.38% of the release limit specified in 10 CFR 20.106(a) for an unrestricted area.

a. Criticality Accident

A criticality accident is unlikely in a separations or plutonium dioxide conversion plant, because equipment and process limitations are designed to prevent such incidents. Safe spacing is ensured in storage basins by physically spacing the fuel elements in storage racks in a safe array. Process systems and controls are designed to prevent an unsafe condition. Nevertheless, a criticality accident of 10^{19} fissions is assumed. This yield is approximately an order of magnitude greater than the yield that has been experienced for plutonium systems in past accidents. It is further assumed that all noble gases and 50% of the halogens (or halides) are discharged from the plant stack.

The dose commitment to the thyroid (56 mrem to an individual) is much greater than the dose commitment to other organs.

b. Waste-Concentrator Explosion

At the separations plant, the solvent-extraction process generates solvent degradation products, which may be carried over into the waste streams. Under extreme conditions in early pilot-plant operations, these nitrated degradation products (red oil) have caused concentrator explosions. Red-oil explosions can be prevented, however, by installing equipment to prevent the accumulation of organic materials in the waste and by controlling the process temperature in the concentrator.

Concentrators are installed in highly shielded cells, having a volume of about 100,000 cubic feet (3,000 cubic meters). In the unlikely event of an accident, the explosion is estimated to disperse about 150 gallons (600 liters) of high-level-waste solution into the cell in the form of a finely divided mist. A substantial portion of the mist would rain out or plate out on the surfaces of the cell. Droplets remaining in the air (10 mg/m^3) would be carried through the ventilation ducts to the high-efficiency filters. Moisture separators upstream from the filters would remove most of the mist.

The plant ventilation filters are located some distance from the separations-plant process cells. Most of the explosive energy would be expended on the destruction of the concentrator. Pressures developed by the explosion would be damped by expansion into the cell and would be further attenuated in the ductwork. The final filters are not expected to be affected.

It has been estimated that plateout of the droplets on the walls and floors of the cell and removal by the filtration system will reduce the fraction of material released to 3.6×10^{-8}. It has been estimated that the material leaving the final filter would be 30.5 mg of high-level-waste solution in the form of an aerosol.

Table 5-5 identifies the nuclides that would contribute significantly to the off-site dose and summarizes the off-site bone-dose commitment that would result from this hypothetical accident. The maximum off-site dose commitment to an individual is estimated to be about 2.6 mrem (bone) for uranium dioxide fuel. This is an increase of only about 3 to 7% of the dose received from natural background radiation.

c. Plutonium-Concentrator Explosion

The postulated explosion of a plutonium concentrator in the reprocessing plant is typical of upper-level accidents that could release plutonium to a cell or glovebox area. Typically, the plutonium-processing equipment tends to be smaller and to be installed in smaller rooms (cells or gloveboxes) than the waste concentrator previously discussed. The release rate is derived by assuming that the room (cell or glovebox) atmosphere contains the same mass of aerosol (10 mg/m^3) as the atmosphere of the waste-concentrator cell. For a 1,000 cubic-meter plutonium-concentrator cell, the postulated accident would release about 2.2 mg of plutonium.

Table 5-6 shows the radionuclide releases and the bone-dose commitment to the maximally exposed off-site individual. The off-site dose commitment is less than 11% of that received from natural background radiation.

d. Fire

A range of other accidents is possible, but the evaluations show smaller effects than those tabulated above. Among these are the following:

1. First-cycle solvent fire
2. Ion-exchange-resin fire (plutonium cycle)
3. Second-cycle solvent fire (plutonium cycle)

For example, estimates have been made of the potential maximum individual dose commitment (off site) from first- and second-cycle solvent fires. The estimated bone-dose commitment is 1.6 mrem.

e. Other Accidents

Listed below are examples of other types of accidents that are not expected to result in significant off-site contamination or significant exposure to the public.

Abnormal event	Potential consequences
Head-end	
Fuel-cask drop into cask unloading pool	Possible rupture of fuel rods and release of fission gas to atmosphere; contamination of storage-pool water
Fuel element hung up in air during transfer to shear	Possible overheating and rupture of fuel rods and release of fission gas to atmosphere
Ignition of zirconium fires	Small fire of short duration--little, if any, damage
Rapid chemical reaction in dissolver	Vessel pressurized, seals blown, fission gas released to atmosphere; cell contaminated

Separation

Leak in recovered-acid line	Contamination of cell or pipe trench in area of leak
Leak of any vessel or line confining radioactive material	Contamination of cell or pipe trench
Excessive entrainment of radioactivity in evaporator overheads	Contamination of recovered acid or condensate that must be recycled
Failure of an iodine scrubber	Reduced iodine-removal efficiency
Filter failure	Detectable increase of radioactivity in stack effluent
Loss of ventilation zone differential-pressure control	Possible migration of radioactivity from controlled area to limited-access area
Solvent fire	May plug filter and contaminate cell and ventilation or exhaust ducts

Uranium hexafluoride conversion

Hydrogen fluoride tank leak	Air concentration of toxic hydrogen fluoride could be lethal to unprotected persons in vicinity of tank

Product conversion

Glove failure	Local contamination of operating area
Window failure	Local contamination of operating area
Equipment leak or release of product to glovebox	Contamination of glovebox; transfer material to installed spare storage tank

5.1.3 ENVIRONMENTAL CONSIDERATIONS

A summary of environmental considerations relating to a model reprocessing plant of 1,500-MT/yr capacity is presented in Table 5-7. Land use, water use, and power requirements are small. The major environmental effect of reprocessing compared with other fuel cycle steps is the radiological dose. Derivations of dose commitments are described in Chapter IV, Section J, of Reference 2.

5.1.3.1 Population Dose Commitment from Normal Operations

Table 5-8 gives the estimated population dose commitment from a model plant reprocessing uranium dioxide fuel.

Estimates of the population dose commitment are affected most by the release of tritium, carbon-14, and krypton-85 to the atmosphere during reprocessing operations that results in an increased exposure to the world population of 5 billion people.

To put these exposure estimates into perspective, the world population receives about 500 million man-rem each year from natural background radiation. A 1,500-MT/yr model plant reprocessing plutonium recycle fuel (89% uranium dioxide fuel + 11% uranium/plutonium dioxide fuel) may cause a 0.006% increase in the radiation exposure of the world population. The additional whole-body dose commitment for the U.S. population of approximately 250 million persons would be about 18,600 man-rem, or 0.07% higher than the dose from natural background radiation.

5.1.3.2 Dose Commitment for the Nearest Exposed Resident (Normal Operations)

It is assumed that the nearest resident lives about 1,300 meters (about 0.8 mile) from the plant's stack. The estimated dose for the nearest exposed resident represents a 50-year dose commitment to an individual living at this location for 1 year. Table 5-9 shows the estimated dose commitment from a 1,500-MT/yr model plant that reprocesses uranium dioxide fuel. The dose commitment to the whole body of the nearest resident exposed to the reprocessing plant effluents would be about 5.7% higher than the dose received from natural background radiation.

5.1.4 LICENSING STATUS

Three commercial fuel-reprocessing facilities have been constructed under U.S. Nuclear Regulatory Commission (NRC) licenses, but only one of these was actually operated, the Nuclear Fuel Services (NFS) plant at West Valley, New York. The NFS plant, as constructed and operated, would not meet current regulations. The General Electric plant at Morris, Illinois, was not successfully commissioned, and licensing proceedings on the Barnwell Nuclear Fuel Plant (Allied-General Nuclear Services) in North Carolina have been suspended. Licensing proceedings for the construction of a fourth plant (Exxon Nuclear Corporation) were begun but were later suspended by Federal order. Safety analyses and environmental reports for the latter two facilities were prepared by the applicants, but final actions have not been taken by the NRC. Generic environmental and safety evaluations of both facilities have been prepared by the NRC.

The licensability of Purex reprocessing facilities will be affected by the necessity to reduce, by 1985, off-gas emissions of krypton-85 and iodines to the environment. In the future, reduced emissions of tritium and carbon-14 probably will be required to reduce population exposures. Another constraint is the need to assess the environmental impacts (radiological) of immobilizing and encapsulating high-level and cladding wastes from reprocessing plants. An assessment of a waste-solidification facility that could be located at the Barnwell plant indicates that radiological impacts would be acceptable under current regulations.

5.1.5 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

Areas of the Purex 1 process that require additional development work include the following:

1. Shearing of spent fuel, especially with regard to the remote maintenance of the associated equipment
2. Methods for handling the highly radioactive residues that remain after the dissolution of high-burnup fuel
3. The chemistry of plutonium during the reprocessing of high-burnup fuel
4. Technology for reducing radioactive off-gas releases (e.g., krypton-85, iodines, tritium) to conform to current regulations
5. Technology for immobilizing high-level, intermediate-level, and gaseous wastes
6. Remotely prepared and directly maintained conversion processes for plutonium from power reactor fuels
7. Methods for immobilizing and encapsulating wastes, particularly high-level and cladding wastes

If carbide fuels are to be reprocessed, the head-end treatment process and off-gas systems will require development to control the carbon-14 and to prevent spread of contamination during the oxidation step.

Sodium-bonded fuels will also require development of head-end processes to remove the sodium before the fuel is dissolved. In addition, the sodium hydroxide waste will require development of waste-processing operations to handle and dispose of the radioactive sodium.

5.1.6 DECOMMISSIONING AND DECONTAMINATION

No commercial reprocessing facility has been decommissioned. The General Electric plant has not been contaminated with irradiated fuel and does not require decontamination except in the fuel-storage-pool area, where several hundred tons of spent fuel are stored. The NFS plant has been partially decontaminated in preparation for the planned modification to expand throughput (since canceled).

Reprocessing facilities become contaminated in the course of processing and waste-treatment operations. On retirement, these facilities become a waste that requires managing by decommissioning. Of the various alternatives available, two basic decommissioning modes have been considered: immediate dismantling and safe storage with deferred dismantling. Regulatory criteria are being developed.

If immediate dismantling is selected as the mode of decommissioning, at shutdown all radioactive contamination above regulatory limits is removed from the facility to an approved disposal site. Depending on further uses of the fuel-cycle facility site, noncontaminated portions of the facility remaining after dismantling may be demolished and removed.

If safe storage with deferred dismantling is selected, preparations are made to leave the facility in place for an extended time. Continuing surveillance and maintenance are required, and either passive safe storage or hardened safe storage may be used. For passive storage, the radioactivity in the facility is isolated at shutdown by installing temporary physical barriers. When the radioactivity levels have been sufficiently reduced by decay, the facility is usually dismantled. For hardened storage, radioactive materials in the facility are isolated at shutdown by installing hardened physical barriers. The facility is maintained in this condition until dismantlement or until all residual radioactivity has decayed to nonhazardous levels.

During initial plant design, consideration should be given to the ultimate decommissioning of the plant to minimize the cost of decommissioning. In addition, information on decommissioning must be submitted to the NRC at the time of initial application for facility licensing.

Decommissioning activities include chemical decontamination, mechanical decontamination, and equipment deactivation and removal.

5.1.6.1 Chemical Decontamination

The extent to which chemical decontamination is used depends on a variety of factors, including the radiation-dose levels, the potential effectiveness of chemical decontamination; the amount of chemical decontamination equipment installed in the facility; and the availability of facilities for treating, packaging, and disposing of the chemical decontamination solutions. Decontamination solutions may include corrosive acids, detergents, and high-pressure water or steam. The decontamination solutions may be applied remotely with installed equipment or manually with portable equipment.

5.1.6.2 Mechanical Decontamination

Contaminated surface layers are generally removed from structures. It is usually necessary to remove some noncontaminated materials to ensure that all contamination is removed. Stainless-steel structural components or lines may be removed by sectioning in place with plasma torches, arc saws, or explosives. Contaminated concrete can be removed with explosives, by drilling and rock-splitting, or by jackhammering. Explosives are usually preferred for removing contamination from large concrete surfaces. Jackhammers or hand tools are generally used in small areas. Rock splitters may be used for moderate-size areas or in large areas where explosives are not desirable. These techniques may be used remotely or by direct contact, depending on the radiation levels in the area being decontaminated.

5.1.6.3 Equipment Deactivation and Removal

Equipment is deactivated by removing bulk quantities of process materials or other hazardous substances, closing valves or installing blank flanges, and disconnecting electricity and other utilities. Extensive equipment-removal operations are not required for safe storage, but for dismantling, all radioactively contaminated equipment is removed. The equipment-removal techniques used depend on the location and construction of the equipment and the radiation levels near the equipment. Stainless-steel equipment can be removed by using oxyacetylene torches, plasma torches, or arc saws; power hacksaws or explosive cutting techniques may also be used. Remote-removal techniques are employed in high-radiation areas, with installed remote-maintenance equipment being used where available. Long-handled tools, portable shielding, or specially constructed remotely controlled equipment may also be used. Normal maintenance techniques and demolition techniques (adapted as necessary for radiation areas) are used in low-level areas.

Table 5-1. Reprocessing modes used for various reactor/fuel cycle options

Reactor/fuel cycle	Purex 1	Purex 2	Purex 5	Thorex 1	Thoxex 3
Volume--I LWR					
2-1[a]	NA[b]	NA	NA	NA	NA
3-1	NA	NA	NA	NA	NA
4-1	X	--	X	--	--
5-1	--	--	--	--	X
6-1	--	--	--	--	X
Volume II--HWR					
2-1		NA	NA	NA	NA
Volume III--LWBR					
2-6	--	--	--	X	--
2-7	--	--	--	X	--
3-5	X	--	--	X	--
3-6	--	--	--	X	--
4-2	--	--	--	X	--
4-3	--	--	--	X	--
5-4	X	--	--	X	--
5-5	--	--	--	X	--
Volume IV--HTGR					
2-2	NA	NA	NA	NA	NA
3-1	X	--	--	X	--
4-1	X	--	--	X	--
5-1	--	--	--	X	--
Volume V--GCFR					
2-3	--	X	--	X	--
Volume VI--LMFBR					
2-1	--	X		--	--
3-1	--	X	--	--	--
4-1	--	X	--	--	--
5-1	--	X	--	X	--
6-1	--	X	--	X	--
7-1	--	--	--	X	X
8-1	X	--	--	X	--

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[a] Figure numbers of fuel-cycle mass-flow diagrams in PSEID Volumes I through VI.

[b] NA = Not applicable (once-through cycle).

Table 5-2. Model Purex 1 reprocessing plant parameters

Plant capacity, MTHM/yr	1,500
Operating time, days/yr	300
Accountability	Real time (no campaigns)
Fuel cooling time, days before reprocessing	180
Fuel receipt and storage	
Type	Pool
Capacity, MTHM (90 days)	450
Head-end capacity, MTHM/day	
Size reduction (shearing or crushing and burning)	
Design	6
Normal operation	5
Tritium removal (veloxidation)	
Design	6
Normal operation	5
Dissolution	
Design	6
Normal operation	5
Solvent extraction	
Capacity, MTHM/day	
Design	5.5
Normal operation	5
Storage capacity for $UO_2(NO_3)_2$, $PuNO_3$, and/or $Th(NO_3)_4$, MTHM (30 days)	150
Product-conversion capacity, MTHM/day	
Fluorination to UF_6	
Design	6
Normal operation	5
Denitration to oxide powder	
Design	6
Normal operation	5
Internal gelation to spheres	
Design	6
Normal operation	5
Product storage, MTHM (30 days)	150
Off-gas treatment	
Gases removed	I_2, NO_x, ^3H, Kr, Rn, $^{14}CO_2$
Iodine removal	$Hg(NO_3)_2$ scrub, silver zeolite polish
NO_x removal	Absorb in H_2O, catalytic destruction with NH_3
Tritium removal	Oxidize to ^3H_2O, adsorb on molecular sieve
Radon removal	Adsorb on molecular sieve for decay holdup
$^{14}CO_2$ removal	Scrub with lime slurry, fix as $CaCO_3$
Krypton removal	Absorb in refrigerant, concentrate, store

Table 5-2. Model Purex 1 reprocessing plant parameters (continued)

Off-gas treatment (continued)	
Capacity	
Design	Equivalent to reprocessing 5.5 MTHM/day
Normal operation	Equivalent to 5 MTHM/day
Particulate removal	HEPA filters as required
Waste treatment	
Extent of treatment	Fix as immobile solids for ultimate disposal
Processing capacity, MTHM/day	
Design	5.5
Normal operation	5
Storage capacity, years	
High-level waste	5
Intermediate-level waste	5
Reserve capacity	One spare tank always available
Solidified-waste storage capacity, days	30
High-level solid-waste treatment	Fix in concrete
Low-level trash disposal	Land burial
Low-level liquid-waste volume	Assume 100 m^3/MTHM processed

Table 5-3. Average annual occupational exposure for Purex 1 plant

Facility	Number of persons[a]	Exposure (man-rem)
Separations	150	300
Uranium hexafluoride conversion		
Plutonium conversion	169	300-375[b]
Waste solidification and packaging	112	225
Total	431	940-1,000

[a] Does not include administrative and support personnel who are not engaged in activities related to handling radioactive materials.

[b] Allowance for exposure related to handling a greater quantity of plutonium associated with plutonium recycle.

Table 5-4. Environmental impact of accidents at a Purex 1 uranium dioxide fuel-reprocessing plant

Maximum offsite individual dose commitment[a]

Accident	Dose (mrem)	
	UO_2 fuel	MOX fuel
Criticality	56	56
Waste-concentrator explosion	2.6	6.9
Plutonium-concentrator explosion	11	19
Fire	1.6	13.5

[a] Critical organ for all accidents except criticality is bone. Critical organ for criticality accident is thyroid.

Table 5-5. Effects of waste-concentrator explosion
(Purex 1 uranium dioxide fuel reprocessing)

Nuclide	Half-life	Radioactivity released in accident (mCi)		Accident bone-dose contribution (mrem)	
		UO_2 fuel	MOX fuel	UO_2 fuel	MOX fuel
Plutonium		0.02	0.15	0.01	0.06
Americium-241	433 years	0.007	0.05	0.02	0.17
Curium-242	162 days	0.76	12.3	0.04	0.60
Curium-244	18 years	0.05	3.25	0.07	4.38
Strontium-90	29 years	2.80	1.62	1.92	1.14
Ruthenium-106	369 days	2,000	3,400	0.02	0.04
Cerium-144	284 days	27	23	0.04	0.03
Other fission products		1.5	1.5	0.48	0.48
				2.6	6.9

Table 5-6. Effects of plutonium evaporator explosion
(Purex 1 uranium dioxide fuel reprocessing)

Nuclide	Half-life (years)	Radioactivity released in accident (mCi)		Accident bone-dose contribution (mrem)	
		UO_2 fuel	MOX fuel	UO_2 fuel	MOX fuel
Plutonium-238	88	1.02	2.11	5.94	12.27
Plutonium-239	24,000	0.08	0.04	0.51	0.28
Plutonium-240	6,540	0.11	0.13	0.76	0.88
Plutonium-241	15	2.94	4.01	3.98	5.42
				11.2	18.9

Table 5-7. Summary of environmental considerations related to reprocessing: 1,500-MT/yr model Purex 1 plant reprocessing uranium dioxide fuel[a]

Natural resource use	
Land, acres	
Temporarily committed	1,800
Permanently committed	200
Total	2,000
Water, 10^6 gal/yr	
Discharged to air	275
Discharged to water	2,325
Total	2,600
Fossil fuel (per year)	
Electrical energy, 10^3 MW-hr	175
Equivalent coal, 10^3 MT	63
Fuel oil 10^6 gal	10
Propane gas, 10^3 gal	90
Effluents	
Chemicals, MT/yr	
Gases	
Sulfur oxides	230
Nitrogen oxides	940
Hydrocarbons	20
Carbon monoxide	18
Particulates	25
Fluorides	2
Liquids	
Sulfate	<1
Nitrate	--
Chloride	>4
Sodium and potassium	<1
Iron	--
Thermal, 10^9 Btu/yr	
Fossil fuel	2,025
Decay heat	1,200
Total	3,225
Radiological, Ci/yr	
Gases (including entrained matter)	
Tritium	772,500
Carbon-14	1,020
Krypton-85	1,710,000
Strontium-90	0.24
Ruthenium-106	6.10
Iodine-129	1.43
Iodine-131	5.6

Table 5-7. Summary of environmental considerations related to reprocessing: 1,500-MT/yr model Purex 1 plant reprocessing uranium dioxide fuel[a] (continued)

Other fission products, Ci/yr	7.5
Uranium	1.67×10^{-3}
Plutonium (alpha)	0.04
Plutonium (beta)	0.95
Americium-241	6.10×10^{-4}
Americium-243	4.83×10^{-5}
Curium-242	6.37×10^{-2}
Curium-244	4.58×10^{-3}
Population dose commitments, man-rem	
Occupational	940
U.S. population	14,100
Foreign population	9,150
Total	24,190

[a] Data scaled from 2,000-MT/yr model plant (Ref. 2).

Table 5-8. Population dose commitment from Purex 1 uranium dioxide fuel reprocessing[a,b]

Nuclide	Population dose commitment (man-rem)							
	Whole body	GI tract	Bone	Liver	Kidney	Thyroid	Lung	Skin
Krypton-85	860	860	860	860	860	860	1,875	73,275
Tritium	8,325	8,325	8,325	8,325	8,325	8,325	8,325	8,325
Carbon-14	4,750	4,750	23,700	4,750	4,750	4,750	4,750	4,750
Iodine-129 and iodine-131	43	31	23	23	23	11,100	25	23
Other fission products	125	6,820	820	159	263	75	101	71
Uranium and transuranics	19	1.5	840	93	78	--	23	0.75
U.S. total	14,100	20,775	34,575	14,175	14,250	25,125	15,075	86,250
Foreign total	9,150	9,150	29,325	9,150	9,150	9,150	9,150	336,750
World total	23,250	29,925	63,900	23,325	23,400	34,275	28,800	423,000

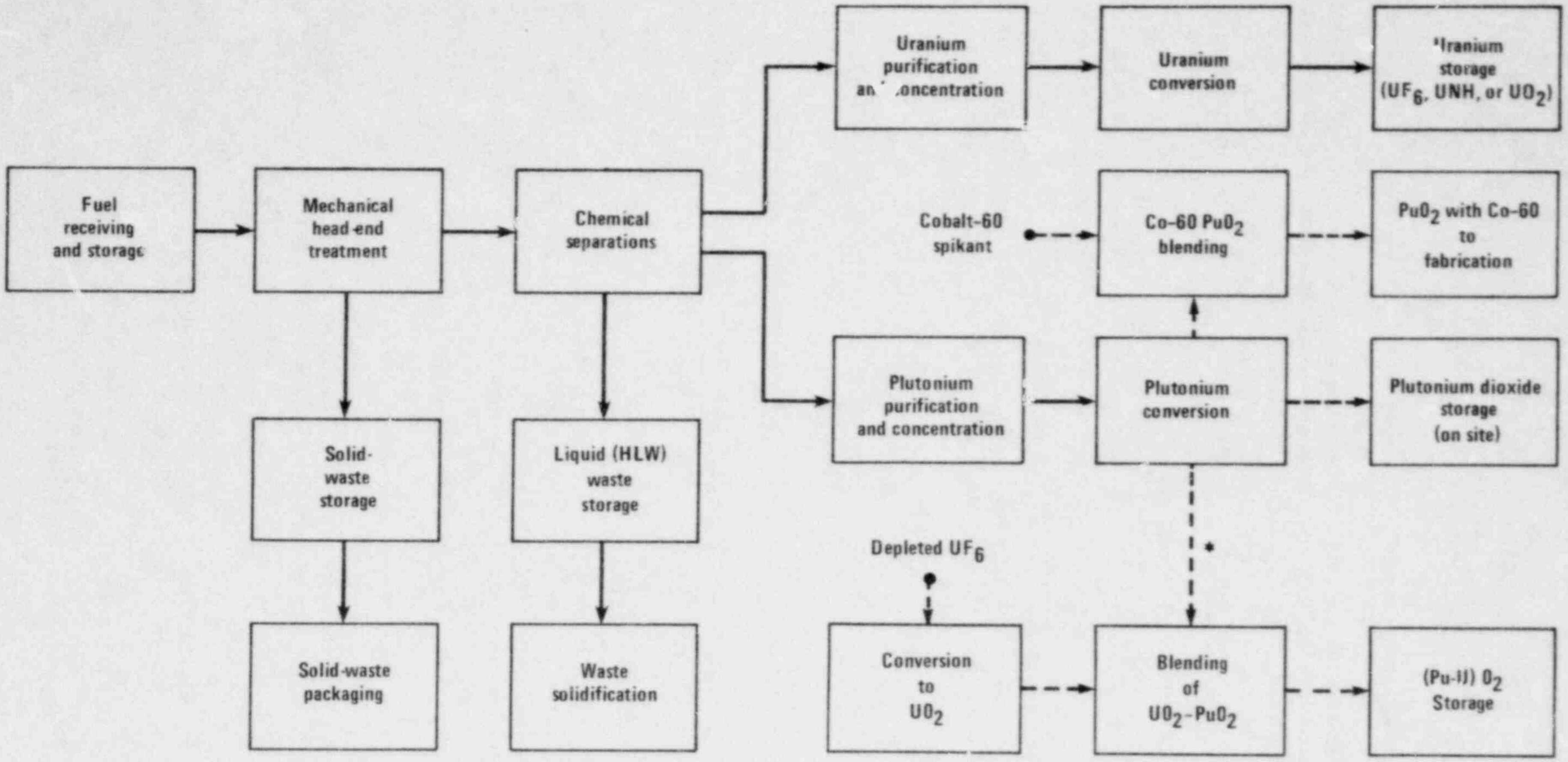
[a] Basis: 1,500-MT/yr fuel-reprocessing plant; fuel with 33,000-MWd/MT burnup, aged 160 days; 100% release of tritium, carbon-14, and krypton-85 to air.

[b] Natural background will cause 500-million man-rem/yr dose to a 5×10^9 world population.

Table 5-9. Dose commitment to nearest resident from Purex 1 uranium dioxide fuel reprocessing[a]

Nuclide	Dose commitment (mrem/yr)							
	Whole body	GI tract	Bone	Liver	Kidney	Thyroid	Lung	Skin
Krypton-85	0.32	0.32	0.32	0.32	0.32	0.32	0.70	27.2
Tritium	3.17	3.17	3.17	3.17	3.17	3.17	3.17	3.17
Carbon-14	1.19	1.19	5.91	1.19	1.19	1.19	1.19	1.19
Iodine-129 and iodine-131	0.22	0.17	0.14	0.14	0.14	43.3	0.15	0.14
Other fission products	0.62	25.1	3.26	0.75	1.13	0.42	0.59	0.42
Uranium and transuranics	0.11	--	4.98	0.55	0.46	--	0.14	--
Total	5.63	29.9	17.8	6.12	6.41	48.4	5.94	32.1

[a] Basis: 1,500-MT/yr fuel-reprocessing plant (2); fuel with 33,000-MWd/MT burnup, aged 160 days; 100% release of tritium, carbon-14, and krypton-85 to air; $X/Q = 3.7 \times 10^{-8}$ sec/m^3.



* Alternate processes.

Figure 5-1. Purex 1 flow diagram: partitioned uranium and plutonium.

5.2 PUREX 2 REPROCESSING: COPROCESSING; URANIUM AND/OR URANIUM/PLUTONIUM PRODUCTS

5.2.1 GENERAL DESCRIPTION

The fuels to be reprocessed by this option are shown on Table 5-1. In the Purex 2 coprocessing option (Figure 5-2), fissile material from spent fuel is recovered and utilized without ever producing a purified plutonium product (Ref. 3). By not separating the plutonium from the uranium in a Purex 1 reprocessing plant, a measure of protection against diversion of purified plutonium is afforded to all subsequent fuel-cycle operations. However, a plant designed to operate in a coprocessing mode can be made to yield chemically pure plutonium by making fairly minor adjustments to process conditions such as flow rates, temperatures, acidity, and the addition of plutonium reductant.

Coprocessing can be effected in the reference Purex system by appropriately varying the partitioning conditions in the solvent-extraction step. This necessitates modifying the product of the reference system. For this purpose, there are two possibilities: (a) coconversion of the uranium/plutonium product solution (by the Coprecal process, for example) and (b) the formation of gel microspheres.

5.2.1.1 Fuel Receiving and Storage

The fuel receiving and storage is the same as described in Section 5.1.1.1 for the Purex 1 process.

5.2.1.2 Head-End Treatment

The head-end treatment for gas-cooled fast reactor (GCFR) rod-type fuel, LMFBR oxide fuel, or carbide fuel is the same as described in Section 5.1.1.2.

5.2.1.3 Product Separation and Purification

Uranium and plutonium can be coextracted and costripped by appropriate modification of the conditions in the first two solvent-extraction cycles. In the third and final cycle, the partitioning contactor is operated in such a fashion that all of the plutonium and uranium is stripped into an aqueous stream. The product is highly decontaminated, and only the radioactivity normally associated with the several isotopes of plutonium is present in significant quantity.

5.2.1.4 Product Conversion--Coprecal Process

The two product-conversion steps applicable to Purex coprocessing are coprecipitation and slurry calcination (Coprecal process) and gel-microsphere production (described in Section 5.2.1.5). The Coprecal process consists of the following four basic steps:

1. Addition of ammonium hydroxide to concentrated plutonium/uranyl nitrate solution to produce a finely divided slurry of ammonium diuranate and plutonium hydroxide
2. Calcination of the entire slurry in a fluidized-bed unit
3. Reduction of the calcined powder in an atmosphere of 6% hydrogen in nitrogen at elevated temperatures to produce uranium/plutonium dioxide
4. Treatment of the reduced powder with hot carbon dioxide gas to stabilize the powder with respect to reoxidation in air

The stabilized uranium/plutonium dioxide powder is screened to product specifications, inspected, packaged, and sent to the refabrication plant.

5.2.1.5 Product Conversion--Microsphere Formation

Sintered gel microspheres may be suitable for loading into fuel pins with the use of low-energy vibratory compaction. This process offers potential advantages over conventional coprocessed powder conversion. These advantages include simpler processing steps, assurance of fuel homogeneity, and elimination of dusty process operations.

Sphere conversion consists of the following three major steps:

1. Preparation of a special solution called "broth"
2. Gelation of broth droplets
3. Washing and drying to make gel microspheres suitable for shipment to a fuel-refabrication plant

5.2.1.6 Waste Pretreatment

Radioactive waste streams from a Purex 2 type reprocessing facility will be similar to those described in Section 5.1.1.5 for a Purex 1 facility.

5.2.2 SAFETY CONSIDERATIONS

The safety considerations described in Section 5.2.1 for a Purex 1 plant would apply to all Purex plants. The additional facilities for product conversion must be evaluated for criticality and radiation exposure (to meet the "as low as reasonably achievable" criterion).

5.2.3 ENVIRONMENTAL CONSIDERATIONS

The environmental considerations described in Section 5.1.3 for a Purex 1 facility would also apply to a Purex 2 facility. The facilities for product conversion, especially coprecipitation and slurry calcination to produce uranium/plutonium dioxide powder, will introduce the need for dust control and criticality control to prevent radioactive releases in excess of regulatory limits.

Shielding calculations (Ref. 4) show that actinide activity is similar for Purex 1 and Purex 2 (coprocessing). For coprocessing, the fission-product activity remains at the same level from recycle through recycle. The coprocessed uranium contributes little to the overall dose rate.

5.2.4 LICENSING STATUS

Licensing applications for a Purex 2 facility have not been submitted to or reviewed by the NRC. The discussions in Section 5.1.4 should apply. Added considerations are necessary for the amended flow sheets and the Coprecal mixed-oxide conversion facilities or microsphere-production facilities.

5.2.5 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

Coprocessing solvent extraction has all of the technical problems of the reference Purex process (Section 5.1.5) and, in addition, would require careful optimization of all flow rates and mass balances and extremely careful process control (Ref. 3). Conversion of the coprocessed product to a form acceptable for fabrication would also require development. The Coprecal and gel-microsphere formation processes described in Section 5.2.1 are currently under development.

5.2.6 DECOMMISSIONING AND DECONTAMINATION

The general description of decommissioning in Section 5.1.6 applies to a coprocessing Purex 2 plant.

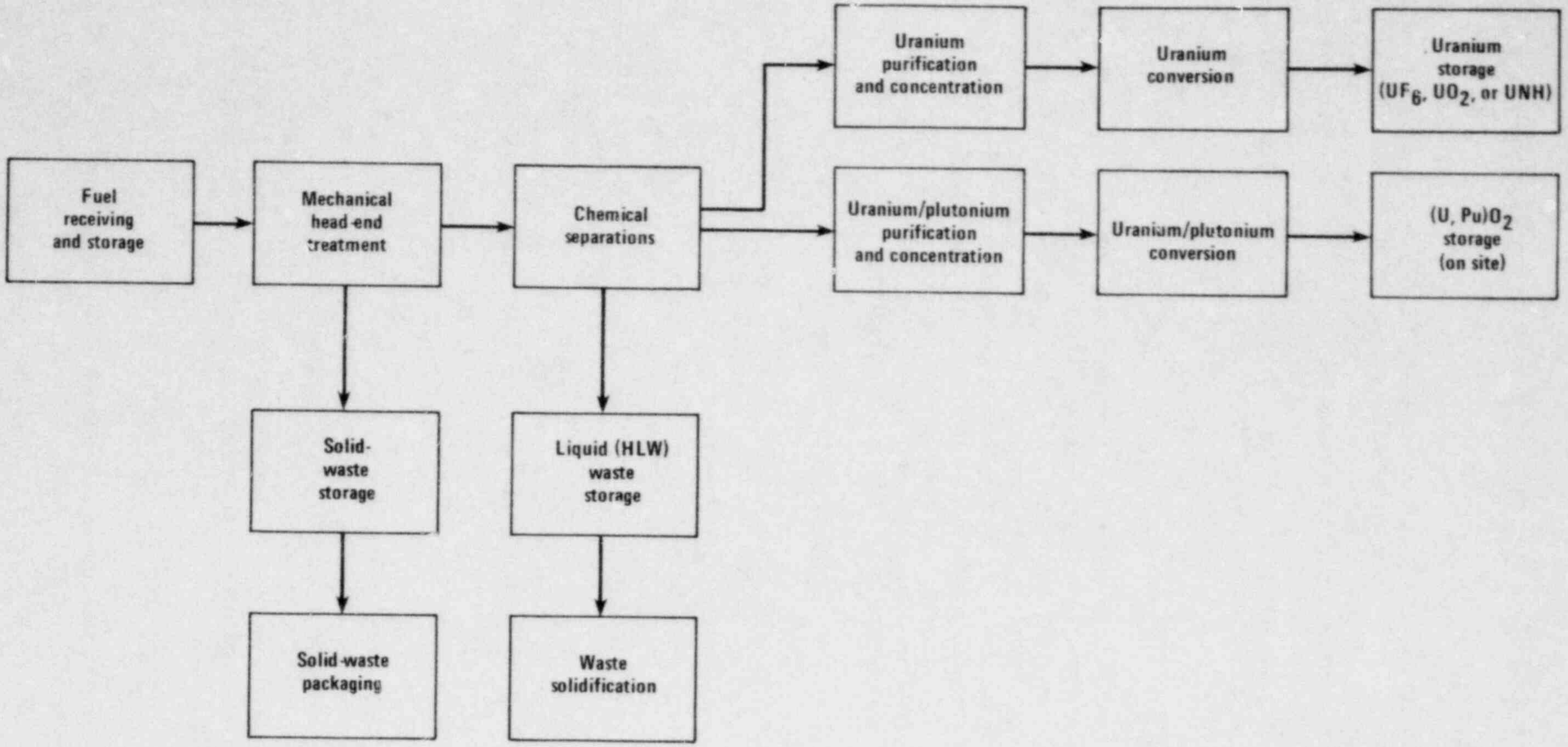


Figure 5-2. Purex 2 flow diagram: coprocessing; uranium/plutonium products.

5.3 PUREX 5 REPROCESSING: COPROCESSING WITH ADDED SPIKANT

5.3.1 GENERAL DESCRIPTION

The Purex 5 process (Ref. 3) is the same as the Purex 2 process described in Section 5.2.1 but modified to include the addition of a spikant, such as cobalt-60, to the plutonium-bearing product stream in the solvent-extraction coprocessing step. The fuel to be reprocessed by this option is shown on Table 5-1. The spikant is added in sufficient quantity to produce a dose-rate increase of about four orders of magnitude greater than that for Purex 2 coprocessed product material. The uranium/plutonium nitrate solution and the powder or sphere product are spiked to produce about 100 R/hr/kg HM at a distance of 1 meter. The Purex 5 flow diagram is shown in Figure 5-3.

5.3.2 SAFETY CONSIDERATIONS

The safety considerations described for the basic Purex 1 reference process (Section 5.1.2) and the Purex 2 process (Section 5.2.2) apply to the Purex 5 process as well. In addition, the receipt, storage, and handling of the spikant material before it is added to the coprocessed mixed nitrate solution will increase the cumulative occupational exposure of operators and maintenance personnel. The safety implications of potential overexposure from concentrated quantities of high-activity material must be evaluated. In addition, all product handling must be conducted remotely, and heavy shielded containers must be used for shipping the spiked product. These operations increase the potential hazards to plant operators.

5.3.3 ENVIRONMENTAL CONSIDERATIONS

The environmental considerations described in Section 5.2.3 for the coprocessed-product scheme will apply to the Purex 5 reprocessing as well. In addition, the receiving and handling of highly radioactive spikant material as a solid or as a solution and the accident scenarios that can be postulated for receiving, handling, and storage will need to be evaluated for potential impacts on the environment. A release of this material to the atmosphere could have an impact greater than that from releases of the spiked product. In addition, the spikant and product may be in soluble form, whereas the spiked oxide would be essentially insoluble. If the process requires a solid-form spikant compound to be dissolved in plant equipment, the off-gas from this operation would need to be evaluated for the environmental impact of a potential release.

5.3.4 LICENSING STATUS

No large-scale Purex 5-type reprocessing plant has ever been operated or licensed. Neither Allied-General Nuclear Services nor the Exxon Nuclear Corporation has applied for a coprocessing license, and the NRC has dropped consideration of all applications for a reprocessing license. A Purex 5 plant would face the same licensing problems as would a Purex 2 plant, with the additional requirements of demonstrating the safety of special equipment and operations for receiving, storing, and handling highly radioactive spikant material.

5.3.5 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

In addition to the development requirements of all of the other conceptual Purex plants, including the Purex 1 base process, Purex 5 coprocessing will need development work in the laboratory as well as in pilot and demonstration plants to work out the techniques of adding a spikant and the methods of handling and storing the spikant material. It will also be necessary to determine the quantity and chemical form that are feasible and desirable for use, the equipment required, the extent of equipment maintenance, the environmental releases during normal operations and under postulated accident conditions, and inspection equipment and techniques for ensuring product homogeneity and operational safety.

Laboratory development will be needed to determine the magnitude of spikant needed to give a desired level of radioactivity in the product, such as the proposed levels in Appendix A. Development is needed on a process and a procedure to handle and package the product in approved shipping containers, which will also need to be developed and licensed.

5.3.6 DECOMMISSIONING AND DECONTAMINATION

The general description of decommissioning in Section 5.1.6 applies to a coprocessing Purex 5 plant as well.

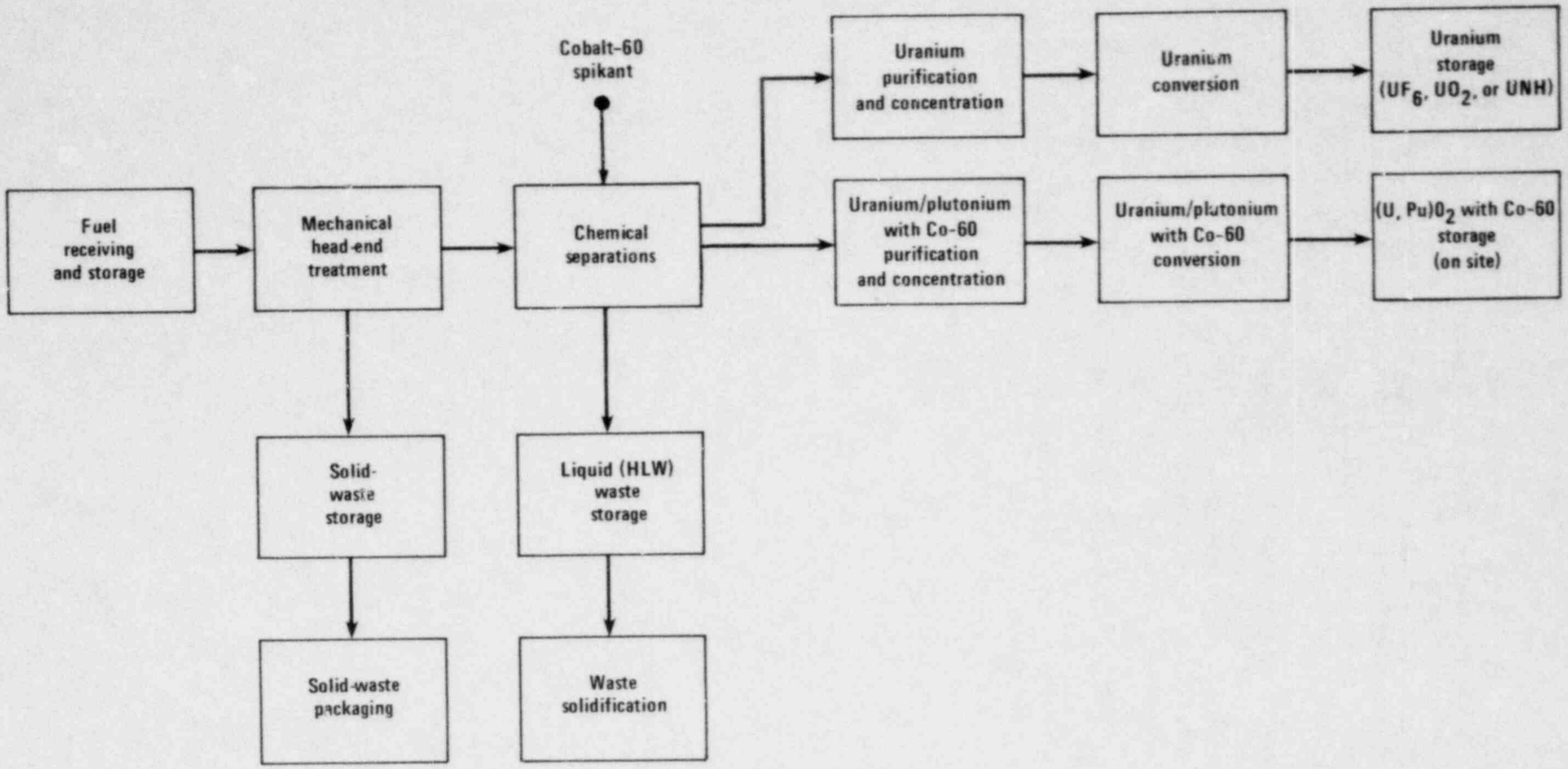


Figure 5-3. Purex 5 flow diagram: coprocessing; uranium and/or uranium/plutonium product, spiked with cobalt-60.

5.4 THOREX I REPROCESSING: URANIUM AND THORIUM RECOVERY[a]

5.4.1 GENERAL DESCRIPTION

A Thorex I fuel-reprocessing plant recovers uranium and thorium from spent fuel assemblies by a head-end treatment and a chemical treatment (Thorex process). The recovered uranium and thorium products of purified nitrate solutions are converted to oxide powders or gel spheres for subsequent use. Thorium to be stored is assumed to be a solid (see Chapter 6). The separated fission products are concentrated and stored on site in the form of an acidic nitrate solution and later solidified and packaged for disposal in a geologic repository.

In the Thorex process (Figure 5-4) the fuel is dissolved in nitric acid containing a small quantity of fluoride ions.

The model Thorex I separations plant is essentially identical to a Purex I plant except possibly for alternate process materials of construction to resist fluoride corrosion. In a Thorex I plant for HTGR fertile-material reprocessing, the head-end and waste handling are different from a Purex facility. The plant is assumed to process 1,500 MTHM/yr.

5.4.1.1 Fuel Receiving and Storage

a. Rod-Type Fuel

The spent-fuel assemblies arrive in shielded truck or rail casks. The cask and carrier are monitored for external contamination and washed to remove external dirt. The cask is removed from the carrier and the condition of the fuel and the cask is determined. The cask is vented, cooled if required, and filled or flushed with water.

The cask is moved to the unloading pool, opened, and the fuel removed under water. Fuel-element identity is established before the element is placed in storage canisters for transfer to the storage pool. The stored fuel is retained until scheduled for reprocessing.

b. HTGR Fuel

The spent fuel arrives by truck cask or rail cask. The casks are transferred to a hot cell for opening and unloading of the containers of fuel. Fuel is expected to be stored in these sealed containers in air or in a spent-fuel pool until scheduled for reprocessing. An alternative may include the opening of the shipping containers, removal, and inspection of each fuel assembly, and encapsulation of the assemblies in storage containers. These containers could be stored in a spent-fuel pool to permit return of the shipping containers in the spent-fuel cask to the nuclear power plant.

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[a] Data from References 5, 6, and 7.

5.4.1.2 Head-End Treatment

a. Rod-Type Fuel

Fuel assemblies containing ThO_2 pellets, UO_2ThO_2 pellets or gel microspheres of oxides or carbides of thorium or Th-U will be sheared into segments. Sodium-bonded fuel must be pretreated as discussed in Section 5.1.1.2 e. Carbide microspheres will be oxidized prior to dissolution. The segments will be leached in hot nitric-hydrofluoric acid. The solution is filtered and adjusted for acidity before entering the product-separation process.

b. HTGR-Type Fuel

HTGR fuel blocks contain fissile particles of uranium dioxide or uranium carbide and fertile particles of thorium oxide or carbide. The fuel blocks are crushed and ground to a size suitable for feed to the fluidized-bed burners. The graphite and pyrocarbon fuel-particle coatings are burned. The ash is leached in hot nitric acid to dissolve the uranium and the plutonium fissile particles leaving the fertile particles. The solution of uranium and plutonium is transferred to accountability tanks for sampling and final acid adjustment before being fed to the Purex solvent-extraction process. (See Section 5.1.) The fertile particles of thorium oxide containing uranium-233 are dissolved in hot nitric-hydrofluoric acid. The solution is adjusted for acidity and is fed to the Thorex extraction process for separation of the uranium-233, the thorium, and the fission products.

5.4.1.3 Product Separation and Purification

After acid adjustment, the feed solution is filtered or centrifuged and sent to solvent extraction where it is processed using 5 to 30% TBP in a diluent similar to kerosene. The solution is contacted by this counterflowing immiscible organic solvent to extract the uranium-233 and thorium from the bulk of the fission products. The fission products and transuranics remain in the aqueous phase and flow to waste treatment. The coextracted products are contacted in a second extraction cycle by an aqueous scrub solution to partition the uranium and thorium. The products are treated separately by one or two cycles of solvent extraction for additional fission-product removal. Each cycle produces an aqueous waste solution that must be processed to immobilize all radioactive constituents. The uranium-233 is processed through a third extraction cycle to remove the residual fission products, neptunium and plutonium, resulting in a purified uranyl nitrate product. The small amount of plutonium is routed to the high-level waste stream. The thorium is in the form of a purified nitrate solution which is still highly radioactive.

5.4.1.4 Product Conversion

The products from a Thorex 1 reprocessing cycle are purified solutions of uranyl nitrate and thorium nitrate. The recovered uranium-233 would be converted to solids in the form of oxide for recycle or storage. The highly enriched oxide may be stored or sent to fabrication as is or spiked with an appropriate spikant, possibly cobalt-60, before shipment to a fabricator. Storage would be controlled under secure conditions. Another alternative for proliferation protection would be the dilution of the $HEU(3)O_2$ with depleted UO_2 to reduce the fissile assay to <12%.

The thorium nitrate is assumed to be stored in shielded containers if not immediately required for recycle. Otherwise, the solution will be converted in a shielded hot cell to ThO_2. The oxide is recycled to a fabricator. An alternative, the Gel-Sphere-Pac process, would require the blending of the uranium-233 and thorium nitrate solutions and the formation of dried gel spheres of (U-Th) oxide for shipment to a fabricator.

5.4.1.5 Waste Pretreatment

Radioactive aqueous waste streams from a Thorex 1 fuel facility will be similar to those described in Section 5.1.1.5 for a Purex 1 facility except the fission-product and decay-product inventory will be different. The raffinate from the solvent extraction cycles are concentrated and stored in cooled stainless-steel tanks to allow decay of the high-heat producers. After cooling, the waste will be immobilized by vitrification or other such processes, such as calcination, cermet manufacture, or synrock manufacture, that may be approved as acceptable for geologic disposal. The immobilized waste in sealed canisters will be stored until shipment. The conversion of depleted UF_6 to UO_2 for denaturing of uranium-233 may result in some calcium fluoride waste requiring packaging and disposal in a shallow land burial site with other low-level wastes.

Radioactive gases generated during the head-end treatment (burning, crushing, and leaching) are channeled to the off-gas treatment system for segregation and treatment to remove krypton-85 and carbon-14, as well as particulates, radioiodine, and nitrogen oxides. The krypton may be absorbed in liquid CO_2 (KALC process). The gases may have to be converted to a solid form for disposal in a geologic repository.

5.4.2 SAFETY CONSIDERATIONS

A Thorex reprocessing facility has similar safety considerations to those described in Section 5.1.2 for a Purex 1 facility for handling uranium-235 and plutonium. In processing HTGR fuels, the off-gas treatment system for krypton, radon-220, and carbon-14 must be considered.

5.4.2.1 Occupational Exposure

A Thorex 1 facility, including conversion facilities for uranium-233 and recycle thorium require the handling of much more radioactive products and actinide wastes because of (1) the presence of uranium-232 in the uranium-233 produced by nuclear reactions with thorium-230, thorium-232, and uranium-233 resulting in a number of energetic gamma-ray emissions, and (2) because of the presence of radioactive thorium-228 in the recycle thorium. The actinide wastes will have gamma-ray emissions from uranium-232 requiring shielding not normally required for actinide wastes. The estimated average occupational exposure for a Thorex 1 plant is expected to be similar to, but possibly higher than, that shown in Table 5-3 for a Purex 1 plant due to the increased possibility for exposure in handling more radioactive material and in maintenance operations on hot-cell equipment.

Listed in Section 5.1.2.2 are examples of the types of accidents postulated for irradiated-fuel separation and conversion facilities. For a Thorex 1 facility, instead of plutonium dioxide conversion and release of plutonium, which are not applicable, the plant must deal with thorium and uranium-233 conversion and release. Those accidents listed for Purex 1 are expected also to apply to Thorex 1 and are not expected to result in significant off-site contamination or exposure to the public.

The more severe accidents include:

1. Criticality
2. Explosion in the high-level waste concentrator or calcinator
3. Explosion in the product concentrators.
4. Fire

The dose commitment from a Thorex I facility accident has not been calculated but the dose commitments for uranium-233 or recycle thorium release are expected to be equivalent to, or on the same order of magnitude as, those described in Section 5.1.2.2 for plutonium release.

5.4.3 ENVIRONMENTAL CONSIDERATIONS

At present there is very little actual experience relevant to the commercial reprocessing of thorium-based fuel assemblies. In general, the problems to be expected in the design, construction, and operation of reprocessing plants for spent thorium-based fuel will be of similar magnitude to those for a facility reprocessing spent uranium/ plutonium recycle fuels. Much work will be needed on effluent controls and cleanup systems for such plants to meet effluent release restrictions such as those set by the U.S. Environmental Protection Agency in 40 CFR 190 for normal operations in the uranium-fuel cycle.

Environmental analyses of some HTGR fuels indicate that reductions of radon-220 and carbon-14 emissions in the off-gas will be necessary to meet current regulatory standards. Because of the short half-life of radon, off-gas holdup may reduce radon releases, but retention capacity and subsequent disposal methods for the long-lived isotope, carbon-14, need to be identified. Another constraint on reprocessing plants is the requirement for much remote-control operation and maintenance to keep occupational radiation exposures within allowed limits. Indeed, these limits may be further reduced in the future.

The environmental impacts of the fluorides used in Thorex processes, and also in UF_6 conversion and in enrichment were discussed in ERDA 1541, Final Environmental Statement, Light-Water Breeder Reactor Program, Volume IV. The radioactive effluents associated with reprocessing of LWBR thorium-based fuels were identified, and their radiological impacts were assessed in the same report.

In general, the environmental, health, and safety issues currently associated with the reprocessing of thorium fuels are not expected to introduce significant new problems. Adequate control of emissions and occupational exposures will be required for commercial development just as similar issues were or must be resolved for the full-scale commercialization of uranium-fuel reprocessing.

5.4.4 LICENSING STATUS

The first commercial mixed uranium/thorium dioxide fuel used in an LWR (Indian Point) was reprocessed at the Nuclear Fuel Services, Inc., plant at West Valley, New York, under its existing license in 1970-1971.

In anticipation of the need for larger quantities of uranium-233 to be used for a demonstration of thorium-fueled reactor concepts such as the HTGR or the LWBR, approximately 870 tons of thorium (mostly as thorium dioxide) were irradiated; this thorium was then processed in unlicensed U.S. Atomic Energy Commission (AEC) production-scale equipment to recover 1.5 tons of purified uranium-233. The large-scale programs used either the Thorex flowsheet or a modified version of it (i.e., the acid Thorex flowsheet) to effect the separation and recovery of uranium-233 and thorium.

The licensing of a Thorex-reprocessing facility should not result in problems exceeding those of a Purex facility. A plant for reprocessing HTGR fuel would have the added requirement of containing and controlling carbon-14, which could have some licensing implications.

5.4.5 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

In the mid- to late 1950s, the AEC Thorex Pilot Plant Demonstration Program had been completed, and 35 tons of irradiated thorium metal had been processed in a plant having a throughput of 150 to 200 kg of thorium per day to recover 55 kg of purified uranium-233. The principal emphasis was on processing, demonstrating the Thorex flowsheet, defining process capabilities, and identifying problem areas in the reprocessing of spent thorium fuel.

Studies on the HTGR thorium fuel cycle have resulted in the development of preliminary designs and cost estimates of a large-scale demonstration facility. Differences between HTGR and LWR fuel reprocessing are found mainly in the head-end treatment. Much of the experience in solvent extraction, off-gas treatment, and waste treatment for the metal-clad fuel-element recovery is applicable to HTGR fuel. Some of the additional effort that would be required to commercialize thorium fuel cycle reprocessing is listed below.

5.4.5.1 Head-End Treatment

a. Rod-Type Fuel

The thorium fuel cycle rod-type fuel will receive the same head-end treatment as rod-type fuel for Purex reprocessing. The rods will be sheared into segments and the fuel contents dissolved, leaving the cladding hulls for disposal as waste. Duplex fuels consisting of annular pellets of either uranium dioxide or ternary uranium-zirconium-calcium oxides surrounding a core of thorium dioxide would require sequential dissolution treatment to separately dissolve the annulus and the core. The annulus would be processed using the Purex-1 flowsheet; the core would be processed using the Thorex-1 flowsheet. Other fuels consisting of solid solutions of thorium and uranium oxides would be processed using only the Thorex-1 flowsheet.

b. HTGR Fuel

The head-end treatment of HTGR fuel blocks is to expose the fuel and convert the fuel to oxide, with subsequent dissolution of the fuel. These operations--with emphasis on primary burning, separation of fertile and fissile particles, fuel leaching, crushing of silicon carbide coatings, secondary burning, and transfer of granular and powdered solids--require additional research and development in prototype equipment. The crush/burn/leach operations have been demonstrated in engineering-scale and cold prototype-scale equipment.

The burning of the graphite generates large quantities of carbon dioxide, which may have to be contained because of the carbon-14 content; carbon fixation as calcium carbonate is the preferred containment process at this time, and laboratory-scale tests indicate that fixation could be performed before krypton removal. The burning operation also has the potential of releasing tritium before aqueous processing, although the conditions for adequate release need further study, and some tritium will be released in the dissolver. Since the burner technology is relatively new and has not been used with radioactivity present, it may be expected that much development work will be required to demonstrate practical burner operations.

5.4.5.2 Solvent Extraction

There has been fairly extensive cold engineering-scale testing of the Thorex process. Additional work must be carried out to determine the influence of solvent and diluent degradation and of long-term radiolysis effects on the efficiency of the Thorex process.

5.4.5.3 Off-gas Treatment

Considerations for off-gas treatment have included processes for controlling the release of radon-220 (by holdup to permit decay), of carbon-14 (by fixation), of krypton-85 (by absorption in liquid carbon dioxide), and of iodine and tritium. Engineering-scale work on the krypton-absorption process has been completed. Removal systems for iodine and tritium would probably not be greatly different from those for other reactor fuels, particularly if fixation of the carbon dioxide were carried out first. All of these processes need to be demonstrated with remotely operated engineering-scale equipment.

5.4.5.4 Waste Treatment

With regard to waste treatment, the presence of fluoride in the wastes causes corrosion of equipment (possibly to the off-gas systems also) and may complicate high-level liquid-waste treatment and subsequent calcination and vitrification. The HTGR fuel wastes that have to be treated include graphite reflector blocks, silicon carbide hulls, and waste fissile particles clad with silicon carbide, as well as wastes from solvent-extraction and off-gas systems. There is limited experience in these areas.

5.4.5.5 Integrated Process Demonstration

Additional research and development work is required for the design, construction, and operation of a hot engineering test complex to test engineering-scale processes and equipment in the presence of uranium-233 (containing uranium-232) and fission products; and for the design, construction, and operation of selected cold prototype equipment for a reprocessing facility.

5.4.6 DECOMMISSIONING AND DECONTAMINATION

The discussion of decommissioning in Section 5.1.6 applies to a Thorex-reprocessing facility as well.

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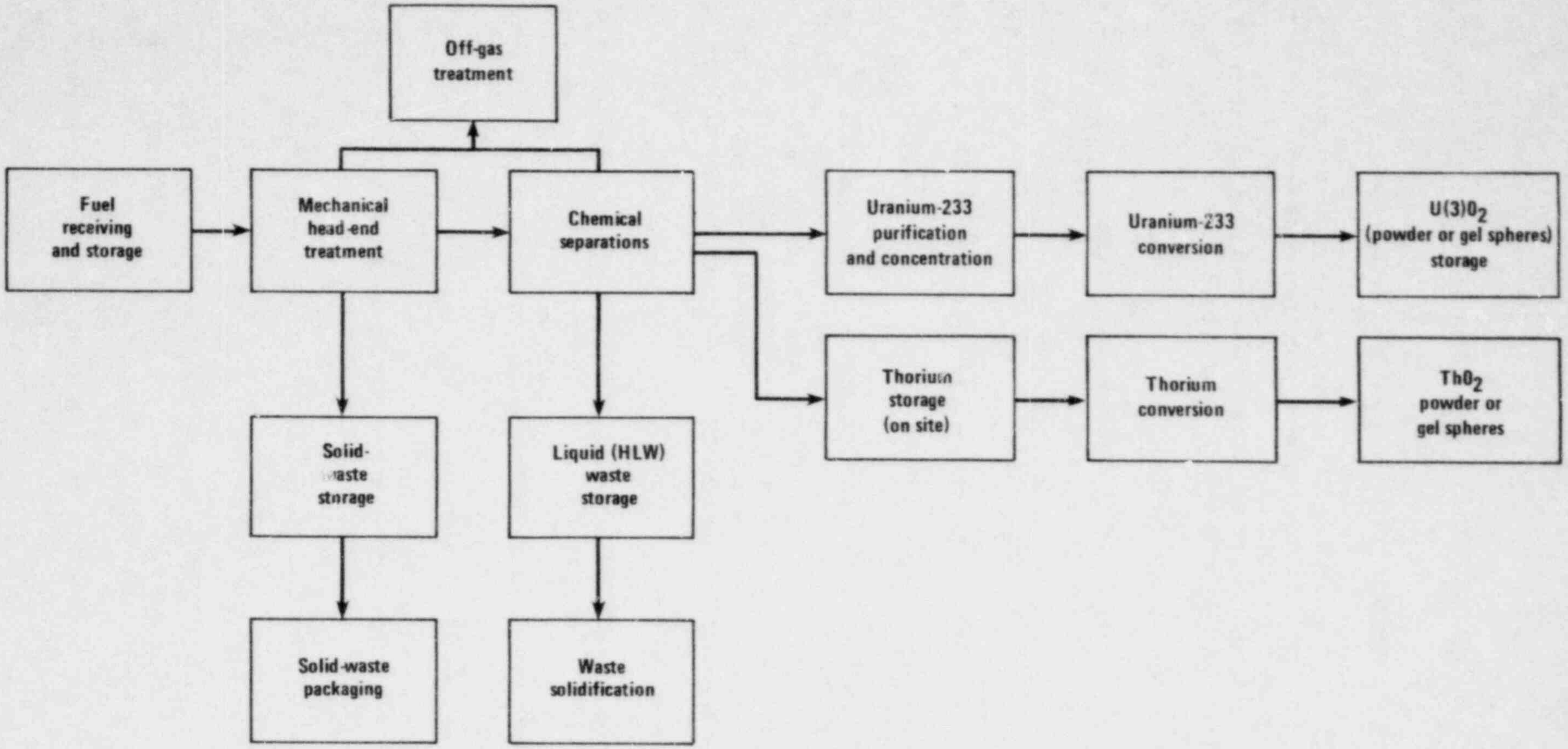

Figure 5-4. Thorex 1 flow diagram: partitioned uranium-233 and thorium.

5.5 THOREX 3 REPROCESSING: PARTITIONED URANIUM/PLUTONIUM/THORIUM FUEL

5.5.1 GENERAL DESCRIPTION

The Thorex 3 process (Ref. 6) is designed to treat the three-component system of uranium, plutonium, and thorium that occurs in denatured thorium cycles. Specifically, this system decontaminates and partitions the three heavy metals, giving separate product streams that are suitable feed materials for refabrication or storage. The flow diagram for this process is shown in Figure 5-5.

5.5.1.1 Fuel Receiving and Storage

The fuel receiving and storage described in Section 5.4.1.1 for a Thorex 1 facility is applicable to the Thorex 3 facility.

5.5.1.2 Head-End Treatment

The head-end treatment is as described in Section 5.4.1.2 for a Thorex 1 facility.

5.5.1.3 Product Separation and Purification

Solvent-extraction operations are similar to those described in Section 5.4.1.3; however, operating conditions for the initial cycle must be adjusted to coextract all three nuclides from the bulk of the fission products. Subsequent operations are regulated to first partition plutonium from thorium and uranium and then to partition thorium from uranium. When partitioning is completed, each of the heavy-metal streams is treated in additional cycles of solvent extraction for further purification.

5.5.1.4 Product Conversion

In addition to the product storage and conversion presented in Section 5.4.1.4, a Thorex 3 facility will also include the conversion of the purified plutonium nitrate product stream to the oxide with the addition of a cobalt-60 spikant. The spiked PuO_2 will be stored in a secure, shielded facility or shipped to a fabricator.

5.5.1.5 Waste Pretreatment

The waste pretreatment discussed in Section 5.4.1.5 applies to a Thorex 3 facility. In addition, the conversion of spiked plutonium to oxide will generate highly radioactive transuranic wastes (contaminated with cobalt-60) that must be immobilized and stored pending shipment to a geologic repository. The wastes will not be thermally hot and the bitumen process for immobilization may be usable.

5.5.2 SAFETY CONSIDERATIONS

This process will have essentially the same safety considerations as the Thorex 1 process described in Section 5.4.2. However, the plutonium product stream will require facilities for conversion to the oxide and storage. The added plutonium-processing facilities will require off-gas treatment and waste-treatment systems that may need to be separate from the uranium and thorium product processing to prevent cross contamination of the product streams.

5.5.3 ENVIRONMENTAL CONSIDERATIONS

The environmental considerations for a Thorex 3 reprocessing plant will be essentially the same as those described in Section 5.4.3.

5.5.4 LICENSING STATUS

The discussion in Section 5.4.4 applies to the Thorex 3 plant as well.

5.5.5 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

The research, development, and demonstration of this fuel cycle is a combination of that required for Purex 1 plutonium purification, the design and demonstration of a facility for the conversion of plutonium nitrate to oxide, and that required for a Thorex 1 facility (Section 5.4.5).

5.5.6 DECOMMISSIONING AND DECONTAMINATION

The discussion on decommissioning in Section 5.1.6 is applicable to a Thorex 3 plant.

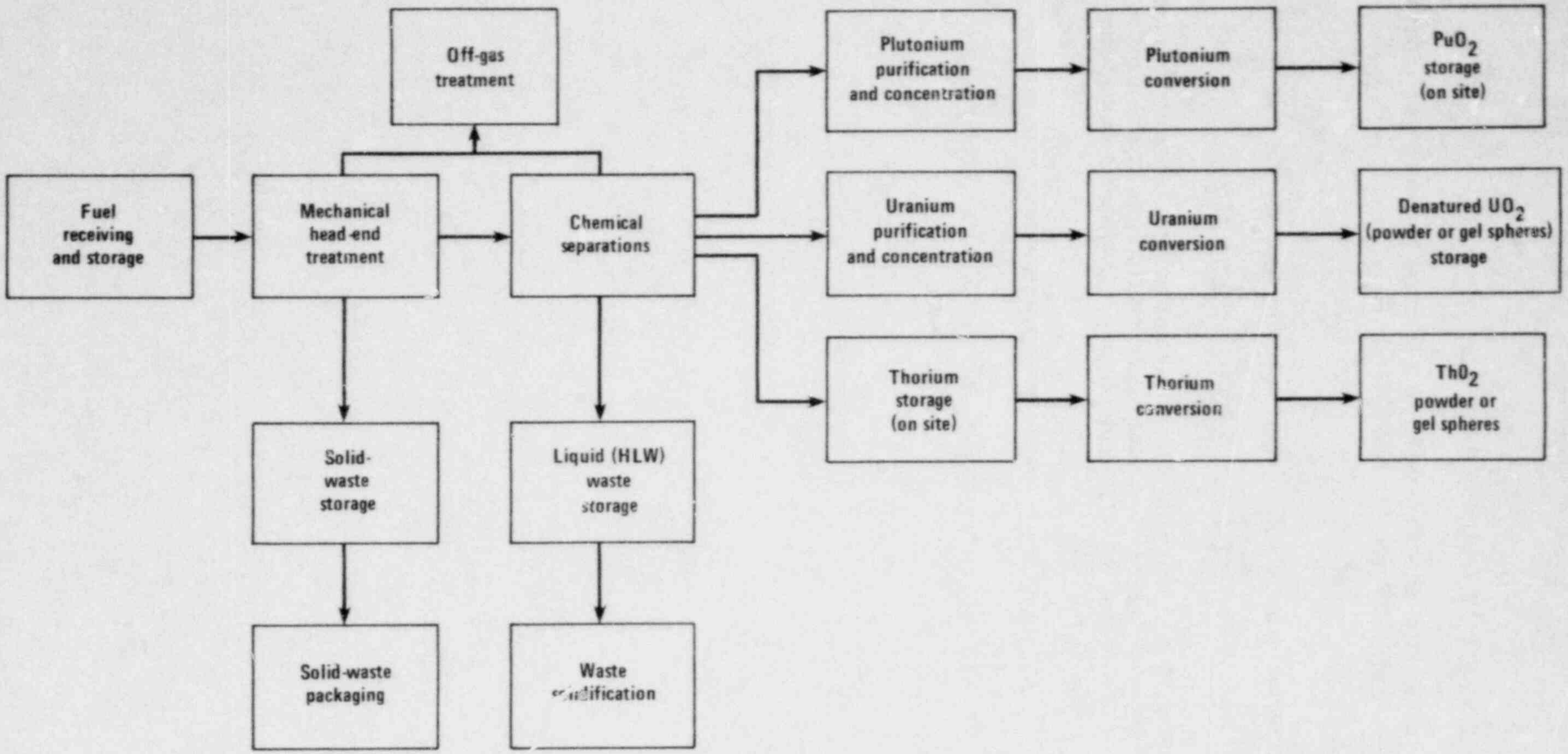

Figure 5-5. Thorex 3 flow diagram: partitioned uranium, plutonium, and thorium products.

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1. Oak Ridge National Laboratory, Correlation of Radioactive Waste Treatment Costs and the Environmental Impact of Waste Effluents in the Nuclear Fuel Cycle for Use in Establishing "as Low as Practicable" Guides--Fabrication of Light-Water Reactor Fuel from Enriched Uranium Dioxide, ORNL/TM-4902, May 1975.

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5. Fuel Fabrication and Recycle Evaluation for High Temperature Gas-Cooled Reactors, draft NASAP report (NUS Z-2).

6. U.S. Department of Energy, Preliminary Safety and Environmental Information Document, Nonproliferation Alternatives Systems Assessment Program, second preliminary draft report, July 1978.

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Chapter 6

STORAGE OPTIONS

6.1 STORAGE 1: THORIUM

6.1.1 GENERAL DESCRIPTION

Thorium recovered during the reprocessing of spent fuel operated on the thorium fuel cycle is in the form of thorium nitrate solution. It is radioactive due to the presence of uranium-232 in aged thorium and must be handled and stored in shielded conditions for 14 to 20 years before it can be fabricated in a contact operation, contact maintenance facility. The thorium nitrate may be evaporated to dryness or converted to ThO_2 before being sealed in storage canisters by the reprocessor. The loaded canister may be stored in a hot cell with convection or forced-air cooling. An alternative would be storage in a water-filled pool of sufficient depth to provide shielding.

In the past, thorium has been discarded as radioactive waste. Because thorium is not usable as weapons material, the extensive safeguards criteria for plutonium-handling and packaging are not applicable. Packaging criteria to prevent release would be similar to those used for plutonium oxide but the size of the containers would be limited by the thermal and handling considerations and is not mass or geometry limited.

It is assumed the thorium compound (oxide or nitrate) will be packaged in a gasketed steel container with provisions for pressure relief at intervals, if necessary. These containers may be overpacked in a gasketed steel canister for a second barrier if the material is stored in an air-cooled vault or in storage holes bored into the ground. The overpack possibly will also be provided with venting capability. Large tonnages of thorium may be involved; therefore, an economic impact analysis may show that a simple, minimal-cost storage facility may be required considering the relatively low cost of new thorium which will not require remote fabrication.

6.1.2 SAFETY CONSIDERATIONS

The thorium oxide or nitrate product canisters should be assayed, sealed, and decontaminated. An advanced identification and tamper-safe system may be provided to ensure continued validity of previously made measurements. The canisters, being highly radioactive and handled remotely, may not need the tamper-safe system. The product canisters will be transferred remotely to the storage vault or pool. Storage locations should be provided with features to expedite identification, inventory, and retrieval of the canisters. Provisions should also be included for recording ingress and egress of personnel and material to the operating areas of the storage facility. Continuous radioactive monitoring provisions should be included to establish the occupational exposure to operators and the radiation field level in the various operating areas and in exit ventilation air. Continuous thermal monitoring should be included in the hot cells, or of the pool water if underwater storage is used. Pool-water cleanup systems, similar to those used for spent-fuel pools, should be operated to protect the facility and personnel in the event of a container breach.

6.1.3 ENVIRONMENTAL CONSIDERATIONS

The land use associated with the storage of recovered thorium has not been developed. The facility should be expandable to store the cumulative quantities of thorium for periods up to 20 years to allow time for activation product decay to a level permitting contact fabrication. The facility size is a function of the reprocessing plant capacity and the amount of thorium processed annually. Waste heat generated in the thorium is expected to be minimal and fission product gases are not present. Water, fossil fuel, and electric power requirements are limited to what is required for the sanitary system, heating and ventilation, and monitoring instruments. The thorium will be sealed in containers that are decontaminated before being sent to the storage facility; therefore, unless a container develops a leak, the pool water (if pool storage is used) or air atmosphere of shielded vaults will not become contaminated. However, the pool water would require a purification system (just in case) and ventilation air will be processed through HEPA filters before release. There should be essentially no volatile radionuclides in the thorium. The principal health impact is associated with incremental occupational radiation doses, primarily during cask receiving and handling.

The movement of the shipping cask into a hot cell or pool pit for remote unloading may result in limited exposure. Neither the hot cell nor the pool, however, will become contaminated except under accident conditions such as the rupture of a canister during cask unloading. Such contamination would be removed before the empty cask is decontaminated and withdrawn from the hot cell or before the empty cask is decontaminated as it is removed from the pool. Occupational dose rates are expected to be negligible. Handling of pool-water-cleanup resins and changing HEPA filters are also expected to result in negligible exposure.

6.1.4 LICENSING STATUS

A thorium storage facility has not been licensed to date, nor is one under consideration. All thorium recovered from reprocessing has been stored as high-level liquid waste (HLLW).

The storage concepts appear to be straightforward and technically feasible. Pool storage may encounter fewer problems in licensing because of no criticality constraints; however, siting, type of construction, and required facilities would impact the licensing about the same amount as a spent-fuel storage facility or an interim waste storage facility.

6.1.5 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

Since there has been no experience in storage of recovered thorium other than as part of liquid high-level wastes, a technique must be conceived, researched, developed, and demonstrated. There should be few corrosion or degradation mechanisms that will affect the integrity of the stored canisters of the thorium compound. Long-term corrosion studies on potential canister materials are available in the literature to aid in design criteria development.

6.1.6 DECOMMISSIONING AND DECONTAMINATION

A thorium storage facility will have minimal, if any, contamination; therefore, decontamination activities will include checking and wash-down operations, if necessary, before the facility is dismantled.

In terms of resource commitment, other than land use, dismantlement will generate solid waste requiring disposal. The resources committed are expected to be about the same as for a spent-fuel storage pool and are as follows.

Land, hectares[a]	0
Water, m^3	1,000
Materials, metric tons	
Steel shipping containers	800
Paper, wood, plastic	30
Equipment (mostly steel)	80
Energy	
Diesel fuel, m^3	
Waste transportation	4,700
Electricity, kW/hr	5,000
Man-years	95

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[a] At site after decommissioning.

The steel containers and diesel oil requirements may be higher than necessary because most of the wastes can go to a landfill area instead of to a commercial burial ground.

6.2 STORAGE 2: PLUTONIUM

6.2.1 GENERAL DESCRIPTION

Most of the fuel cycles recovering plutonium or plutonium-uranium require interim storage of the plutonium. The plutonium or plutonium-uranium will be in the form of oxide powders, carbide powders, or unsintered spheres of either oxides or carbides. The product is packaged, in quantities of a few kilograms, in scaled metal cans of safe geometry within an inner gasketed steel container supported inside an outer steel drum of 100 to 110 gallons capacity. The gasketed steel cylinder is supported in the drum by thermal and shock-insulation material such as cane fiberboard, vermiculite, or foamed phenolic plastic. Mass limits of up to 4.5 kg of plutonium per package are defined in present package designs, primarily by heat-dissipation requirements. The U.S. Nuclear Regulatory Commission (NRC) regulations (10 CFR 71) require that plutonium in excess of 20 curies be shipped as a doubly contained solid.

A proposed scheme for packaging and storing plutonium oxide is described in detail in Reference 1. The plutonium oxide is unloaded into a storage-transfer-canister filling machine in a loading cell. The full canister is rotated from the filling position to the canister lidding machine. A lid is mechanically attached to the canister, which is then transferred with the in-cell crane to the decontamination chamber. The sealed canister is decontaminated in the decontamination chamber by a cleaning-fluid spray. After passing contamination inspection, the canisters are transferred to the loadout cell for weighing and insertion into pressure-vessel storage containers. Up to four canisters (32 kg of plutonium) are placed in each pressure vessel, which is then transferred in a shielded cask to the storage vault. The plutonium oxide pressure vessels are stored at the plutonium storage station (PSS). The floor of the PSS is a large, borated concrete slab containing an array of storage holes for pressure vessels containing plutonium dioxide.

The storage holes are on a 17.75-inch center-to-center spacing. They are neutronically isolated by the borated concrete. The PSS is divided into two rooms, each containing nominally 500 holes. A cover block over each hole, flush with the floor provides a level surface over which the electric forklift with the shielded pressure-vessel transport cask is driven. The slab contains conduit and instrumentation for measuring the pressure and temperature of the pressure vessel. The sleeve is stepped to accommodate the pressure vessel and the cover block. Air-flow passages are provided through the cover block, around the pressure vessel, and through the bottom of the slab into the storage exhaust plenum gallery to carry away radiolytic decay heat.

6.2.2 SAFETY CONSIDERATIONS

The plutonium dioxide product canisters should be assayed, sealed, and decontaminated (Ref. 1). An advanced identification and tamper-safe system should be provided to ensure the continued validity of previously made measurements. The product canisters should be transferred to the vault for storage. Vault storage locations should be provided with features to expedite identification, inventory, and retrieval of items containing plutonium. Possible features that could be evaluated include the following:

1. Grid layout and specific storage location for each canister
2. Inventory listing by unique identification and storage location
3. Separate plutonium oxide storage stations

4. Expansion of storage space so that individual storage containers can be retrieved
5. Provisions for recording the ingress and egress of personnel and material

6.2.3 ENVIRONMENTAL CONSIDERATIONS

According to information presented in Reference 2, approximately 6 acres of land would be required for a plutonium dioxide storage facility servicing a 1,000-MWe nuclear power plant (95 kg of plutonium per year), and an additional 9 acres would be used for roads and parking space. The storage facility would be contained within an exclusion area of approximately 250 acres. There would be negligible water use at the facility, and the use of electricity would be limited to lighting, safety interlocks, and cranes.

With the exception of sanitary wastes, no solid or liquid effluents should be generated by the storage facility. Approximately 46 x 10^9 Btu of thermal effluent would be released to the atmosphere annually.

6.2.4 LICENSING STATUS

No commercial facilities for the interim storage of plutonium have been licensed in the United States.

6.2.5 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

Interim storage of plutonium may require research, development, and demonstration in the following areas (Ref. 1):

1. Determination of the effect of powder characteristics on mechanical handling
2. Development of powder valves
3. Development of powder blenders
4. Determination of hydrogen-generation rates in powder in filled storage vessels

6.2.6 DECOMMISSIONING AND DECONTAMINATION

Decommissioning and decontamination has been recently recognized as an important aspect of every phase of the nuclear fuel cycle. To date, however, a sufficient quantity of data on the decontamination and decommissioning of a plutonium storage facility has not been identified.

6.3 STORAGE 3: SPENT FUEL

6.3.1 GENERAL DESCRIPTION

Two concepts for spent-fuel storage are currently being considered for use in the United States: (a) storage in water-cooled basins and (b) storage in air-cooled vaults. The general design characteristics of each type can be obtained from Reference 1. The technology of storing bare fuel in water basins is well developed and demonstrated by about 30 years of experience (Ref. 3). A facility for water-cooled storage of unpackaged (as-discharged) spent fuel would consist of near-surface, water-filled cells of heavily reinforced concrete lined with stainless steel, with auxiliary equipment for cooling and controlling the quality of the water. In addition to serving as a heat-transfer medium, water provides radiation shielding, partial containment for escaping radioactivity, and a heat sink under some accident conditions. In addition, water allows visual observation of the stored fuel. Spent fuel from LWRs is currently stored in water-cooled basins. Most of these basins are at reactor sites and are designed to hold from 50 to 150 metric tons (MT) of irradiated fuel (Ref. 3). The model away-from-reactor storage facility is assumed to be a water-basin facility with capacity of 5,000 MT of heavy metal in the form of fuel assemblies, expandable in increments of 1,000 to 15,000 MT, with a maximum receipt rate of 2,000 MT/yr. It is assumed for economy of scale and contingency coverage that the 5,000-MT away-from-reactor facility would be the minimum facility provided by the program (Ref. 4). The facility would accept 5-year-old fuel, which would be unloaded once a geologic facility became available. No fuel would be placed in it after the repository is available unless transfers from reactor basins exceeded the repository receipt rate. The away-from-reactor storage facility would include a main building, storage baskets, cooling towers, water treatment, receiving facilities for both truck and rail shipments, waste treatment, and service facilities. Standards of construction, including environmental standards, would be commensurate with commercially licensed nuclear storage facilities. It should be repeated that the spent fuel discharged from the reference reactor must be stored at the reactor for 5 years before it is shipped to the facility.

As pointed out in Reference 5, the quantity of fission products and actinides will be significantly increased for high burnup LWR fuel, thus potentially affecting the design of the spent-fuel pool and the design of systems in support of the spent-fuel pool. Further, with this higher burnup, an additional time lag in spent-fuel shipping (to either the away-from-plant storage facility, a repository, or a reprocessing facility) may be required.

The annual spent-fuel-storage requirements for the heavy-water-reactor (HWR) once-through fuel cycle will be greater than those for the reference light-water reactor (LWR) once-through fuel cycle. However, as shown in Reference 5, the quantity of fission products and actinides per metric ton of heavy metal (MTHM) will be less than that of the reference LWR once-through fuel cycle, since the anticipated burnup is 19.8 MWd/kg HM. Thus, as indicated in Reference 5, the decay heat of the HWR fuel 3.5 years after removal from the core is equivalent to that for the reference LWR once-through fuel cycle 5 years after removal from the core. Less lag time would therefore be required before spent fuel is shipped to the repository.

The annual spent-fuel-storage requirements for the high-temperature gas-cooled reactor (HTGR) once-through fuel cycle will be approximately 16% of those for the reference LWR once-through fuel cycle. However, as shown in Reference 6, the quantity of fission products and actinides per metric ton of heavy metal will be approximately four times greater than those for the reference LWR once-through fuel cycle.

6.3.2 SAFETY CONSIDERATIONS

Although the inventory of radioactive materials contained in aged spent fuels in a storage pool may be on the order of a billion curies or more, very little is in a dispersible form; there is no mechanism available for the release of radioactive materials in significant quantities from the facility; and the only mechanism for offsite transport is atmospheric dispersion. Increased spent-fuel storage with at-reactor or away-from-reactor storage normally involves only aged fuel. Underwater storage of aged spent fuels presents an extremely low risk of a catastrophic release of radioactivity (Ref. 7).

The spent fuel in storage is highly radioactive, with a total inventory of radionuclides on the order of 10^6 Ci/MTU. The gross radioactivity in curies per metric ton of uranium as a function of time since discharge from the reactor (decay time) is shown in Table 4.4 of Reference 7.

The fission-product radionuclides are beta and gamma emitters, and only those few that enter into biological processes are of major concern. For freshly discharged fuels at a reactor, a principal concern is the 8-day iodine-131 that is absorbed by plants, animals, and humans, particularly at inland locations that are naturally iodine deficient. However, since the quantity of iodine-131 present in discharged fuel is reduced by a factor of about 10^6 in the first 160 days of decay, it is not a major concern for the storage of spent fuels in an away-from-reactor storage facility (Ref. 7).

The fission-product nuclides of primary concern for long-term spent-fuel storage are krypton-85, cesium-134, cesium-137, and possibly iodine-129. These nuclides are present in significant quantities, are soluble in water, and are biologically mobile. Cesium enters the muscle tissue of animals and man. The isotope iodine-129 has a low specific activity (1.4 dpm/g) and hence minimal direct radiological effect; that is, it is physically impossible to absorb enough to give a dose that is more than a small fraction of that from natural radionuclides (e.g., potassium-40, 1,780 dpm/g) present in thyroid tissue. However, because of its half-life (1.7×10^7 yr), the release of iodine-129 to the environment should be minimized (Ref. 7).

The only way in which the radionuclides in spent fuel could be made dispersible is by physical rupturing of fuel pins. Since fuel assemblies must be handled under water to provide the necessary protective shielding, a rupture of fuel pins would allow the escape of free gases, primarily krypton-85, and contact between fuel material by the pool water. However, since the corrosion rates of ceramic fuel materials are low, the only observable effect might be a slight increase of cesium-137 in the pool (Ref. 7).

It is reported in Reference 7 that Nuclear Fuel Services, Inc., reported a pool-water impurities composition of 76% cesium-137, 6% cesium-134, 6% antimony-124, 6% cerium-144, and 1% strontium-90. At the General Electric plant in Morris, Illinois, cobalt-60 was also identified as a minor contaminant in pool water. Because of the direct relationship between levels of pool-water radioactivity and occupational exposures, there is an incentive to keep the levels of pool-water radioactivity under control at all times; values in the range of 10^{-4} to 10^{-3} Ci/ml are common.

Airborne radioactivity within a spent-fuel storage facility is a function of pool-water radioactivity, care in handling fuel, frequency of fuel-transfer operations, and good housekeeping practices (Ref. 7). Based on Nuclear Fuel Services, Inc., experience, the airborne activity is lower by a factor of 10^8 than the pool-water radioactivity and is routinely less than 5% of the occupational-exposure limits specified in 10 CFR 20, Appendix B, Table I.

For additional information related to accidents considered for the spent-fuel-storage facilities for the reference LWR once-through fuel cycle, refer to Section 4.2.3 of Reference 7. In summary, all anticipated credible accidents associated with fuel-pin rupture, missile penetration, fires and explosions, criticality, high pool-water radioactivity, waste-tank or piping ruptures, lowering of pool-water level, and loss of cooling would result in releases smaller than the allowed Federal limits.

A higher burnup can be achieved by increasing the feed uranium-235 enrichment to compensate for uranium-235 depletion and fission-product poisoning. However, the peak discharge burnup is limited by the original design for fuel performance. The fuel performance reliability is directly related to the peak discharge-burnup level because the cladding integrity for a safe and reliable operation is a function of irradiation time (Ref. 7).

The utilization of fuel at a significantly higher burnup level would require a stronger cladding (either a high-strength material or an increased cladding thickness) to maintain fuel-rod integrity during the longer fuel life. More generally, safety analysis, licensing procedure, and the economics of design and manufacturing standardization favor the continuation of proved fuel designs and burnup levels. Hence, changes in the fuel design to accommodate a higher burnup and subsequent modification of the fuel-management strategies will not be realized quickly. Furthermore, an increased burnup requires (1) increased uranium-235 enrichment to provide additional available reactivity for a longer fuel life and (2) increased reactivity-control margins. The increased enrichment of the fuel would require a reevaluation of the safety analysis (Ref. 7).

The use of slightly enriched uranium fuel (approximately 1.2 wt%) will increase discharge burnup to approximately 20,000 MWd/MT, as compared to approximately 7,500 MWd/MT for the current Canadian designs. Although major improvements in current fuel technology are not considered necessary in order to achieve these higher burnups, development testing for mechanical performance acceptability will be required (Ref. 8). Assuming that these tests and irradiation experiments show an acceptable fuel design, the safety considerations associated with this fuel cycle will be similar to those defined above.

Since HTGR spent fuel, assumed to be stored in water pools, is composed of graphite, the spent fuel will require double containment (Ref. 6). Furthermore, on a metric ton of heavy metal basis, there will be additional fission products and actinides to consider in an overall safety analysis report (SAR). All support systems for the away-from-reactor storage facility will require design and submittal of an SAR before actual construction. Studies to date indicate that structural considerations for spent-fuel storage will override either decay-heat removal or criticality (Ref. 6).

6.3.3 ENVIRONMENTAL CONSIDERATIONS

Volatile and nonvolatile radionuclides with short half-lives will have decayed to negligible levels. Consequently, the radiological and heat-load impacts of this older fuel are lower by factors of 10 than those of the newer fuel stored at a nuclear power plant. The principal health impact is associated with incremental radiation doses (Ref. 7).

The radiological impact from spent-fuel storage is as follows:

1. Population dose due to the release of krypton-85 from leaking fuel elements
2. Occupational exposure of plant personnel incurred while working in the vicinity of the spent-fuel-storage pool, for example, changing water-purification filters and ion-exchange resins

These types of impact are generic to spent-fuel-storage operations regardless of whether the fuel is stored at a nuclear power plant or at an away-from-reactor storage facility.

For the "aged" fuel involved in relatively long time storage, krypton-85 leakage rates are too low to be detected. For the final GESMO, Chapter IV-K, Extended Spent Fuel Storage, however, a conservative release rate of 1 Ci/MT-yr was used. (Based on experience at the General Electric plant at Morris, Illinois, this figure could be high by a factor of 10^6.) The resultant population dose factors were: United States, 0.004 man-rem/MT-yr; foreign, 0.02 man-rem/MT-yr.

Occupational dose rates, based primarily on at-reactor experience, used in the final GESMO were 20 man-rem per 1,000 MT-yr.

The above doses are applicable to conventional water-basin storage pools. The doses for the various types of passive dry-storage systems under development are expected to be comparable or less. Based on these figures, the calculated doses from all spent fuel in storage are shown in Table 6-1 as obtained from Reference 7. Note that the population doses are not corrected for krypton-85 decay.

The water serves as a coolant to remove decay heat of the spent fuel, and as a radiation shield. The occupational radiation exposure results from the radioactivity in the water and the required operational activities. The spent fuel contributes a negligible amount to dose rates in the pool area because of the depth of water shielding the fuel.

Radioactivity in the pool water comes from the dislodging of crud from the surface of the spent-fuel assemblies during handling of the assemblies, and the leakage of fission products from defective spent-fuel elements. It is not expected that there would be a significant increase in the number of times the assemblies are handled before shipment offsite. Any significant removal of crud from the surface of an assembly would occur during the initial fuel handling when the assembly is transferred from the core to the storage pool. Experience with spent fuel stored at the General Electric plant at Morris and at the New York plant of Nuclear Fuel Services, Inc., has indicated that there is little or no leakage of radioactivity from spent fuel that has cooled several months.

The pool-cleanup system serves to clarify the pool water and to remove the radioactive materials. The technology is well developed, and it is not uncommon to find fuel-pool water with radioactivity content comparable to the 10 CFR Part 20 limits for occupational use. Water carried out of the fuel pool by mechanical means or seepage is collected in sumps and recycled through a radwaste cleanup system. Small amounts of pool water eventually reach the environment but only after several levels of radwaste treatment, so that the quantities of radioactivity released are insignificant.

The only gaseous radionuclides released to the atmosphere in significant quantities are the noble gases, principally krypton-85. Some radiation reaches the environment in the form of direct radiation from the fuel within the pool and some from the transportation of intermediate-level wastes to the final disposal site. Direct radiation in the vicinity of the spent-fuel storage pool is extremely low, on the order of 1 to 2 mrem/hr. If this were the only contribution to the occupational dose, that dose would be quite small. However, the occupational dose is dominated by the exposures involved in handling and moving the fuel, in handling radwaste, and in decontaminating tools, operations during which the dose rates are higher. For additional information, refer to Section 4 of Reference 7.

The increased burnup of the fuel will produce additional quantities of fission products and actinides. As noted in Section 6.3.2, the specification of a burnup of 50 MWd/kg HM will require a redesign of the fuel and cladding, thus implying that the final product would be as safe as the reference fuel. To date, the environmental impact of this type of fuel has not been defined.

Facility designs for HTGR spent-fuel storage have typically followed the concept of total containment and isolation from groundwater by (1) appropriate seismic design criteria and (2) treatment of all potentially contaminated liquid waste and conversion to a solid form for long-term storage. The medium-enrichment fuel cycle presents no basic differences from the previous reference high-enrichment cycle. In facility designs for HTGR spent-fuel storage, the probable future requirement for the control of krypton, tritium, and carbon-14 has been recognized and suitable retention technology is being developed (Ref. 8).

6.3.4 LICENSING STATUS

The spent-fuel storage facility at Morris, Illinois, operated by General Electric, has a current operating license and is storing spent LWR fuel. An away-from-reactor interim spent-fuel storage facility should be licensable within current regulatory limitations. No modification of 10 CFR 51.20(e), the summary of environmental considerations for the uranium fuel cycle, appears necessary for spent-fuel storage considerations for reference LWR fuel.

To date, no programs have been initiated to license storage facilities for 50-MWd/kgHM burnup spent LWR fuel, for 20 MWd/kgHM burnup spent fuel from the HWR, or 133 MWd/kgHM burnup spent fuel from the HTGR.

6.3.5 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

Several years' experience indicates that the storage of spent fuel in reference water-cooled pools is a practical method for storing fuel for 15 to 20 years and possibly longer (Ref. 9). This conclusion is supported by a recent assessment of spent-fuel storage experience (Ref. 10).

Compared to the high temperature and harsh environment of the reactor core, there seem to be few corrosion or degradation mechanisms that will affect the integrity of the spent fuel in the pool. This should be confirmed with exploratory sampling and examination of pool-stored fuel (Ref. 10). To date, there has been little attempt to examine the condition of the fuel after extended storage in a pool.

For longer periods--from 20 to 50 years--there is no experience with the stability of spent fuel (Ref. 9). The longest storage time reported for Zircaloy-clad fuel is 18 years (for one fuel element); for stainless-steel-clad fuel the longest time is 12 years (Ref. 10). Because of the lack of detailed information on the status of fuel stored underwater for long times, corrosion mechanisms that may affect the long-term integrity of spent fuel are not fully understood.

Research, development, and demonstration for HWR fuel will be similar to that required for the improved LWR once-through fuel cycle.

It is anticipated that research, development, and demonstration efforts, especially in the design of fuel canisters, will be required for the HTGR once-through fuel cycle.

6.3.6 DECOMMISSIONING AND DECONTAMINATION

Two modes of decommissioning the interim spent-fuel-storage facility (ISFSF) have been considered: immediate dismantlement and hardened safe storage (Ref. 11). Dismantlement results in the removal of contaminated equipment and structures from the site 2 to 3 years after shutdown. Hardened storage involves isolation of contaminated equipment and structures within existing facilities so as to preclude the release of any radioactive material. The 10-hectare exclusion area will be maintained under continuing care for 100 years.

In terms of resource commitment, other than land use, dismantlement requires 10 times more resource material than does hardened storage. Dismantlement will also generate 10 times more nontransuranic waste requiring disposal elsewhere. Manhours required for decommissioning are similar when 100 years of surveillance after hardened storage are included. Physical, chemical, and thermal effects will be greater when dismantlement is used. In either case, these effects are easily mitigated or insignificant.

Comparison of the radiological aspects during decommissioning indicates that "maximum individual" and population doses for facility decommissioning are similar for both modes. Accidents postulated for decommissioning an ISFSF would not result in any radioactive releases to the environment.

In summary, differences between the two decommissioning modes are due mostly to the quantities of waste shipped to distant repositories. Dismantlement requires the commitment of 10 times more resources and thus a 10 times greater population dose (from the transportation of nontransuranic wastes generated) than does hardened storage. No significant differences in terms of manpower, ecological effects, or radiation doses to the population within an 80-km radius of the facility exist between the two modes of decommissioning discussed.

Tables 6-2, 6-3, and 6-4 list the environmental effects of these modes for purposes of comparison.

Table 6-1. Radiation doses from spent-fuel storage

Year	Fuel in storage (MT)	Occupational dose, whole body (man-rem)	Population dose, skin (man-rem)	
			U.S.	Foreign
1980	7,200	150	31	140
1985	18,000	360	77	350
1990	33,000	660	140	650
1995	59,000	1,200	250	1,200
2000	95,000	1,900	410	1,900

Table 6-2. Resources committed to facility decommissioning

Resource	Dismantlement	Hardened storage[a]
Land, hectares[b]	0	10
Water, cubic meters	1,000	1,000
Materials, metric tons		
Steel shipping containers	800	50
Paper, wood, plastic	30	20
Equipment (mostly steel)	80	5
Energy		
Diesel fuel, cubic meters		
Waste transportation	4,700	180
Gasoline, cubic meters		
Electricity, kW-hr	5,000	3,000
Man-years	95	90

[a] Includes resource commitments during 100 years of continuing care.

[b] Land commitment at facility site after completion of decommissioning.

Table 6-3. Radionuclides released to biosphere during the decommissioning of the reference intermediate spent-fuel storage facility

	Radioactivity released (Ci/yr)			
	Air		Water	
Nuclide	Dismantlement	Hardened storage	Dismantlement	Hardened storage
Iron-55			6.5×10^{-3}	6.5×10^{-3}
Cobalt-60	1.9×10^{-8}	1.9×10^{-10}	9.5×10^{-3}	9.5×10^{-3}
Strontium-89			4.7×10^{-5}	4.7×10^{-5}
Strontium-90	7.2×10^{-9}	7.2×10^{-11}	3.6×10^{-3}	3.6×10^{-3}
Tellurium-127m	4.8×10^{-11}	4.8×10^{-13}	2.4×10^{-5}	2.4×10^{-5}
Tellurium-129m			2.3×10^{-6}	2.3×10^{-6}
Cesium-134	4.1×10^{-8}	4.1×10^{-10}	2.0×10^{-2}	2.0×10^{-2}
Cesium-137	4.3×10^{-7}	4.3×10^{-9}	2.2×10^{-1}	2.2×10^{-1}

Table 6-4. Comparison of the radiological aspects of decommissioning

Organ	Dismantling	Hardened storage
Doses (rem) to maximum individual, 70-year residency, from decommissioning operations		
Whole body	1.7×10^{-4}	1.7×10^{-4}
Thyroid	8.3×10^{-7}	8.3×10^{-7}
Lung	2.8×10^{-5}	2.8×10^{-5}
Bone	2.2×10^{-4}	2.2×10^{-4}
Doses (rem) from direct radiation from shipment of non-high-level nontransuranic waste		
Whole body	1.5×10^{-4}	5.8×10^{-5}
(Dose from natural background radiation 7.0 rem)		
Doses (man-rem) to regional population--2×10^6 persons within 80-km radius, 70-year residency, from decommissioning operations		
Whole body	6.2	6.2
Thyroid	3.4×10^{-2}	3.4×10^{-2}
Lung	1.5×10^{1}	1.5×10^{1}
Bone	1.4×10^{2}	1.4×10^{2}
Population dose (man-rem) from direct radiation from shipment of non-high-level nontransuranic waste		
Whole body	2.5	1×10^{-1}
(Dose from natural background radiation 1.4×10^7 man-rem)		
Doses to work force (man-rem)		
Decommissioning	6.9×10^{1}	2.4×10^{1}
Waste transportation	1.1×10^{2}	4

Note: There is no release of tritium, carbon-14, or krypton-85; therefore, there is no worldwide dose.

6.4 STORAGE 4: DEPLETED URANIUM

6.4.1 GENERAL DESCRIPTION

Depleted uranium is stored by the DOE at the gaseous diffusion plant storage areas, primarily in the form of uranium hexafluoride (UF_6). The UF_6 is packaged in 10-ton-capacity steel cylinders which are stored out of doors on wood chocks located on a level, graveled area. Periodically, the DOE converts a quantity of UF_6 to uranium tetrafluoride (UF_4) to recover the fluorine and release the steel cylinders for reuse. The UF_4 is packaged in plastic lined, 55-gallon steel drums which are stored on a concrete pad either in the open or under cover.

6.4.2 SAFETY CONSIDERATIONS

The cylinders of UF_6 must be routinely checked to ensure that there is no valve leakage. Leaking valves are replaced. While the cylinders are slightly radioactive, no special precautions are required for occupational exposure. Care must be taken in moving cylinders to prevent accidental breakage of a valve. Even with a valve leaking or broken, the solid UF_6 reacts slowly with airborne moisture to form crystals of UO_2F_2 which are readily visible and are not dispersable in the atmosphere. Leaks and breaks tend to be self-sealing.

Drums of UF_4 must be checked regularly for trapped water on the lids that could cause rust. Moisture reaching the UF_4 will react with free fluorine to form HF which is corrosive to the drums.

UF_6 and UF_4 have been stored safely in large quantities for many years.

6.4.3 ENVIRONMENTAL CONSIDERATIONS

The radiological impact from depleted uranium storage is negligible to both the public and to the operators of the storage facility. There are no resources required other than land area which is temporarily committed to storage yards or warehouses. There are no effluents under normal operating conditions. Valve leaks or breaks may cause very local contamination which is easily controlled. Corrosion of storage drums may also cause local spot contamination and require re-drumming of the UF_4. This operation would be conducted in the 6/4 conversion plant, separate from the storage facility.

6.4.4 LICENSING STATUS

The depleted uranium storage areas are a part of the DOE gaseous diffusion plant complex and would not be licensed under present criteria.

6.4.5 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

Depleted-uranium storage has been conducted successfully for many years. Research, development, and demonstration work is not considered necessary.

6.4.6 DECOMMISSIONING AND DECONTAMINATION

The storage yards would need to be monitored for radioactivity and contaminated spots would have to be removed by digging up the gravel/soil to a nonactive level. The soil would be drummed and disposed of in a low-activity waste-disposal shallow land burial facility.

6.5 STORAGE 5: URANIUM-233

6.5.1 GENERAL DESCRIPTION

The uranium-233 produced in reactors would be recovered in the reprocessing step and stored for eventual use (Ref. 2). At present, there are no large-scale facilities for storing these materials.

A model facility capable of storing approximately 20 metric tons of uranium-233 in the form of an aqueous nitrate solution has been conceived. The present concept is to have a uranium storage facility located at one site, possibly an integrated fuel-cycle facility.

For the storage of uranium-233, which is a fissile material, the controlling factor would be criticality. The solution storage containers would have to be geometrically safe--either thin slabs, annular vessels, or cylinders (Ref. 2). Tanks containing fixed neutron absorbers (poisons) might also be used. It is assumed that the uranium-233 could be stored in 10-liter bottles designed for the safe handling of plutonium nitrate solution.

To provide the necessary spacing, it is envisioned that two 55-gallon drums would be welded together, and the 10-liter bottle would be placed in the center of the drums inside a stainless steel pressure vessel. The uranium concentration in the nitrate solution would be approximately 375 g/liter (Ref. 2).

Assuming that drums are not stacked, the required floor space, allowing as much space for aisles and shielding as for drum storage, would be approximately 2 acres. The building area used for uranium-233 storage would include design features to prevent spilled liquid from leaving the building (Ref. 2).

Freshly recovered uranium-233 can be handled in a contact facility. After about 2 weeks, however, the decay product will increase the activity level to a point requiring 1 to 2 feet of concrete shielding and remote operations. Therefore, the storage building would need to provide shielding and remote handling in the storage area and provisions for shielded cask handling for removal of stored uranium-233 for reuse.

6.5.2 SAFETY CONSIDERATIONS

Because of the passive nature of storage, the likelihood of accidents would be remote. The uranium nitrate solution is not flammable. The building structures would be designed to resist earthquakes and tornadoes. Two accidents, both highly unlikely, are nevertheless postulated. The first is an explosion resulting from a lack of ventilation of evolved hydrogen from radiolysis of the stored uranium solution. The second is criticality of the stored uranium.

In the unlikely event of an explosion, the amount of radioactive material that could be released to the environment is estimated to be less than 0.4 microgram of uranium. In the event of a criticality involving 10^{18} fissions, the fission products that could be released to the environment have been estimated and are discussed in Reference 2.

6.5.3 ENVIRONMENTAL CONSIDERATIONS

The storage facilities themselves would cover approximately 9 acres, but an additional 13 acres would be needed for parking and roads. No permanent commitment of land is envisioned. If the storage facility were isolated, approximately 228 additional acres would be required to provide an exclusion area with a distance to the site boundary of roughly 500 meters.

With the exception of sanitary requirements, no significant use of water is anticipated for the model storage facility. Air cooling would be sufficient to handle the decay heat generated by the stored material (Ref. 2).

The model storage facility would not consume significant amounts of energy or materials, with the exception of the construction materials. Electricity would be used only for lighting and powering the safety interlocks--a negligible use when compared with the other components of the fuel cycle (Ref. 2).

Gases, including radiological species, would be produced in the storage facility on a continuous basis. Hydrogen, approximately 10,000 ft^3 per year, and oxygen would be evolved by the dissociation of water in the aqueous solution. It is estimated that approximately 0.14 MT of nitrogen oxides would be evolved annually from the nitrate solution. However, even if the storage facility were isolated on a 250-acre site, the resulting nitrogen oxide concentration at the site boundary would be less than 1% of ambient air quality standards (Ref. 2).

Radon-220 gas would be formed, and approximately 700 to 1,800 curies per year would be released after treatment. The half-life of radon-220 is about 56 seconds, and hence the gas would undergo considerable decay before reaching the site boundary (Ref. 2).

No radiological effluents would be released to the watershed or ground, and no nonradiological effluents would be released to the ground.

The total thermal effluents from the cooling of the stored material are estimated to be (3.1 to 15) x 10^9 Btu per year for a combined uranium-thorium storage facility.

Cleaning and sanitary water used at the storage facility would likely use the water-treatment system of the integrated fuel-cycle facility and add a negligible increment to the effluents from the treatment system. In the event of a spill, the liquids would be contained within the storage area and the grounds would be decontaminated.

6.5.4 LICENSING STATUS

Large-scale storage facilities for the storage of uranium-233 do not exist. To date, no large-scale licensed facility has been identified.

6.5.5 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

To date, research, development, and demonstration requirements for the safe storage of uranium-233 have not been identified. It is anticipated, however, that minimal to moderate effort will be required, depending on the applicability of other licensed facilities for storing fissile materials.

6.5.6 DECOMMISSIONING AND DECONTAMINATION

Decommissioning and decontamination has been recently recognized as an important aspect of every phase of the nuclear fuel cycle. However, to date, a sufficient quantity of data on the decontamination and decommissioning of a uranium-233 storage facility has not been identified. It should be noted, however, that overall decommissioning and decontamination studies for a fuel-reprocessing facility and mixed-oxide fuel-fabrication facility have been defined and are presented in Reference 11.

6.6 STORAGE 6: URANIUM-235

6.6.1 GENERAL DESCRIPTION

Low-to-medium (<20%) assay uranium-235 recovered from one LWR and one HTGR fuel cycle requires storage. The recovered uranium will probably be in the form of uranyl nitrates stored in 10-liter bottles designed for safe handling of plutonium nitrate solution. At present, there are no large-scale facilities for storing enriched uranium except at fuel fabrication facilities. The present concept is to have a uranium storage facility located at the reprocessing plant. An alternative to storage in the 10-liter bottles would be storage in safe-geometry tanks or conversion of the uranium to oxide, with storage in 5-gallon pails. Because of criticality considerations, the storage containers would need to be on a safe spacing. The uranium-235 is only slightly radioactive and would not require a shielded facility. A vault may be required for physical protection of the materials.

6.6.2 SAFETY CONSIDERATIONS

The same safety considerations discussed in Section 6.5.2 for uranium-233 storage facilities will apply to a uranium-235 storage facility. Because of the lower radiation from uranium-235, the amount of evolved hydrogen will be much less and may preclude the possibility of an explosion postulated for the uranium-233 facility.

6.6.3 ENVIRONMENTAL CONSIDERATIONS

The environmental considerations for a uranium-235 storage facility would be similar to those discussed in Section 6.5.3 except that the amount of hydrogen would be expected to be much less. The thermal effluents would be essentially zero.

6.6.4 LICENSING STATUS

Many storage facilities have been licensed as part of other facilities and licensing should be relatively easy even though a uranium-235 storage facility as a separate entity has not been identified.

6.6.5 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

Research, development, and demonstration effort is not expected to be required for a uranium-235 storage facility since the parameters have been established for years.

6.6.6 DECOMMISSIONING AND DECONTAMINATION

See discussion in Section 6.5.6 as applied to a uranium-235 storage facility.

REFERENCES FOR CHAPTER 6

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5. W. L. Carter, An Evaluation of Fuel Fabrication and the Back End of the Fuel Cycle for Water Reactors (draft), July 10, 1978.

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10. A. B. Johnson, Jr., Behavior of Spent Nuclear Fuel in Water Pool Storage, BNWL-2256, Battelle Pacific Northwest Laboratories, September 1977.

11. U.S. Department of Energy, Environmental Impact Statement, Management of Commercially Generated Radioactive Waste (draft), DOE-1559, Vol. II, September 15, 1978.

Chapter 7

WASTE DISPOSAL OPTIONS

7.1 WASTE DISPOSAL 1: GEOLOGIC DISPOSAL OF SPENT FUEL

7.1.1 GENERAL DESCRIPTION

Although several alternatives (Ref. 1) have been investigated for the geologic disposal of high-level waste (HLW) and spent fuel (Ref. 2) no actual demonstration facilities have been constructed to verify the applicability of the various recommended models (Ref. 3). To satisfy the spent-fuel-disposition model for the reference light-water reactor (LWR) once-through fuel cycle, the bedded-rock-salt geologic formation was chosen as described in Reference 2 and the repository was sized according to the expected spent-fuel generation scenario described in Reference 4.

The spent fuel received at the repository is assumed to be generated in various LWR fuel cycles. All spent-fuel assemblies are assumed to arrive at the repository in suitable stainless-steel canisters. If a damaged canister were received, it would be overpacked at the repository. If a canister were excessively contaminated with surface radioactivity on receipt, it would be decontaminated before handling. No other packaging operations would be required (Ref. 2). Spent fuel for the reference LWR fuel cycle is assumed to be 10 years old on receipt at the repository.

The underground and surface areas were chosen to be 2,000 and 200 acres, respectively. These specific acreages were assumed because they represent reasonable design values. The underground area is large enough to accept spent fuel through the year 2000 or later, depending on the fuel cycle and rock type. The surface area is sufficiently large to contain all of the surface buildings, the railroad and road systems, and the rock excavated from the rooms that would be filled during the retrieval period, yet it is conveniently sized for security and surveillance.

The assumption of retrievability probably has the greatest impact on repository design and operation. The concept of retrievability assumed in these studies requires that the fuel assemblies be retrievable for a defined period of time with approximately the same effort and at approximately the same rate at which it was emplaced. During the period of time that the spent fuel is retrievable, the repository rooms must remain open to waste placement and removal equipment, and the waste containers must remain structurally intact. It is not obvious that spent-fuel retrievability will be required for a repository in the National Waste Terminal Storage program, but such a concept has been assumed in the repository designs discussed here. The assumption of a retrievability requirement complicates repository design and operation and increases costs.

Two reasons for requiring spent-fuel retrievability can be postulated. First, retrievability of all wastes for a period of time might be required for further verification of deep geologic repository concepts, reactions of the host rock, and facility operational procedures. Engineering considerations dictate that the period of retrievability be fixed before detailed engineering design. A period of 5 years has been selected to define this operational mode in these designs.

A second reason for retrievability stems from the concept of interim storage for spent unreprocessed fuel (SURF). It may be desirable to recover the spent-fuel assemblies for reprocessing at a later date. If this option is to be maintained for a specified time, engineering design must ensure that each spent-fuel assembly placed in the repository is retrievable for that entire period. A retrieval period of 25 to 100 years might be considered appropriate for the SURF cycle. At least 25 years would appear to be required under this concept because of the lead time necessary to establish a reprocessing capability after a decision to reprocess fuel has been made. In this second concept, the waste from reprocessing could possibly be returned to the same repository from which the spent fuel was retrieved. In these studies, 25-year retrievability is analyzed as an alternative design for the SURF cycle. Additional information on repository design can be obtained from Reference 2.

The baseline salt repository for the reference LWR once-through fuel cycle will have a capacity of 62,000 and 36,600 MTU for pressurized-water reactor (PWR) and boiling-water reactor (BWR) fuel assemblies, respectively (Ref. 2), yielding a total thermal loading of 166 kW/acre. An underground storage area of approximately 1,700 acres is required. The local thermal loading limits for 5- and 25-year retrievability of spent fuel are 150 and 36 kW/acre, respectively, in a salt formation.

The implementation of the improved LWR once-through cycle will ensure that the yearly offload of spent fuel from the reactor will be approximately 40% less than that from the reference LWR once-through fuel cycle. However, as shown above, a critical design parameter for the repository is the thermal loading generated by the spent fuel. The scenario for the reference once-through fuel cycle requires that spent fuel be cooled for 10 years before disposal. Thus, it can be concluded that, during the same time frame for interim storage, fewer fuel assemblies could be stored in the repository because of the increased heat load (see Ref. 5) or, alternatively, the improved spent fuel would require several years of additional interim storage before disposal.

The implementation of the heavy-water reactor (HWR) once-through fuel cycle will ensure that the yearly offload of spent fuel from the reactor will be approximately 60% greater than that associated with the reference LWR once-through fuel cycle. However, a critical design parameter associated with the repository is the thermal loading generated by the spent fuel. The reference LWR once-through fuel scenario requires that spent fuel be cooled 10 years before disposal. Thus, during the same time frame for interim storage, more HWR fuel assemblies could be stored in the repository because of the decreased heat load (see Ref. 6) or, alternatively, the HWR fuel would require approximately 8.5 years of storage before disposal.

The high-temperature gas-cooled reactor (HTGR) once-through fuel cycle will require approximately 16% of the storage space needed for the reference LWR once-through fuel cycle. However, the quantity of fission products and actinides per metric ton of heavy metal will be approximately four times greater than that generated in the reference LWR once-through fuel cycle. Thus, during the same time frame for interim storage, fewer fuel assemblies could be stored in the repository because of the increased heat load or alternatively, the HTGR fuel would require additional interim storage time before disposal. It is not clear at present whether terminal repositories can, or will, accept HTGR fuel (Ref. 7). It may be necessary to burn off the graphite and recan the fuel particles in an inert matrix because of combustion or canister configuration requirements at repositories.

7.1.2 SAFETY CONSIDERATIONS

A number of accident scenarios have been considered, and the amounts of radioactive material released during the postulated accidents have been conservatively estimated. Estimated maximum releases in each category of interest are presented in Table 7-1. These maximum releases encompass all the accident scenarios. The magnitude of even this release is such that notification of the director of the appropriate NRC regional office is required only within 24 hours of the accident; immediate notification is not required. An accident involving spent fuel was predicted to release only about 25 times the limit specified in 10 CFR 20 for the continuous release of beta- and gamma-emitting particulates to the unrestricted area. The estimated maximum 24-hour average concentrations in an unrestricted area would occur at the repository 200-acre perimeter and assume worst-case atmospheric conditions (fumigation). Site-specific calculations will have to be performed when detailed designs are available.

A waste repository will contain appreciable amounts of fissionable isotopes, but, to form a critical mass, one or more of these isotopes would have to follow a complex and highly unlikely process of selective leaching, selective transport, and/or selective deposition. Despite the low probability of such a process, preliminary calculations of plutonium dioxide critical mass requirements under various conditions have been performed. They are described in more detail in Reference 2, which also discusses criticality considerations during the handling and transport of spent fuel. No circumstances are envisioned in which criticality can occur during the operation of the repository.

Accidental release rates from the improved spent fuel may be equivalent to those of the reference spent fuel (allowing for several years of additional interim storage before disposal) or greater than those presented in Table 7-1. Detailed information on the quantification of the accidental release for fuel with a discharge burnup of 50 MWd/kg HM has not been defined to date.

Accidental release rates from the HWR once-through fuel cycle may be equivalent to or less than that associated with the reference LWR once-through fuel cycle (see Table 7-1). Detailed information on the quantification of the accidental release rates for fuel with a discharge burnup of approximately 20 MWd/kg HM has not been defined to date for the HWR fuel design.

Accidental release rates from the HTGR once-through fuel cycle may be greater than those listed in Table 7-1. Detailed information relevant to the quantification of the accident release rates for fuel with a discharge burnup of 133 MWd/kg HM has not been identified to date. Furthermore, if the spent-fuel particles are stored in canisters after the graphite is burned off, it is anticipated that accident analysis incorporating this model will be required to update the information presented in Table 7-1 for this fuel cycle.

7.1.3 ENVIRONMENTAL CONSIDERATIONS

The resource requirements and environmental impacts associated with the construction and operation of a deep geologic waste repository are discussed in Reference 2, which analyzes both normal operations and releases from accidents. Specific consideration is given to the repository design details for each host rock and the parameters associated with each waste form from each fuel cycle.

The major resource requirements for the construction and operation of a deep geologic waste repository are listed in Table 7-2. The entire surface area is committed only until the respository is decommissioned, after which the Federal Government will maintain surveillance, but the land can be returned to normal surface use. Even during the operational phase of the repository, surface rights might be leased selectively on some of the surface area overlying the 2,000-acre mine but lying outside the fenced facilities area. The temporarily committed surface area not directly above the mine (i.e., some 16,000 acres) will have no restrictions on surface use, but drilling and mining will be only selectively allowed. In some states, fee-simple acquisition of the 16,000 acres may be required. All other resource requirements seem to be typical of large mining operations or facilities handling nuclear material. The availability of water and power would have to be assessed on a site-specific basis.

Estimates of the maximum airborne emissions from repository construction and operation (excluding waste-related emissions) are listed in Table 7-3, including a comparison of estimated concentrations with occupational exposure guidelines. In every case the emissions are below occupational exposure guidelines, even though conservative assumptions were employed. Site-specific considerations will have to include the additive effect of this facility with others in the area (if there are any), and perhaps, for example, better treatment of the coal-fired-boiler exhaust would be required. All emission-control-equipment requirements are well within the present state of the art.

Estimates of the maximum waste-related releases to the atmosphere during normal operations are presented in Table 7-4. These releases originate from the contamination of canister surfaces and undetected leaking canisters. In every case, the concentrations released to the atmosphere are only a small fraction of the Federal limits for continuous release to unrestricted areas (10 CFR 20, Appendix B, Table II), even without considering atmospheric dispersion. Again, all emission-control equipment (roughing filters and double HEPA filtration) is well within the state of the art.

7.1.4 LICENSING STATUS

A pivotal issue in waste management is the means of providing long-term permanent storage for nuclear wastes in a manner that best ensures their isolation from the biosphere. Analyses spanning two decades have generated the widely supported concept of isolating nuclear wastes in deep geologic formations.

The concept of disposal in deep geologic formations was first advanced in 1957, when an advisory committee of the National Research Council-National Academy of Sciences suggested the burial of solid radioactive wastes in salt deposits. To date, most of the research, development, and demonstration activities have been in salt. The current DOE National Waste Terminal Storage (NWTS) program calls for the selection by 1979 of two sites overlying suitable salt formations, followed by the construction and startup in 1985 of one NRC-licensed repository designed for the permanent disposal and isolation of commercial nuclear wastes in a salt formation at one of these two sites. At this time, no deep geologic repository has been licensed for the disposal of spent-fuel assemblies in either a nonretrievable or a retrievable mode.

7.1.5 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

Listed below are some of the major research, development, and demonstration efforts that are necessary for ensuring that spent fuel can be safely disposed of in a geologic repository.

1. If ready retrievability of the spent fuel is to be required, a major engineering requirement is that the length of such a period be established early in the design phase and held constant during the final design stage (Ref. 2).
2. Two major uncertainties about rock behavior as it applies to a repository are thermomechanical interactions in the immediate vicinity of the spent fuel and thermal-hydrologic interactions in the host rock and the surrounding strata (Ref. 2).
3. Major uncertainties exist in the data base for nuclide leaching from spent fuel, the groundwater chemistry to be expected at repository depths, nuclide solubility in the groundwater, and nuclide transport through the various rocks. These uncertainties are of particular concern for designs in which the host rock is not salt (Ref. 2).

All aspects of mine excavation, containerized spent-fuel handling and emplacement, interim storage, and retrievability appear to be amenable to engineering solution. Analyses should be made to find the best techniques for ensuring canister integrity during the retrievable phase. The thermal restrictions used during these designs before the final concept is chosen, especially those associated with the maximum canister and spent-fuel temperatures, should be verified. The values of thermal properties of the rocks need to be verified by in-situ environments, and the fracturing potential of the various rocks other than salt under thermal stress need analytical and experimental investigation. Such analyses, coupled with a better understanding of the heat-removal potential of a repository ventilation system, can lead to acceptable repository designs.

The concept of ready retrievability assumed requires that the spent fuel be retrievable for a defined period of time with approximately the same effort and at approximately the same rate at which it was emplaced. During this period, the repository room must remain open to waste-removal equipment, and the spent-fuel containers must retain their integrity. Such requirements have considerable impact on repository design and operation and therefore need to be specified early in the design phase. Retrievability periods of varying lengths can dictate different design approaches; thus, the period must be defined and held constant throughout the final design stages. For example, these studies show that the alternative of 25-year retrievability appears to be achievable through modifications in the engineering design of the reference repository and/or by reducing the thermal load during the extended retrievable period. The optimum approach to retrievability from the various host rocks should be determined through further investigation.

Before detailed repository designs can be completed, an indication of Federal, State, and local requirements that may be imposed on the repositories must be forthcoming. These include potential requirements regarding the safeguarding of nuclear materials, guidelines on acceptable environmental releases, licensing requirements, and such policy-oriented decisions as the length of any retrievability period.

7.1.6 DECOMMISSIONING AND DECONTAMINATION

Decommissioning, sealing, and long-term monitoring studies are presently being conducted under the direction of the DOE in an attempt to analyze these aspects of repository operation and waste isolation (Ref. 2). Although preliminary results from some of these studies may be incorporated into the generic environmental impact statement, no additional work in these areas has been performed under the preconceptual alternative repository design studies.

Facilities at provisional storage repository sites should be designed, built, and operated to provide for the ready retrieval or recovery of the emplaced radioactive-waste containers (Ref. 1). If the retrievability option is not exercised, the repository would be converted to permanent storage and facilities could be decommissioned. The various options for decommissioning such facilities as reactors or fuel-reprocessing plants can be used to decommission facilities at geologic storage sites.

The feasibility of using dismantling to decommission a geologic repository was demonstrated on a small scale in Project Salt vault (Ref. 1). In this experiment, 21 canisters containing about 200,000 curies of activity were emplaced, stored for about 19 months, and retrieved; and the mine was decontaminated to radiation levels sufficiently low to permit its return to its owners with no restrictions on use.

Procedures for converting a provisional storage repository to permanent storage will provide for (a) backfill of the underground workings, (b) sealing of the underground workings, (c) decommissioning of surface facilities, and (d) site control after decommissioning.

a. Backfill of the Underground Workings. The magnitude of the deformation and other geologic disturbances (i.e., subsidence, room closure, etc.) that can occur around the underground workings within the protective geologic formation can be reduced by backfilling the storage rooms, access corridors, etc. While the technology for backfilling exists, the backfill material and the extent of the backfill will depend on the type of waste, the emplacement method, and the rock formation.

b. Sealing of the Underground Workings. The mine shafts, boreholes, and other access ways from the surface to the underground workings penetrate the protective geologic formation, compromising the containment to some degree. It will be necessary to seal these penetrations in such a way that the ability of the seal to preserve the integrity of the repository containment will equal or surpass that of the protective rock formation surrounding the repository. The technique, criteria, and testing methods for placing and monitoring such seals are being developed (Ref. 1).

c. Decommissioning of Surface Facilities. After the boreholes, shafts, and other accessways are sealed, the surface facilities will be dismantled and removed. This will eliminate any problems with the leakage of radioactive material from the surface facilities (Ref. 1).

d. Site Control After Decommissioning. Site-control procedures would be based on Government ownership of the land and would include appropriate fencing, warning signs, and a low level of surveillance. Surveillance would include the actual but infrequent patrol of the restricted areas as well as the monitoring of the level of the land surface over the underground workings, groundwater, and a few fairly deep wells that approach but do not penetrate the geologic formation (Ref. 1).

Table 7-1. Estimated maximum accidental release rates--baseline waste data[a]

Nuclide	Maximum stack release rate (Ci/sec)		Estimated maximum 24-hr average concentration in unrestricted area[b] (Ci/ml)	Releases relative to Federal limits[c]
	Initial	24-hr average		
Tritium	3.1×10^{-2}	2×10^{-4}	1.1×10^{-7}	0.55
Carbon-14	9.3×10^{-5}	5×10^{-7}	2.8×10^{-10}	0.003
Krypton-85	2.4	1.3×10^{-2}	7.1×10^{-6}	18.3

[a] Details of these release estimates can be found in Ref. 8, Section 3.0.

[b] The computation of the maximum concentration in the unrestricted area assumes worst-case atmospheric conditions (fumigation).

[c] Federal limits are from Table II of Appendix B, 10 CFR 20. A 24-hour notification accident is one that exceeds these limits 500 times.

Table 7-2. Commitment of natural resources for generic repository operations[a]

Resource	Quantity	Comments
Land		
Temporarily committed	18,000 acres	Total surface area--disturbed and undisturbed.
Disturbed area	200 acres	Facility site; on decommissioning, this area will be returned to a natural state. Fee-simple acquisition of this land will be required to restrict drilling or mining activities.
Undisturbed area	1,800 acres	Area surrounding facility site; this area encloses the maximum extent of underground development. Fee-simple acquisition required. Surface rights may be selectively leased.
	16,000 acres	Annulus surrounding repository area; acquisition of subterranean rights (or fee-simple acquisition if necessary) required. No restrictions on surface usage.
Permanently committed	2,000 acres	Underground area required for repository; approximately a 200-ft formation thickness at a depth between 1,000 and 3,000 ft.
Excavated-rock disposal	15×10^6 tons	Salt repository design for spent unreprocessed fuel.
Water	40×10^8 gal	Includes water for decontamination, radwaste treatment, and general plant operation.
Fossil fuel		
Coal for electricity	1.4×10^6 MT	Equivalent to the consumption of a 30-MWe coal-fired power plant for 25 years.
Equivalent electricity	3.9×10^6 MW/hr	
Coal for boiler operation	1.2×10^6 MT	

[a] Data from Reference 2.

Table 7-3. Estimated maximum airborne emissions from baseline repository construction and operation[a]

Contaminant	Estimated maximum concentration	Fraction of occupational exposure guideline	Guideline
Radon-222 (from rock)	4×10^{-14} μCi/ml	10^{-6}	(b)
Radon-220 (from rock)	3×10^{-14} μCi/ml	10^{-7}	(b)
Hydrogen (corrosion, radiolysis)	37,000 ppm	--	(c)
Diesel exhaust			
Carbon monoxide	2 ppm	0.04	(d)
Nitrogen oxides (as NO)	17 ppm	0.7	(d)
HCHO (aldehydes)	0.04 ppm	0.02	(d)
Particulates	0.7 mg/m^3	0.14	(d)
Coal-fired boiler emissions			
Sulfur dioxide	0.6 lb/10^6 Btu	0.5	(e)
Nitrogen oxides	0.2 lb/10^6 Btu	0.3	(e)
Particulates	0.03 lb/10^6 Btu	0.3	(e)

[a] From Ref. 2. Details of these release estimates can be found in Ref. 8, Section 1.0, waste-related emissions not considered.

[b] Code of Federal Regulations, Title 10, Energy, Part 20, "Standards for Protection against Radiation," Appendix B, Table I, revised January 1, 1977.

[c] The lower combustible limit, from CRC Handbook of Chemistry and Physics, 50th edition (Ref. 9), is 40,000 ppm.

[d] From Ref. 10.

[e] Code of Federal Regulations, Title 40, Protection of the Environment, Part 60, "Standards for Performance for New Stationary Sources," revised July 1, 1976.

Table 7-4. Maximum waste-related release concentrations during normal operations--baseline waste data[a]

Contaminant	Airborne concentration[b] (μCi/ml)	Highest fraction of 10 CFR 20 guidelines[c]
Main stack release[d]		
Beta- and gamma-emitting particulates	4×10^{-17}	0.04
Alpha-emitting particulates	6×10^{-19}	0.004
Tritium	3×10^{-11}	0.0002
Carbon-14	3×10^{-14}	3×10^{-7}
Krypton-85	2×10^{-9}	0.007
Low-level-waste building		
Beta- and gamma-emitting particulates	3×10^{-18}	0.004
Alpha-emitting particulates	1×10^{-19}	0.0001

[a] Details of these release estimates can be found in Ref. 8, Section 2.0.

[b] Assumes double high-efficiency particulate air filter filtration. Concentration on release from stack, no atmospheric dispersion.

[c] These guidelines (10 CFR 20, Appendix B, Table II) are for exposure in unrestricted area. These values are a direct comparison of stack releases; concentrations at the repository boundary would be even lower because of atmospheric dispersion.

[d] Main stack processes air from the canistered-waste building and the mine. A ventilation flow of 341,000 cfm is assumed.

7.2 WASTE DISPOSAL 2: GEOLOGIC DISPOSAL OF WASTE

7.2.1 GENERAL DESCRIPTION

A geologic repository for storing high-level waste along with intermediate- and low-level transuranic waste and cladding hulls from the reprocessing of LWR spent fuel is similar to the spent-fuel repository discussed in Section 7.2.1. The spiked recycle fuel cycle flow rates have been reviewed and compared with the generic data presented in References 2, 11, and 12, and it was concluded that, for the current state of the art, the Reference 2 data and design descriptions are applicable. The LWR heavy-metal waste generated annually is about 260 kg, and the fission products are about 838 kg. The applicable repository design source terms for the recycle fuel-fabrication plant and fuel-reprocessing plant are defined in Tables B-1, B-2, and B-4 of Reference 12. Table 7-5 compares the baseline salt-repository excavated-material summary for this waste-disposal requirement and that for the disposal of spent fuel described in Section 7.1. Table 7-6 compares the baseline salt-repository capacities for the plutonium/uranium spiked recycle and the fuel-cycle scenarios of Section 7.1.

7.2.2 SAFETY CONSIDERATIONS

A number of accident scenarios have been evaluated, and the amounts of radioactive material that could potentially be released have been estimated under conservative assumptions. The estimated maximum releases in each category of interest are presented in Table 7-7. These maximum releases encompass all the accident scenarios. A major breach of a canister containing calcined high-level waste and a subsequent complete spillage of its contents would generate the greatest predicted accidental release. However, the magnitude of even this release is such that notification of the director of the appropriate NRC regional office is required only within 24 hours of the accident; immediate notification is not required. An accident involving vitrified high-level waste was predicted to release only about 25 times the 10 CFR 20 limit for continuous release of beta- and gamma-emitting particulates to the unrestricted area. These estimates of maximum 24-hour average concentrations in an unrestricted area occur at the repository 200-acre fenceline and assume worst-case atmospheric conditions (fumigation). Site-specific calculations will have to be performed when detailed designs are available.

7.2.3 ENVIRONMENTAL CONSIDERATIONS

Section 7.1 generally defines the total commitment of natural resources for the generic repository operations and the estimated maximum airborne emissions from the construction and operation of the baseline repository (see Tables 7-2 and 7-3). Note, however, that the quantity of salt excavated for the storage of waste from plutonium/uranium recycle is 19 x 10^6 tons rather than 15 x 10^6 tons for spent-fuel disposal.

The estimated maximum waste-related releases to the atmosphere during normal operations are presented in Table 7-8. These releases arise from contamination of canister surfaces and undetected leaking canisters. In every case, the concentrations as they are released to the atmosphere are a small fraction of the Federal limits for continuous release to unrestricted areas (10 CFR 20, Appendix B, Table II), even without considering atmospheric dispersion. Again, all emission-control equipment (roughing filters and double high-efficiency filtration) is well within the state of the art.

7.2.4 LICENSING STATUS

To date, no deep geologic repository has been licensed for the disposal of either transuranic or high-level waste.

7.2.5 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

Listed below are some of the major research, development, and demonstration efforts that are necessary for insuring that high-level waste can be safely disposed of in a geologic repository.

1. If ready retrievability of the waste is to be required, a major engineering requirement is that the length of such a period be established early in the design phase and held constant during the final design stage (Ref. 2).
2. Two major uncertainties about rock behavior as it applies to a waste repository are thermomechanical interactions in the immediate vicinity of the waste and thermal-hydrologic interactions in the host rock and the surrounding strata (Ref. 2).
3. Major uncertainties exist in the data base for nuclide leaching from spent fuel or the various waste forms, the groundwater chemistry to be expected at repository depths, nuclide solubility in the groundwater, and nuclide transport through the various rocks. These uncertainties are of particular concern for designs in which the host rock is not salt (Ref. 2).

All aspects of mine excavation, containerized waste handling and emplacement, interim storage, and retrievability appear to be amenable to engineering solution. Analyses should be made to find the best techniques for ensuring canister integrity during the retrievable phase. The thermal restrictions used during these designs before the final concept is chosen, especially those associated with the maximum canister and waste temperatures, should be verified. The values of thermal properties of the rocks need to be verified by in-situ environments, and the fracturing potential of the various rocks other than salt under thermal stress need analytical and experimental investigation. Such analyses, coupled with a better understanding of the heat-removal potential of a repository ventilation system, can lead to acceptable repository designs.

The concept of ready retrievability assumed requires that the waste be retrievable for a defined period of time with approximately the same effort and at approximately the same rate at which it was emplaced. During this period, the repository room must remain open to waste-removal equipment, and the waste containers must retain their integrity. Such requirements have considerable impact on repository design and operation and therefore need to be specified early in the design phase. Retrievability periods of varying lengths can dictate different design approaches; thus, the period must be defined and held constant throughout the final design stages. For example, these studies show that the alternative of 25-year retrievability appears to be achievable through modifications in the engineering design of the reference repository and/or by reducing the thermal load during the extended retrievable period. The optimum approach to retrievability from the various host rocks should be determined through further investigation.

Before detailed repository designs can be accomplished, an indication of Federal, State, and local requirements that may be imposed on the repositories must be forthcoming. These include potential requirements regarding the safeguarding of nuclear materials, guidelines on acceptable environmental releases, licensing requirements, and such policy-oriented decisions as the length of any retrievability period.

7.2.6 DECOMMISSIONING AND DECONTAMINATION

Section 7.1.6 describes the decommissioning and decontamination of a geologic repository and is applicable to a high-level waste repository.

Table 7-5. Material excavated, backfilled, and to be disposed of--waste repository in bedded salt[a]

Cycle	Volume mined (10^6 ft^3)	Weight mined (10^6 tons)	Total backfill (10^6 tons)	Permanent disposal (10^6 tons)
Recycle	677	45	26	19
Spent-fuel disposal	526	35	20	15

[a] Data from Ref. 2.

Table 7-6. Baseline salt-repository capacities

Cycle	Waste Type	Capacity (number of containers)	Repository area (acres)	Power per container (kW)	Average power density (kW/acre)	Power delivered to repository (MW)	Fuel equivalence of storage capacity (MTHM)
Pu/U recycle[a]	High-level waste (vitrified)	41,340	1,018	2.8	148.0	116.0	86,400
	Intermediate-level transuranics	274,485	649	0.0003	0.164	0.08	151,000
	Cladding waste	57,055	135	0.022	12.9	0.14	139,800
	Low-level transuranics	571,200	108	0.0002	—	0.11	—
Spent-fuel disposal[b]	PWR fuel	134,648	700	0.55	125.0	74.0	62,000
	BWR fuel	200,016	1,035	0.18	41.0	36.0	36,600
	Low-level transuranics	60,192	97	0.0002	—	0.01	—

[a] Repository to be filled by the year 2006.
[b] Repository to be filled by the year 2008.

Table 7-7. Estimated maximum accidental release rates--baseline waste data[a]

Contaminant	Maximum stack release rate (Ci/sec)		Estimated maximum 24-hr average concentration in unrestricted area[b] (μCi/ml)	Federal limits[c]
	Initial	24-hr average		
Beta- and gamma-emitting particulates	4.8×10^{-6}	2.5×10^{-8}	1.4×10^{-11}	700
Alpha-emitting transuranic particulates	1.6×10^{-7}	8.3×10^{-10}	4.6×10^{-13}	23

[a] Details of these release estimates can be found in Ref. 8, Section 3.0. The estimates are for calcined high-level waste. Maximum accidental releases from vitrified high-level waste would be 20 to 30 times less.

[b] The estimated maximum concentration in the unrestricted area assumes worst-case atmospheric conditions (fumigation).

[c] The Federal limits are from Appendix B, Table II, of 10 CFR 20. A 24-hour notification accident is one that exceeds these limits 500 times.

Table 7-8. Maximum concentrations of waste-related releases during normal operation--baseline waste data[a]

Contaminant	Airborne concentrations[b] (μCi/ml)	Highest fraction of Federal guidelines[c]
Main stack release[d]		
Beta- and gamma-emitting particulates	7×10^{-16}	0.04
Alpha-emitting particulates	7×10^{-17}	0.004
Low-level-waste building		
Beta- and gamma-emitting particulates	7×10^{-17}	0.004
Alpha-emitting particulates	2×10^{-18}	0.0001

[a] Releases estimated under the assumption of double HEPA filtration.

[b] Concentration on release from stack, no atmospheric dispersion.

[c] The guidelines (10 CFR 20, Appendix B, Table II) are for exposure in unrestricted area. These values are a direct comparison of stack releases; concentrations at the repository boundary would be even lower because of atmospheric dispersion.

[d] Main stack processes air from the canistered-waste building and the mine. A ventilation flow of 341,000 cfm is assumed.

7.3 WASTE DISPOSAL 3: SHALLOW LAND DISPOSAL OF LOW-LEVEL WASTE

7.3.1 GENERAL DESCRIPTION

Low-level wastes are generated at fuel-fabrication plants, nuclear power plants, and reprocessing plants during the conversion of uranium hexafluoride to uranium dioxide and by the preparation of fuel elements from uranium dioxide (Ref. 12); the most significant solid radioactive waste of the fabrication plant is contaminated calcium fluoride (Ref. 13). Water from the air-scrubber systems contains most of the fluoride released during operation. The combined liquid effluent from the fluoride-to-oxide conversion process and scrap-recovery operations is treated with lime to form a calcium fluoride precipitate. The calcium fluoride is filtered from the waste stream, and the water is released to a holding pond or lagoon. The calcium fluoride is buried in onsite trenches, which are backfilled. The total quantity produced is estimated at about 26 MT (volume ~8.4 m^3) per 1,000 MWe-yr. The calcium fluoride buried at the site contains about 0.06 curie of uranium per 1,000 MWe-yr (Ref. 13).

There are also miscellaneous low-level wastes from all nuclear fuel-cycle facilities. Combustible wastes are incinerated, and the residual ash constitutes an additional solid waste. Miscellaneous wastes that contain a significant amount of uranium are processed in the scrap-recovery system to recover the uranium.

The solid wastes consist of a wide assortment of solid materials, including items such as paper, cloth, wood, plastic, rubber, glass, ceramic, and metal, as well as salts and sludges produced in the treatment of liquid-waste streams, and filters from the cleanup of offgases and contaminated equipment (Ref. 14). The density of the uncompacted wastes would vary from about 2 to as much as 200 lb/ft^3. About one-half to two-thirds (by volume) of these wastes would be combustible; incineration could reduce much of this by about 50 and about 20 times in volume and weight, respectively. About one-half to three-fourths of the wastes could be reduced in volume by factors of 2 to 10 by compaction (Ref. 14).

About 60% of the solid waste would be placed in 55-gallon drums; the balance would be packaged in steel boxes whose approximate dimensions would be 4 by 6 by 6 feet. This material can be shipped by any number of methods that would protect the waste during shipment (Ref. 14).

The liquid effluents from process vessels and miscellaneous liquids from operating areas contain dissolved and suspended compounds of uranium, thorium, and protactinium. In most of the process systems the relative proportions of uranium, thorium, and protactinium in the liquid effluents will be the same as in the solids formed in a given operation. The liquids are filtered, evaporated, and solidified before packaging in 55-gallon drums for disposal.

7.3.2 SAFETY CONSIDERATIONS

The quantity of radioactive material released to the environment in the shallow land disposal of low-level waste is small in comparison with the overall fuel cycle, and the concentration of uranium and/or thorium is insignificant. Thus, according to the Federal regulations currently in effect, there are no major safety problems associated with the disposal of waste from the fabrication plant.

7.3.3 ENVIRONMENTAL CONSIDERATIONS

A shallow land disposal facility for low-level wastes receives solidified, packaged wastes in drums or boxes for burial. Reactor components containing activation products making them highly radioactive are received in shielded shipping casks. They are removed semiremotely (behind a barrier) and buried in a deep trench. All containers received must be free of surface contamination to prevent spread of contamination. The buried trenches are equipped for directional drainage to a sampling sump where routine checks are made for the presence of liquids. As received, packages must be covered with sufficient earth to reduce the radiation level at the trench being filled.

The burial sites are located remotely and are fenced for controlled access. All persons working on or visiting the sites are monitored for radiation exposure; areas with levels above 100 mrem/hr are monitored by health physics personnel until the exposed waste is covered with dirt to reduce the radiation level. The environmental impacts associated with waste-burial facilities are dust and noise during construction of the offices and laboratory, periodic excavations and filling of trenches, the presence of an exclusion fence around the area, the permanent commitment of land, and the possibility of the movement of small quantities of radioactivity into the groundwater. The release of heat to the environment is not a problem because of the very low heat-generation rate of nontransuranic wastes (Ref. 12).

Monitoring wells at each burial site are sampled periodically to check for migration of radioactivity. In addition, air and vegetation samples are taken from the site area for analysis.

Accidental opening of a waste package and release of the contents of the package prior to burial could occur. Normally, the scattered waste would be confined to the immediate vicinity of the accident, within the fenced area, and the impact on the environment would be negligible.

7.3.4 LICENSING STATUS

The current methods used for the disposal of low-level waste from LWR fuel-fabrication plants have been acceptable from a licensing standpoint in the past. However, according to Reference 15 and rulemaking that may develop as a result of this document, current methodologies of disposal may be modified in the future.

7.3.5 RESEARCH, DEVELOPMENT, AND DEMONSTRATION (RD&D)

Some RD&D may be required to ensure that waste releases from the LWR fuel-fabrication facility are in accordance with the "as low as reasonably achievable" concept. Furthermore, according to Reference 15, modified rulings on shallow land burial may require extensive RD&D effort to meet new regulations--if required.

It can be concluded that additional RD&D, discussed in Reference 16, yet undefined, and pending general RD&D requirements described in Reference 17, will be required for low-level-waste shallow land burial.

7.3.6 DECOMMISSIONING AND DECONTAMINATION

Decommissioning and decontamination has been recently recognized as an important aspect of every phase of the nuclear fuel cycle. The DOE has drafted the Generic Program Plan, Formerly Utilized Manhattan Engineering District/Atomic Energy Commission

Sites Remedial Action Program, also known as FUSRAP. It is planned to decontaminate and decommission the identified sites to a level for general use, to decontamination-level criteria yet to be established, and to dispose of the low-level wastes accumulated during the decontamination/decommissioning operations. Data to be developed during the action program may be applicable to decommissioning of shallow land burial sites used for burial of low-level waste. To date, a sufficient quantity of data applicable to decontamination and decommissioning disposal sites for low-level waste has not been identified. The importance of this effort has been discussed in Reference 17.

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Chapter 8

TRANSPORTATION

The various steps in the nuclear fuel cycle such as mining, milling, conversion, enrichment, fabrication, reprocessing, and waste disposal are connected to each other by transportation and require the shipment of radioactive materials. This chapter describes the various transportation steps from and among the fuel-cycle facilities.

The discussion that follows is segregated into two major parts--transportation of (1) low-specific-activity material and (2) high-specific-activity material.

8.1 LOW-SPECIFIC-ACTIVITY MATERIAL

8.1.1 GENERAL DESCRIPTION

For the fuel cycles considered in the Preliminary Safety and Environmental Information Document (PSEID) Volumes I through VI, low-specific-activity materials that are transported to or from facilities involved in the nuclear fuel cycle include uranium and thorium ores, unirradiated uranium (natural or depleted), uranium and thorium concentrates, natural UF_6, enriched UF_6, and waste streams containing these materials. All other radioactive materials in the fuel cycle would normally be classified as high-specific-activity, unless diluted to meet the criteria for low-specific-activity as specified in 10 CFR 71.

Some radioactive wastes, primarily from the operation and maintenance of reactors and reprocessing facilities, do not exactly fit into either category of high- or low-specific-activity materials. Such materials might be sufficiently radioactive to require that they be mixed with concrete before shipment in ordinary 200-liter steel drums, or possibly the drums themselves would have an outer concrete shield several inches thick. The levels of radioactivity of these medium-level wastes are not high enough, however, to require special provisions for removal of heat or to require the massive shielding of special shipping casks, such as those required for high-specific-activity spent fuel. In assessing the safety impacts of the transportation steps in the fuel cycle, this section classifies wastes as low-specific-activity materials if they can be transported in either lightly shielded or unshielded drums.

Ore is usually in the form of sandstone or limestone rock and transported from the mine to the mill in bulk by truck or rail. To minimize transportation costs, mills are generally located close to the mines--the average shipping distance has been assumed to be 5 miles. Average annual transportation requirements per 1,000 MWe for ore in the light-water reactor (LWR) once-through cycle have been estimated at 91,000 MT, requiring 3,350 shipments and 16,800 truck miles (Ref. 1).

The mill product is a powder concentrate of U_3O_8, called "yellowcake." If the uranium is to be enriched, the yellowcake is shipped to another facility for conversion into UF_6. Thorium concentrates would be shipped directly to a fuel-fabrication plant. Typical shipments of concentrates are in 55-gallon steel drums, with 40 drums per truck shipment. Average annual transportation requirements per 1,000 MWe for concentrates in the LWR once-through cycle have been estimated as 183 MT U_3O_8, requiring 12 shipments and 12,000 truck miles (Ref. 1).

Unenriched UF_6 is packaged in cylinders sized to contain either 10 or 14 ST of UF_6. A typical truck shipment would consist of one 14-ton cylinder or two 10-ton cylinders. Average annual transportation requirements per 1,000 MWe for shipment of natural UF_6 to the enrichment plant have been estimated as 270 MT UF_6, requiring 22 shipments and a total of 11,000 truck miles for the LWR once-through cycle (Ref. 1).

Low-enrichment UF_6 is packaged and shipped as Type A fissile material, typically in 2.5-ton cylinders, 5 cylinders per truck shipment, and 750 truck miles between the enrichment and fuel-fabrication plants. Average annual enrichment UF_6 shipping requirements per 1,000 MWe in the LWR once through fuel cycle have been estimated as 52 MT UF_6, requiring 5 shipments and 3,750 truck miles (Ref. 1).

Low- and medium-level wastes arise from all processing steps in the fuel cycle. The waste volumes and number of truck shipments required for such wastes for the LWR once-through fuel cycle have been analyzed in Reference 2. Average annual transportation requirements per 1,000 MWe are estimated to be 13 shipments and 6,500 truck miles.

In summary, a total of about 50,000 truck miles are associated yearly with shipments of low-specific-activity material for the LWR once-through fuel cycle. The majority of this transportation is related to movement of natural uranium (about 40,000 truck miles). Since all other fuel cycles considered herein use less natural uranium than the LWR once-through fuel cycle, their transportation requirements are assumed to be less than that of the LWR once-through system. Some systems requiring higher enrichments of uranium-235 will require more shipments of enriched UF_6, but this increase is not likely to affect the reduction in shipments of natural uranium.

At the other extreme from the LWR once-through fuel cycle are the breeder systems. The following paragraphs summarize transportation distance of low-specific-activity material for the uranium/plutonium fueled liquid-metal fast-breeder reactor (LMFBR).

For the LMFBR cycle, the average annual quantity of heavy metal (depleted U or Th) that must be transported per 1,000 MWe is in the range of 32 to 35 MT. Assuming 14 to 20 MT per shipment and 500 miles per shipment, the shipping requirement for this low-specific-activity material is 2 shipments or 1,000 truck miles.

For the LMFBR, annual shipments of low- and medium-level waste per 1,000 MWe from fabrication and enrichment facilities have been estimated (Ref. 3) as 56 truck shipments and 5.8 rail shipments, corresponding to 28,000 truck miles and 5,800 rail miles.

In summary, a total of about 29,000 truck miles and 5,800 rail miles are involved yearly in shipping of low-specific-activity material in the uranium/plutonium LMFBR fuel cycle.

8.1.2 SAFETY CONSIDERATIONS

The primary safety consideration in transportation of low-specific-activity material is vehicular accidents that cause death or injury to transportation workers or to the public. Compared to the former issue, radiological safety of transportation systems is not a key issue nor is it a major contributor to radiological dose in the whole fuel cycle (see Section 8.1.3).

Statistics on truck accidents have been analyzed (Ref. 4) to give 1.7×10^{-6} accidents per vehicle mile, 0.51 injuries per accident, and 0.03 fatalities per accident. Statistics on rail accidents have also been analyzed (Ref. 4) to give 1.3×10^{-7} accident per vehicle mile, 2.7 injuries per accident, and 0.2 fatalities per accident for a single rail car. For the once-through light-water reactor cycle low-specific-activity materials transportation requirements identified in Section 8.1.1, the annual accident rate per 1,000 MWe would be 0.085, resulting in 0.043 injuries and 0.0026 fatalities.

Although the LMFBR has increased requirements for transportation of low- and medium-level wastes, the overall LMFBR transportation requirements are less than the once-through light-water reactor fuel cycle.

As described in Section 8.1.1, truck transportation requirements for low-specific-activity for the LMFBR cycle are only 59 percent of the requirements for the LWR once-through cycle. The LMFBR cycle, however, has an additional requirement for 5,800 rail miles for shipments of low-specific-activity materials. The annual accident rate per 1,000 MWe would be 0.050, resulting in 0.27 injuries and 0.0016 fatalities.

The packaging requirements and transportation requirements for low-specific-activity material are such that it is most unlikely that a person would ingest or inhale a sufficient mass of material, under any circumstances arising in transport, to cause a significant radiation hazard. Fissile materials are so limited and the packages are so designed as to ensure nuclear criticality safety under both normal and accident considerations of transport.

8.1.3 ENVIRONMENTAL CONSIDER[illegible]

The environmental impact of transportation of low-specific-activity material is minimal. The adverse environmental effects of the shipment of low-specific-activity materials in the nuclear fuel cycle are principally those that are characteristic of the trucking industry in general. The discussion that follows was extracted from Reference 1 and includes the impact of shipment of high-level waste to a final disposal site by rail. Inclusion of this step in this discussion does not alter the conclusions because (1) the impacts are so small, and (2) the contribution of high-level wastes to the impacts are negligible.

All shipments of low-specific-activity material are assumed to be by truck. The increase in density of truck traffic from shipments of nuclear fuel-cycle low-specific-activity material and high-level wastes will be small compared with total truck traffic. Annual diesel fuel requirements per 1,000 MWe are on the order of 10^{-6} of the diesel fuel used by the trucking industry. The NO_x release from diesel fuel combustion in truck shipments is a small fraction (<1%) of the NO_x release from the fuel cycle.

The radiation exposure to onlookers, people along the route, and transportation workers due to the transportation requirements of the nuclear fuel cycles, considering both normal and accident conditions, is negligible compared with radiation exposure arising from the remainder of the fuel cycle.

8.1.4 LICENSING STATUS

No licensing problems are foreseen for transportation of low-specific-activity materials in the fuel cycle.

Packaging and transport of radioactive materials are regulated at the Federal level by the U.S. Department of Energy (DOE), the U.S. Department of Transportation (DOT), and the U.S. Postal Service. Certain aspects, such as limitations on gross weights of trucks and on transportation not subject to Federal regulations, are regulated by the states. Most states have adopted regulations pertaining to intrastate transportation of radioactive materials that require the shipping to conform to the packaging, labeling, and marking requirements of the DOT to the same extent as if the transportation were subject to the rules and regulations of that agency.

Federal packaging standards and criteria are found in 10 CFR 71 and 49 CFR 170-179.

8.1.5 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

No research, development, and demonstration needs are foreseen in connection with the transportation of low-specific-activity materials in the fuel cycle.

8.1.6 DECOMMISSIONING AND DECONTAMINATION

The impacts of decommissioning and decontamination requirements for materials associated with transportation of low-specific-activity materials in the fuel cycle have not been assessed. However, these are expected to be small relative to decommissioning and decontamination impacts in other portions of the nuclear fuel cycle.

8.2 HIGH-SPECIFIC-ACTIVITY MATERIAL

8.2.1 GENERAL DESCRIPTION

The high-specific-activity materials that are transported to and from facilities involved in the nuclear fuel cycles described in PSEID Volumes I through VI include fresh fuel, recycle fuel, spent fuel, recovered plutonium, uranium-233, thorium for recycle, and high-level wastes. Recycle fuel includes fuels fabricated from uranium-233 in which residual uranium-232 contributes to a high gamma radioactivity, spiked plutonium to which cobalt-60 or other spikant has been added to provide a high gamma radioactivity, or fabricated fuel that has been preirradiated.

Fresh fuel does not require heavy shielding and is packaged in steel containers for protection against shipping damage. Spacing between assemblies is maintained in order to prevent accidental criticality. Shipment is usually by truck.

Recycle fuel requires shielding to limit exposure during handling and shipment. Overpacks or modified casks would be utilized to protect the fuel from damage and to provide shielding.

Recycle uranium-233, spiked plutonium, and recycle thorium will require shielded packages for shipment. A packaging and transportation system for recycle-fuel materials has not been developed. These materials would not be shipped commercially if both the reprocessing and refabrication plants are colocated; otherwise, they would probably be shipped by special trucks.

Spent fuel will be shipped by rail or truck cask. Table 8-1 presents the properties of a reference LWR spent-fuel assembly after 120 days of cooling. Table 8-2 presents a listing of the available licensed, present-generation spent-fuel shipping casks. Specific cask designs have been developed for other fuel cycles but only a cask for the highly enriched high-temperature gas-cooled reactor (HTGR) fuel has been licensed.

High-level wastes include alpha-beta-gamma waste, cladding hulls and fuel hardware, compressed noble gases, and immobilized high-level liquid waste. The alpha-beta-gamma wastes contaminated with transuranics in excess of 10 Ci/kg have been packaged in shielded containers by remote-handling techniques. Typical surface dose rates lie between 10 and 1,000 mrem/hr.

Cladding hulls and fuel hardware are radioactive from neutron-induced isotopes and may be contaminated with transuranics in excess of 10 Ci/kg. They would be placed in disposable steel containers and shipped by rail in a cask.

Noble gases are expected to be compressed and loaded into gas cylinders for shipment in a water-filled tank. The water will provide shielding and heat dissipation. Shipment would be by truck.

Solidified high-level aqueous wastes contained in disposable steel canisters with a welded closure are to be shipped in a rail cask.

Table 8-3 presents the estimated shipping vehicular miles per GWe/year of reactor operation for both low- and high-specific-activity materials to show a comparison of the transportation mileage between the LWR (once-through) and the LMFBR fuel cycles. The shipping distances (vehicular miles per GWe-yr) of other fuel cycles may be of the same order of magnitude as those shown in Table 8-3 depending on the assumption made regarding the method of shipping (truck or rail). In particular, truck shipments of materials could substantially increase the vehicle-miles for both the light-water breeder reactor (LWBR) and HTGR. These reactor types require more fuel shipments than the LWR and LMFBR reactors because of the larger quantities of LWBR fuel to be shipped (very low burnup) and the fact that the moderator (graphite) is part of the HTGR fuel, which results in a volume limitation per shipment.

8.2.2 SAFETY CONSIDERATIONS

The primary safety consideration in transportation of high-specific-activity material is vehicular accidents that cause death or injury to transportation workers or to the public, the same issue discussed under Section 8.1.2, Safety Considerations of low-specific-activity material.

Estimates of the accident rate, injuries, and fatalities from transportation in the LWR once-through and LMFBR fuel cycles (shown in Table 8-4) are obtained by combining the vehicular safety statistics from Section 8.1.2, with the estimated shipping distances (in vehicular-miles per Gwe-yr) from Table 8-3.

The radiological safety of shipments of high-specific-activity wastes has not been fully evaluated. However, a detailed comparative evaluation has been performed of the LWR fuel cycle with and without recycle of uranium and plutonium (Ref. 5). This study concluded that the radiological dose (both occupational and to the general population) contributed by all of the transportation steps in the fuel cycle was less than 0.1% of the dose from all sources in the fuel cycle. For this reason, it is concluded that radiological safety is not a major issue in transportation of high-specific-activity materials.

8.2.3 ENVIRONMENTAL CONSIDERATIONS

The same environmental factors result from transportation of high-specific-activity material as those identified in Section 8.1.3 for low-specific-activity material--vehicle traffic density and radiological dose to the general public during normal and accident situations.

Specific data on number of shipments and the vehicular miles for the LWR and LMFBR fuel cycles indicate that the figures are comparable to those for low-specific-activity material (Table 8-3). Furthermore, the discussion in Section 8.2.2 of both occupational and general public radiation exposure indicates a minimal impact in this area for the whole fuel cycle. Therefore, the earlier conclusion that the impacts of the transportation of low-specific-activity material will have minimal environmental effects applies here also.

8.2.4 LICENSING STATUS

The discussion in Section 8.1.4 applies to high-specific-activity material except that shipping containers for other than LWR fresh and spent fuel and for highly enriched HTGR fresh and spent fuel have not been licensed. In addition, certain safeguards requirements may be placed on the shipment of sensitive fissile materials as the need evolves for regulation of these materials. (Note that safeguards, per se, are not discussed in these volumes.)

It is anticipated that the criteria now applied to design of shipping containers for the LWR fuel cycle can be applied to other fuels and fuel cycles. No reason is known that would preclude the licensing of containers and casks for high-specific-activity material for other fuels and fuel cycles.

8.2.5 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

No research, development, and demonstration needs are foreseen in connection with the transportation of high-specific-activity materials other than issues which may be identified for LWR spent fuel. At the present time, the DOE is conducting research on spent-fuel cask integrity during high-speed impacts. To date, the results do not indicate any problems with existing designs or design criteria.

8.2.6 DECOMMISSIONING AND DECONTAMINATION

The discussion in Section 8.1.6 also applies to high-specific-activity material.

Table 8-1. Properties of a reference LWR fuel assembly cooled 120 days

Assembly total weight, kg	454
Uranium per assembly, kg	311
Total radioactivity, Ci	1.9×10^6
Fission-product activity, Ci	1.7×10^6
Transuranic content, kg	5.5
Transuranic activity, Ci	1.1×10^5
Neutron source strength, neutrons/sec	1.3×10^9
Total thermal power, kW	8.4
Total gamma power, kW	1.4
Exposure, MWd/MTU	25,000
Average specific power, MW/MTU	35

Table 8-2. Available shipping casks for present-generation LWR spent fuel

	Usual transport	Weight	Length	Diameter	Working	Available Number			
Cask	mode	(lb)	(in.)	(in.)	length (in.)	Type	PWR	BWR	in 1978
IF-300	Rail	130,000	209.5	58.5	184	Wet	7	18	4
NLI 10/24	Rail	200,000		88	204.5	Dry	10	24	2
NFS-4[a]	Truck	49,000	214	50	202	Wet	1	2	6
NLI-1/2	Truck	47,500	227.25	42.5	193.25	Dry	1	2	5
TN-8	Truck	78,000	217	68	192	Dry	3	--	1
TN-9	Truck	77,600	227	68	202	Dry	--	7	1

[a] The NFS-5/6 cask is essentially the same as the NFS-4 cask, except that the lead shielding has been replaced with depleted-uranium shielding. Further, four of these casks are owned by NAC and can be redesignated as NAC-1 casks.

Table 8-3. Estimated shipping vehicle miles per GWe-year

Materials	LWR (once through)[a]		LMFBR[b]	
	Number of shipments	Vehicle-miles	Number of shipments	Vehicle-miles
		Low-activity materials		
Ore	3,350	16,800	--	--
U_3O_8	12	12,000	--	--
Natural UF^6	22	11,000	--	--
Enriched UF^6	5	3,750	--	--
Heavy metal	--	0	2	1,000
Low- and medium-level wastes	13	6,500	60	33,700
		High-activity materials		
Fresh fuel	5	5,000	--	--
Recycle fuel	--	0	20	15,000
Spent fuel	60	60,000	27	20,000
High-level wastes	--	0	29	29,250
Total		115,050		98,950

[a] Data from References 1, 2, and 6.
[b] Data from Reference 3.

Table 8-4. Annual accidents, injuries, and fatalities from transportation of high-specific-activity material (per 1 GW of generating capacity)

Reactor type	Accidents	Injuries	Fatalities
Light-water reactor, once-through cycle	0.111	0.056	0.0033
Liquid-metal, fast-breeder reactor	0.064	0.042	0.0026

REFERENCES FOR CHAPTER 8

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5. U.S. Nuclear Regulatory Commission, Final Generic Environmental Statement on the Use of Recycle Plutonium in Mixed Oxide in Light Water Cooled Reactors, NUREG-0002, Vol. 3, August 1976.
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Chapter 9

HEAVY-WATER-PRODUCTION FACILITIES

9.1 GENERAL DESCRIPTION

The initial inventory requirement of heavy water for the current CANDU-PHW design (500-MWe class) is about 0.9 to 1.0 MT/MWe (Refs. 1 and 2). It is anticipated that larger CANDU-PHW reactors (1,200-MWe class) will require less heavy water, about 0.75 MT/MWe (Ref. 1). CANDU reactors employing enriched fuel would require somewhat less heavy water. Makeup requirements resulting from losses are generally less than 1% per year but may run as high as 3% in some cases (Refs. 3 and 4). Thus new heavy-water-production plants will be required only if heavy-water reactors are installed at an increasing rate.

As of September 1976, over 9,000 MT of heavy water had been produced worldwide (Ref. 5). The current free-world production capacity of heavy water is 1,650 MT/yr. This capacity may well exceed 2,650 MT/yr by 1980 and reach nearly 6,650 MT/yr by 1985 (Ref. 6).

A comprehensive summary of domestic and known foreign heavy-water-production plants is given in Table 9-1, reproduced from Reference 6. Some major features describing each of the plants include the annual production capacity (in metric tons of heavy water per year), operation status, and production-process characteristics.

Table 9-1. Summary of heavy-water-production plants[a]

Site, constructor, and operator	Nominal capacity (MT/yr)	Year of startup	Status	Process characteristics				
				Boundary conditions	Deuterium source	Deuterium extraction	Enrichment	
							Initial	Final
Rjukan, Norway Norsk Hydro	20	1934	Operating	PA	HY	(b)	WE	WD
Morgantown, U.S.A. Dupont, USAEC	15	1944	Shut down	SC	RW	WD	WD	WD
Trail, Canada Cominco	6	1945	Shut down	PA	HY	GC	WE	WE
Dana, U.S.A. Girdler, Dupont	490	1952	Shut down	SC	RW	GS	GS, DT	WD, WE
Savannah River, U.S.A. Girdler, Lummus, Dupont	190	1952[c]	Operating	SC	RW	GS	GS, DT	WD, WE
Frankfurt, Germany Linde, Hoechst	6	1958	Shut down	PA	HY	HD	HD	HD
Toulouse, France CEA, Air Liquide, ONIA	2	1959	Shut down	PA	HY	HD	HD	HD
Domat EMS, Switzerland Sulzer, Emser Werke	2	1960	Shut down	PA	RW	WE	WE	HD
Nangal, India Linde, DAE	14	1962	?	PA	RW	WE	WE	HD
Mazingarbe, France CEA/SCC/CCM/SAL	26	1968	Shut down	PA	SG	AH	AH, MO	AD
Port Hawkesbury, Canada, Lummus, CGE, AECL	360	1970	Operating	SC	RW	GS	GS, DT	WD
Bruce A, Canada Lummus, Ontario Hydro	720	1973	Operating	SC	RW	GS	GS, DT	WD
Glace Bay, Canada Canatom, AECL	360	1976	Operating	SC	RW	GS	GS, DT	WD
Baroda, India GELPRA, DAE	67	1977	Shut down	PA	SG	AH	AH, MO	AH, MO
Kota, India DAE	100	1978	Construction	SC	RW	GS	GS, DT	WD
Tuticorin, India GELPRA, DAE	71	1978	Construction	PA	SG	AH	AH, MO	AH, MO
Talcher, India UHDE, DAE	63	1978	Construction	PA	SG	AH	AH, DT	WD
Bruce B, Canada Lummus, Ontario Hydro	720	1978	Construction	SC	RW	GS	GS, DT	WD
LaPrade, Canada Canatom, AECL	720	1982	Construction	SC	RW	GS	GS, DT	WD
Bruce C&D, Canada Lummus, Ontario Hydro	1,440	1982	Design	SC	RW	GS	GS, DT	WD

Table 9-1. Summary of heavy-water-production plants[a] (continued)

Site, constructor, and operator	Nominal capacity (MT/yr)	Year of startup	Status	Process characteristics				
				Boundary conditions	Deuterium source	Deuterium extraction	Enrichment	
							Initial	Final
Bruce E&F, Canada Lummus, Ontario Hydro	1,440	?	Design	SC	RW	GS	GS, DT	WD
Atucha, Argentina CNEA	1	1980?	Design	?	?	?	?	?
Arroyitos, Argentina	18 225	1981 1985	Design	?	?	?	?	?
Sixth Plant, India DAE	?	1985?	Design	PA	SG	AH	AH	?

[a] Reproduced from Ref. 6; originally adapted from a table in Ref. 5. The original table was updated with current plans for production plants in Canada, India, and Argentina. Data from Refs. 7-12.

[b] Initially WE; later GC and WD added.

[c] Original capacity about 500 MT/yr; two-thirds shut down and recently dismantled.

ABBREVIATIONS

SC	Self-contained
PA	Parasitic
RW	River water
HY	Hydrogen
SG	Ammonia synthesis gas
GS	Girdler sulfide
AH	Ammonia-hydrogen exchange
WE	Water electrolysis
WD	Water distillation
AD	Ammonia distillation
HD	Hydrogen distillation
GC	Gas-phase heterogeneous catalyst
DT	Dual temperature enrichment
MO	Monothermal enrichment
AECL	Atomic Energy of Canada Ltd.
CCM	Companie de Construction Mecanique Sulzer
CEA	Commisariat a l'Energie Atomique
CGE	Canadian General Electric Ltd.
DAE	Department of Atomic Energy
GELPRA	Groupement Eau Lourde Procede Ammoniac
SAL	Societe a l'Air Liquide
SCC	Societe Chimique de Charbonnages

9.2 PRODUCTION-PROCESS CHARACTERISTICS

Production-process characteristics are divided into four categories that may be used to classify various heavy-water-production processes (Ref. 5).

9.2.1 BOUNDARY CONDITIONS

Boundary conditions restrict the extent to which heavy-water production can occur and are regarded as either self-contained or parasitic. Self-contained plants use only natural sources of heavy water (like river water) and local utilities. Parasitic processes are interconnected with other industrial processes (like hydrogen production) and are combined with dual or multipurpose plants.

9.2.2 DEUTERIUM SOURCES

Sources of deuterium consist of either fresh river water, hydrocarbons, or hydrogen produced in parasitic plants. A number of plants in India plan to extract by a synthesis-gas process whereby large hydrogen streams are produced in commercial ammonia plants.

9.2.3 DEUTERIUM EXTRACTION

Deuterium-extraction processes are described in depth in Nuclear Engineering International articles by Lumb (Ref. 3) and Silberring (Ref. 5). The most widely used is the Girdler sulfide process, which is based on a chemical-exchange reaction between liquid water and gaseous hydrogen sulfide. The ammonia-hydrogen process consists of a chemical exchange of hydrogen isotopes with ammonia. In water electrolysis, deuterium is extracted by isotopic exchange between liquid water and hydrogen gas. This process also finds application in the enrichment of deuterium. Another means of extraction that is also useful for enrichment is hydrogen distillation, in which hydrogen streams are produced in a parasitic plant by burning oil, coal, or gas. Similarly, water distillation has also found use as a means of both enrichment and extraction. Finally, an early Canadian plant used the gas-phase heterogeneous catalyst process, whereby deuterium was extracted by isotopic exchange between steam (rather than liquid water) and gaseous hydrogen. The reaction was aided by using a nickel catalyst.

9.2.4 DEUTERIUM ENRICHMENT

In the initial stages, deuterium-enrichment methods use mostly chemical exchange reactions. They include the extraction processes previously mentioned. In addition, three other processes can be used for enrichment at various stages: ammonia distillation, dual temperature exchange, and monothermal enrichment. Ammonia distillation has been used in place of water distillation in the final stages of enrichment, but it offers no particular advantages.

Dual temperature exchanges (Ref. 5) make use of the temperature dependence of the equilibrium constant of chemical exchange reactions. These processes are driven by heat supply and withdrawal at different temperatures. The exchanging substances (one of which is usually a liquid and the other a gas) circulate in opposing directions across a cold and a hot tower of the plant. Reverse deuterium flow between both substances is achieved in both towers. As a result, deuterium is enriched at one and depleted at another interconnection between the towers.

In the monothermal enrichment system (Ref. 5), a liquid flows through a nearly isothermal cold plant tower in a direction opposing the flow of a gaseous substance. The chemical nature of both substances is exchanged at both ends of the plant tower.

It has been estimated that more than 90% of the heavy water produced worldwide thus far originated from self-contained plants using water as a deuterium source, the Girdler sulfide and dual temperature processes for initial enrichment, and water distillation for final enrichment (Ref. 5). From the summary of Table 9-1, the dominant position of the Girdler sulfide process is evident. As it is, this process is the only one with significant operating experience.

9.3 PRODUCTION CAPACITY

There is only one plant, the U.S. Department of Energy Savannah River Plant, that produces heavy water in the United States. The current capacity of this plant is about 175 MT/yr. However, about two-thirds of the production capacity has been shut down over the years because of limited domestic demand. The U.S. stockpile of heavy water at the beginning of 1977 was about 380 MT (Ref. 1).

The current capacity of the Savannah River Plant and its stockpile appears to be able to provide enough heavy water for a first and second CANDU-type reactor, if constructed in the United States, for operation after the mid-1990s. This conclusion, however, assumes that the plant would be operated at full capacity to produce heavy water. For the third CANDU-type HWR and beyond, an additional heavy-water-production plant would be needed in the United States. For such a case, it can be reasonably assumed that additional capacity would be installed in conjunction with construction schedules of heavy-water reactors (because neither heavy-water reactors nor heavy-water-production plants can be planned independently).

Compared with the limited demand and supply of heavy water in the United States, the Canadian heavy-water industry has the potential to reach a production capacity of over 4,000 MT/yr by 1980 (Ref. 2).

9.4 SAFETY CONSIDERATIONS

Potential accidents in a heavy-water-production plant would be similar to those encountered in many large-scale industrial chemical plants. Because of the potential of explosions, special precautions must be taken for processes involving gaseous hydrogen. In the Girdler sulfide process the problem of dealing with gaseous hydrogen is eliminated, except possibly in enrichment stages if electrolysis is used. As can be seen from Table 9-1, the most recent Canadian plants have avoided having to deal with gaseous hydrogen at any stage.

Use of the Girdler sulfide process requires special precautions in dealing with the highly toxic hydrogen sulfide gas. The operating experience with heavy-water plants at Savannah River and in Canada gives confidence that such plants can be designed and operated with reasonable assurance of adequate safety. The high cost of production results in stringent measures in design and operation to insure safety and prevent process interruption and loss of product.

9.5 ENVIRONMENTAL CONSIDERATIONS

The environmental impacts associated with heavy-water-production plants are expected to be relatively small and would not be a major factor in evaluating the introduction of this fuel cycle into the United States. Relatively few such plants will be required before the year 2000 under any realistic schedule for the introduction of heavy-water reactors.

9.6 LICENSING STATUS AND CONSIDERATIONS

Currently heavy-water-production plants are not subject to NRC licensing regulations.

9.7 DECOMMISSIONING AND DECONTAMINATION

Not applicable.

9.8 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

Not applicable.

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2. Electric Power Research Institute, Study of the Developmental Status and Operational Features of Heavy Water Reactors, EPRI NP-365, February 1977.

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Chapter 10

INTERNATIONAL FUEL SERVICE CENTERS

10.1 BACKGROUND AND GROUNDRULES

Nuclear energy centers have been evaluated in detail for over 10 years (Refs. 1-4). Initially proposed in 1965 for regional industrial development, these centers were visualized as complexes containing many large nuclear plants (Ref. 1); since then, the international fuel service center (IFSC) has been formulated to consist of safeguarded fuel-cycle service plants possibly located together in a power-generating nuclear energy center. In recent work of international significance, the U.S. Nuclear Regulatory Commission (NRC) study (Ref. 5) on nuclear energy and fuel-cycle facility centers in 1975 and the International Atomic Energy Agency (IAEA) study (Ref. 6) on safeguarded fuel-cycle centers in 1977 examined in depth the technical and institutional issues of IFSCs. The NRC study concluded that "it can be feasible and practical, depending on location, to construct up to about 20 nuclear power reactors on a single site" (Vol. I, p. 3-1), with water and land use being the two major limiting considerations, and "nothing was found that would preclude combining power and fuel-cycle Nuclear Energy Centers" (Vol. I, p. 3-33). In another approach, the IAEA evaluated the conditions under which international (nonpower) fuel-cycle centers could function.

Current work under the Nonproliferation Alternative Systems Assessment Program (NASAP) is evaluating both power-generating and non-power-generating safeguarded IFSCs for various uranium/thorium fuel-cycle scenarios requiring reprocessing and refabrication inside the IFSCs and strongly emphasizing the use of plutonium as fuel for in-center reactors only (Refs. 7 and 8). The NASAP evaluation of IFSCs, and the discussion in this section, draw heavily on the assumptions, evaluations, and findings of the NRC site-survey study. In particular, the IFSC is conceived as containing only one 1,500-MTHM/yr multipurpose (Purex, Thorex) reprocessing plant[a] and associated fuel-cycle plants to service a corresponding number of power reactors; there are (if any) at most 20 large power reactors (U.S. vendor design, about 1,300 MWe) inside the IFSC, requiring about 1 acre of land per megawatt electrical of installed capacity based on waste-heat-dissipation considerations.

The safeguarded fuel-cycle service module contains the short-term spent-fuel-storage facility, the reprocessing plant, the fuel-refabrication plant, the radioactive-waste-management plant, and associated utilities and storage areas. All spent fuel, from the reactors inside the IFSC and the offsite reactors[b] serviced by the IFSC, is brought for storage inside the IFSC after a 6-month cooling period at the reactor site. Reprocessing and refabrication of recovered fissile material into reactor fuels are

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[a] The concept of the multipurpose reprocessing plant was evaluated and it was found to be technically feasible (Ref. 8). The plant would be customized to process fuel from specific mixes of reactor/fuel cycles and would use parallel and redundant lines, as necessary. Parallel lines would be required both at the head-end and at the product-end to accommodate different fuel assembly configurations, fuel types, and fissile concentrations, resulting in increased capital and operating costs; the solvent extraction stage is most adaptable to different materials and additional built-in flexibility would add comparatively little to the cost.

[b] Offsite reactors should include reactors owned by foreign nations.

allowed only inside the IFSC. The fuel-refabrication plant consists generally of 200-MTHM/yr lines, and they may be semiremote or of the canyon type, depending on the type of fuel handled. Refabricated fuel is shipped to offcenter reactors in non-weapons-usable form only, whereas conventional plutonium recycle is unrestricted inside the IFSC. Vitrified high-level liquid waste (HLLW) and other treated and reduced wastes can be stored inside the center until they can be shipped away from the center for permanent disposal later. Other considerations in the characterization of the IFSCs are as follows:

1. Specific national or international site layouts are not considered.
2. Fissile material inventories are minimized during and after deployment of the IFSC.
3. The plants operate continuously after startup.

The salient features of 12 IFSC cases considered under NASAP (Ref. 8) are summarized in Table 10-1. The discussion that follows covers, among other points, the IFSCs and the significant environmental, licensing, and development issues that they present. The large concentration of power-generating reactors at one site is the major technical consideration because of the need to dissipate large quantities of waste heat in one region and transmit large quantities of power from one site. A major institutional factor is the international character of the center and the responsibility for safeguards. The various fuel-cycle plants, reactors, and components making up the IFSCs are discussed individually elsewhere in this document.

Table 10-1. NASAP international fuel service center (IFSC) cases

Case	In-IFSC reactors		IFSC development time (yr)	Comments	Off-IFSC reactors		Comments	Total number of reactors serviced
	Type	GWe			Type	GWe		
A.1	LWR (Pu-Th)	10	22	Gradual reprocessing and Pu recycle	LWR (LEU) LWR (DNF-Th)	41 9	Emphasis on LWR (LEU)	60
A.2.a	LMFBR (Pu-Th)	9	22	Full reprocessing at first year for early LMFBR deployment	LWR (LEU) LWR (DNF-Th)	26 23	LWR maturity (119 R-Y spent fuel at first year)	58
A.2.b	LMFBR (Pu-Th)	9	33	Full reprocessing at first year for early LMFBR deployment	LWR (LEU) SSCR (DNF-Th)	17 36	LWR maturity (258 R-Y spent fuel at first year)	62
A.2.c.1	LMFBR (Pu-Th)	6	13	Full reprocessing at first year for early LMFBR deployment	LWR (LEU) HWR (DNF-Th) LWR (DNF-U)	14 15 6	LWR maturity (162 R-Y spent fuel at first year) Early emphasis on HWR	41
A.2.c.2	LMFBR (Pu-Th) (Pu-U Core)	8	13	Full reprocessing at first year for early LMFBR deployment (convert to Pu-U Core in 13 years)	HWR (DNF-Th)	23	382 R-Y (LWR) spent fuel at first year	31
A.2.d	LMFBR (Pu-Th)	21	24	Significant reprocessing early for early LMFBR deployment	LMFBR (DNF-Th) LWR (DNF-U)	32 16	597 R-Y (LWR) spent fuel at first year; early LMFBR deployment	69
A.2.e	LMFBR (Pu-Th)	10	21	Short delay in full reprocessing and LMFBR deployment	LWR (LEU) HTGR (DNF-Th)	29 29	Early LWR deployment; short delay in HTGR deployment	68
A.3	LMFBR (Pu-U)	25	33	Full reprocessing at first year for early LMFBR deployment	LWR (MOX-S)	25	81 R-Y (LWR) spent fuel at first year; delay in LWR Pu recycle (spike)	50
B.1	LWR (Pu-U)	19	19	Gradual reprocessing and Pu recycle	LWR (LEU)	39		58
B.2	LWR (Pu-U)	12	13	Gradual reprocessing	LWR (DNF-Th; U-233 recycle)	54	LWRs converted to U-233 burners on large scale early	66
B.3	HTGR (Pu-Th)	14	16	Gradual reprocessing, gradual HTGR deployment	LWR (LEU) LWR (DNF-U)	53 3	Gradual LWR (LEU) deployment	70
C.1			19	Gradual reprocessing	LWR (LEU) LWR (MOX-S)	39 19	Gradual LWR (LEU) deployment Gradual Pu recycle (spike)	58

10.2 POWER-GENERATING IFSCs

10.2.1 DESCRIPTION

The IFSC contains one centrally located and safeguarded fuel-cycle service module surrounded by reactor modules consisting of one-, two-, or three-unit plants, all including reactor exclusion areas (10 CFR 100) inside a safeguarded perimeter. Specific guidelines that characterize power IFSCs are as follows:

1. In-center reactor startup at the rate of one per year
2. A 75% capacity factor for all reactors
3. Fuel fabrication for the captive reactor systems only except for light-water reactors (LWRs) on the low-enrichment uranium (LEU) cycle outside the center
4. LWR (LEU) deployment and operation schedules as required to meet fissile material needs
5. Large-scale plutonium utilization inside IFSCs

Two types of power IFSCs are considered. The first (type A) uses plutonium-burning converter reactors, mainly breeders, for the purpose of producing weapons-non-usable fuel for reactors at dispersed sites. The second (type B) contains thermal nonbreeder reactors only inside the center and allows no plutonium to be shipped outside the center.

10.2.1.1 Type A IFSCs

Generally in this concept, the IFSC is utilized to produce "denatured" uranium and "adulterated" plutonium fuel for off-center reactors. The recovered plutonium is refabricated into thorium/plutonium fuel for fueling the reactors inside the center in an unrestricted manner; but the recovered uranium-233 is first denatured with uranium (DNF) to about 12% concentration before it is refabricated into thorium-based (DNF-Th) or uranium-based (DNF-U) fuel and shipped outside to various types of reactors at dispersed locations. The reactors inside the center are typically breeders (liquid-metal fast-breeder reactors (LMFBRs)), but LWR centers are also considered; the reactors outside the center are typically thermal systems, but LMFBRs are also considered; a plutonium cycle (plutonium/uranium dioxide) is envisioned for in-center LMFBRs, but plutonium recycle may be used in off-center LWRs under the constraint that the plutonium/uranium dioxide fuel be appropriately spiked with radioactive elements to inhibit diversion for illicit purposes.

The IFSC concept relies heavily on the LWR (low-enrichment uranium) system either long before or during the period of center deployment to provide the large quantities of plutonium for the in-center converters. The reactor types considered here are generally limited to those at the commercial stage (LWRs and heavy-water reactors (HWRs)) and to those at a relatively advanced stage of development (high-temperature gas-cooled reactors (HTGRs), LMFBRs) in the world. Some deployment characteristics (Ref. 8) of IFSCs with type A reactor/fuel-cycle systems are shown in Table 10-1. It is noted that interim LWR spent-fuel storage and full-capacity reprocessing capabilities are assumed for most type A cases at the time the IFSC is instituted, a situation that would be not altogether unrealistic for several industrial countries.

10.2.1.2 Type B IFSCs

In this concept, thermal recycle is practiced; all the plutonium is recovered, refabricated, and used in an unrestricted manner in reactors inside the IFSC, whereas outside reactors are either on the LWR (low-enrichment uranium) cycle or depend on the IFSC for denatured fuel. The reactors included here emphasize the current U.S. commercial system (LWRs) with possibly gradual HTGR deployment, but the fuel cycles are not any less complex than in type A cases, except perhaps in the nonthorium case B.1 of Table 10-1. In contrast to type A IFSCs, type B requires no existing spent-fuel storage and reprocessing capacity at the time the IFSC is instituted.

10.2.2 SAFETY CONSIDERATIONS

For normal operation the limiting dose to be considered, for both reactors and reprocessing plants, is from iodine emission. In IFSCs currently considered (1 acre of land per megawatt electrical of installed capacity, as required for waste-heat dissipation) the offsite radiological impact depends on specific site and climate characteristics, but it is not likely to be a limiting factor because the reactor land areas and exclusion distances are large. At the reprocessing plant a taller stack can be employed to mitigate the potential radiation-dose problem, if necessary. On the other hand, although the incremental exposure to construction workers and operating personnel in IFSCs is somewhat higher than that in existing multiunit plants, the total impact is again too small for this to become a limiting factor.

The accidents that might occur in power plants and reprocessing plants range from essentially no-risk incidents of moderate frequency to accidents with potentially great consequences--including design-basis accidents and severe accidents requiring a number of successive failures--but with very little probability of occurrence either at dispersed sites or at IFSCs. For hypothetical accidents, the potential radiological impact to offsite areas is again not worse with IFSCs because the land areas and exclusion distances are larger than those at dispersed sites. Furthermore, the long-term biological effect to the population at great distances from the reactor would be similar since these effects result primarily from low-level radioactivity spread over a wide area. However, the short-term risk to the local population at IFSCs is somewhat higher for an accident involving a large release of radioactivity. The probability of such an accident, although extremely low, is somewhat higher at IFSCs, where there are many reactors at one site. A severe accident of this type would cause early fatalities at the site and possibly in adjacent reactor areas and offsite communities, depending on site characteristics and climatic conditions.

The hazard from accidents in the transportation of spent fuel and other radioactive material is somewhat lower with IFSCs, although already low, because of the reduced shipping requirements and the shorter distances involved.

Severe natural disasters (e.g., an earthquake of greater magnitude than a safe-shutdown earthquake) could affect an IFSC more than a group of dispersed sites because of the greater aggregation of reactors. However, if an earthquake of this severity should occur in a region, the likelihood that it would affect at least one dispersed reactor site in the region is much higher for dispersed sites than for IFSCs. Damage to IFSC reactors or plants could also arise from terrorist and sabotage activities. In either case, the offsite radiological impact of possible radioactivity releases is less significant for IFSCs because they have more land area and greater exclusion distances.

10.2.3 ENVIRONMENTAL CONSIDERATIONS

10.2.3.1 Waste-Heat Dissipation

Rejection of waste heat generated by the power reactors at the IFSC requires large amounts of water and can potentially cause changes in weather and ecological conditions, thus becoming a major constraint in site selection, IFSC size, and power-plant spacing and operation.

The large flow of cooling water required at a power IFSC (as much as 1.5 million gpm for each 1,200-MWe plant) can be supplied directly from large natural water bodies or by recirculating (evaporative) water systems such as wet towers and cooling ponds with access to major rivers or lakes and reservoirs. Except for exceptionally suitable sites with access to the ocean or the Great Lakes to allow use of the economically preferred once-through cooling system, wet cooling towers are favored for water economy and adaptability. Adequate spacing of such systems must be maintained, however, to mitigate weather effects since their use is expected to have the greatest climatic impact. Typical spacing considered corresponds to approximately 1 MWe/acre. The use of dedicated ponds or lakes and canals on a large scale is generally objectionable because they require too much land; other factors, depending on IFSC site and size, include interaction with groundwater, water consumption, and extent of area fogging.

The large-scale water use at power IFSCs, compared to dispersed power-plant sites, imposes limitations on the withdrawal and discharge systems also because of the concern that the aquatic ecology may be affected locally or as a consequence of far-field water warming. To alleviate the impact on water quality and ecology, engineering solutions are practical in a power IFSC. In contrast to dispersed power sites, the use of a blowdown cooling tower and discharge to a single onsite cooling lake are practical. Diffuser systems of greater length and complexity at discharge systems can also be incorporated.

Other concerns with large IFSCs include the potential for irreversible drawdown effects from dewatering and the long-term groundwater contamination that may result from prolonged construction.

Rejection of heat and moisture to the atmosphere can affect the local and regional climate. The concentrated heat releases from power IFSCs, compared to dispersed small power sites, have a greater potential for increasing convective cloud activity and contributing to storm formation. Local fogging and icing, increase in precipitation in localized areas, downwind rainfall and preferential locations for thunderstorms are possible climatological effects of power IFSCs.

The increased convective activity and local weather effects can be within acceptable limits if heat releases are distributed over a large area (about 2 square miles for one 1,200-MWe LWR), and cooling ponds and lakes are self-limiting because of poor heat-rejection characteristics. However, forced-draft or natural-draft wet cooling towers should be acceptable if widely spaced, allowing 1 acre/MWe. Weather changes at large distances downwind from the IFSC, unlike local weather disruptions, are probably independent of the mode of heat rejection to the atmosphere, whereas their severity depends on the total quantities of heat and moisture rejected and on meteorological conditions. Cooling towers are expected to contribute to increased downwind rainfall in areas with high air stagnation while local fogging is reduced; in unstable atmospheres, the enormous heat and moisture releases may have their greatest impact as "trigger" mechanisms for local precipitation and violent storms.

The potential effect of damage to vegetation and increase in soil salinity from water-droplet dispersion (drift) into the atmosphere from cooling towers is not expected to be significant, except for saltwater systems, which may require special, low-drift cooling towers, depending on site characteristics.

10.2.3.2 Land Use

The IFSC will generally require more land than will a group of dispersed power plants. The actual land area physically disturbed by a safeguarded power IFSC* would be greater, but not very significantly greater, than the area disturbed by existing dispersed sites of equal capacity. In the case of the IFSC, however, all the land is at one location; this suggests that IFSCs should be located in areas of low-value land, generally away from metropolitan areas. On the other hand, the IFSC concept generally requires longer and possibly more power transmission corridors if the IFSC serves geographically dispersed and possibly more distant power-consumption centers. The result is demand for substantially more right-of-way land, amounting to about 37 acres per mile of a 300-foot right-of-way. The use of transmission-corridor land is limited to animal grazing and some farming; the land, however, is not denied to alternative uses as is the land occupied by a power station. Furthermore, the routes of transmission lines are as unobtrusive as possible and not through valued areas (e.g., parks and wildlife preserves).

10.2.3.3 Ecology

The impact of power-generating IFSCs on aquatic and terrestrial ecology is potentially larger than that of dispersed reactor sites because of the greater heat-dissipation and land-use requirements. Most ecological effects are site-sensitive, depending on the water bodies and lands that are used and on the species that are affected as well as their sensitivity. Other effects are caused by the type, level, and duration of construction.

The destruction of aquatic life or habitat alteration and loss caused by water withdrawal, discharge, and warming can be controlled to be not much greater at an IFSC than at dispersed sites, provided that no sensitive species are affected; however, the impact will be more concentrated. The ecological consequences of weather changes caused by heat dissipation are likely to be greater than at dispersed sites, depending both on site characteristics and the nature of these changes. The effects could include reduction in crop yields and disruption of wildlife balances. At an IFSC, the disruption of terrestrial habitats covers a larger area, and since it takes place over a longer construction period, a larger population and possibly more types of wildlife can be affected in comparison with dispersed sites. On the other hand, once construction is completed, the IFSC will provide a large area of undisturbed land to wildlife. Electric-field effects from power lines on bird habitats and migration patterns are also greater because of the concentrated impact in the areas around the IFSC.

Prolonged disruption or permanent loss of habitat and the destruction of aquatic life can result from changes in physical land features, loss of vegetation, soil erosion,

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*The IFSC total land area is greater than 1 acre/MWe because of the centrally located and safeguarded fuel-cycle service module.

adverse impact on water quality from the releases of biocides (e.g., chlorine) during the operation of large IFSCs is assumed to be mitigated by treatment or control of discharges.

sedimentation, and water contamination by dredging operations at the site, at water intakes and discharges, and at power-transmission corridors during several decades of IFSC construction. The ecological impact is both more concentrated than that at dispersed sites and recurring; the exposed ecosystems have therefore less opportunity to recover between disruptions. At each IFSC construction stage new toxic and chemical materials are released or resuspended, resulting in a degradation of water quality in addition to possible new soil-disruption-related effects. The concentrated

10.2.3.4 Costs

It is believed that substantial cost benefits can be realized with large power IFSCs. Cost savings are attributed first to economies of scale, whereby it is practical to build onsite central plants for component fabrication and to employ a stabilized work force (savings of about 12%; Ref. 5, Vol. I, p. 3-3), and second to the potential shortening of power-plant construction times, in comparison with dispersed sites; potentially additional front-end capital costs for land, the construction of onsite fabrication facilities, and transmission lines diminish the construction-cost savings. Unlike construction-cost differentials, incremental transmission costs vary greatly with IFSC location, largely because of the distances to the power-consuming centers, and could cancel most or all of the savings. The extent of the net savings is also subject to the uncertainty of possible future changes in technology, government policy, and power demand.

10.2.3.5 Societal and Socioeconomic Implications

The most important determinant of community impacts for any large power IFSC is the large labor force that is required over a long period of continuous construction. The degree of impact depends largely on the size of the existing population and the infrastructure of the community before construction is begun, with small communities changing the most.

The social impact of an IFSC will be far greater than that of a typical dispersed site, and because an IFSC also requires a large operating work force, the impact will be more permanent. On the other hand, the deployment of a large IFSC will affect a single community infrastructure and not a number of communities, and its impact should be smaller than the collective impact of an equivalent number of dispersed sites with similar community makeup.

The manpower level for the construction of power plants goes through a boom phase during about the first 5 years of construction, after which the labor requirements gradually decrease; but unlike dispersed sites, in IFSCs with many power plants the manpower level goes through a stable second phase. The length of the stable construction phase (and the induced long-term effect on local population and communities) depends on the number of reactors and the reactor-construction schedule at the IFSC. For the 11 power IFSCs considered here, with a construction startup schedule of one reactor per year, the peak work force is in the range of about 7,500 to 10,000 workers (Ref. 8, Section 9.4.6) which constitutes a major construction force. At the peak 2,500 workers are required for constructing the reprocessing plant; nearly half the peak work force is required for constructing the fuel-cycle-service module (including the reprocessing plant). The large IFSCs under consideration have a significant stable construction phase (7 to 10 years) that occurs after the construction of the fuel-cycle module is virtually completed; the corresponding stable reactor-construction work force amounts to approximately 6,000 workers. At a manpower level of 6,000, the population increase would range from some 20,000 people for developed areas not far from metropolitan regions to 60,000 people or more for underdeveloped areas in more rural regions.

The most attractive location for a large IFSC is near metropolitan areas, where a construction force is more readily available without major community disruptions. Such locations are typically objectionable for reasons of adverse public perception, aesthetics, and climatological and radiological impact. Locating a large IFSC in isolated, rural areas, on the other hand, would have a much greater impact initially because it would be required to import a large work force and to establish a new or larger community, which would be disruptive.

Rural areas are generally unable to provide highly developed private and public services (e.g., housing, schools, recreation, transportation, police protection). Therefore, the new demands on the host community or region for services will be substantial and will require front-end capital outlays for public facilities and local private investments. For a large IFSC, both public and private investment would be more attractive since the work-force population would tend to remain as permanent residents in the communities around the site. The permanence of a large work force would also mitigate the potential for economic "boom and bust" situations in communities affected by IFSCs. Thus a large IFSC would be better in this regard than dispersed sites, where work forces and their impacts are transient.

New social values and modernizing forces are expected to contribute to cultural, organizational, and political disruptions in the host community as a result of the massive population inflow needed in a large IFSC at rural and isolated areas. The strain on the community infrastructure would probably be greater if foreigners were brought in through multinational arrangements. However, only one locale will be affected by an IFSC, and the social impact may not be much greater than on the community associated with a single dispersed site. So it must be recognized that socially the IFSC may be a particularly beneficial alternative to an equivalent number of dispersed sites in the same or similar regions. The adverse social effects in a locality can conceivably be mitigated by the expectation for greater employment opportunities and the provision for more and better services to the residents in the vicinity of an IFSC on the premise that the IFSC would promote steady, long-term community growth. Using a large IFSC to trigger and then sustain industrial and economic growth in an underdeveloped rural area would be a consideration in siting, and it requires regional cooperation and planning in a concerted effort with state and Federal government.

A major economic issue in IFSC siting considerations is that of local property taxation and its potential political problems. The IFSCs will generate vast tax revenues, which under conventional property-tax laws in the United States will accrue to the local jurisdiction. The impact will be more severe than with any dispersed site. Lack of tax-revenue redistribution will create gross fiscal inequities, especially for surrounding jurisdictions that are affected by the work-force influx permanently, and it would become a statewide political issue requiring overhaul of state tax codes and new development policies.

The tax issue, the capital-outlay programs for public services, and the application of laws and ordinances will probably be broader and more complex for large power-generating IFSCs with multinational participation and strong government involvement or ownership. In view of the near-monopoly control over most phases of the fuel cycle by national governments, such strong governmental involvement can be reasonably expected.

10.2.4 LICENSING CONSIDERATIONS

10.2.4.1 Siting

The major technical issue in the licensing of large power-generating IFSCs is the concentration of power reactors in a single region and the resultant large demand for cooling water, weather effects from waste-heat rejection through cooling towers, economic and ecological impacts from right-of-way land requirements for transmission lines, and concern for the reliability of the electric power supply. Such concerns are encountered only to a limited extent, if at all, in current multireactor dispersed sites, and the experience of climatic effects from large city centers is only indirectly applicable.

A site of a power-generating IFSC first must be feasible on the basis of a region's characteristics. Important factors to be addressed in this determination include the following:

1. Type of water body available
2. Proximity to developed and/or valued land, such as large cities, parks, and recreation areas
3. Meteorological stability of the atmosphere and weather patterns
4. Types and abundance of wildlife
5. Geographic distribution of power-consuming centers and power-intertie systems
6. Predictability of long-term regional power demand

Arid and remote regions, for example, would be unlikely sites for power-generating IFSCs because of the lack of cooling water and because of costly power transmission, respectively. Furthermore, a region with a poor power intertie system would not be preferred for IFSC siting because of the large dependence by one or more load centers on a single power source.

Acceptance criteria and bases for site selection were developed and generic evaluations of potential IFSC sites ("surrogate sites") were performed as part of the NRC study (Ref. 5, Vol. V) in 1975. One generic criterion for IFSCs with cooling towers, for example, considers to be adequate a power-plant spacing that corresponds to 1 acre of land per megawatt electrical of installed capacity, primarily on heat-dissipation grounds.

Second, the power-generating IFSC must be preferable to an equivalent number of dispersed sites in the same region on public-acceptance grounds as well as from technical considerations since public perception is a crucial element in determining the practicality of a large and expensive project, particularly at the regional level.

Last, the design, layout, and operation of a power-generating IFSC should be optimized on the basis of site-specific characteristics to mitigate the environmental impacts and to increase the safety margins from potential radioactivity releases. Site-specific factors are, for example, local wind patterns for consideration in radiological analyses, river size and water flow for determining the design and spacing of the cooling-water intake and discharge system, and the local hydrology for evaluating the feasibility of using a cooling lake for the discharge of chemically contaminated effluents.

In general, although the climatic effects are poorly understood and the land requirements are high, the deployment of large power IFSCs in the United States should be licensable because acceptable sites are either available or engineering solutions can improve their attractiveness.

10.2.4.2 Safeguards

The most significant nontechnical consideration in the licensing of IFSCs is their international nature and the responsibility for safeguards.

The safeguards system at IFSCs would integrate sensitive (weapons-usable) material accountability and controls with multibarrier physical security to insure that only authorized activities take place in the various facilities; the system can detect, report, and delay unauthorized activities, such as theft of significant quantities of fissile material, until measures can be taken to prevent them (Ref. 8, Vol. I, Section 3.0; bibliography cited).

The design and operation of the power-generating IFSCs under consideration enhances safe-guardability, as the fuel-cycle plants that produce or handle sensitive materials and the power reactors that use them as fuel are located at a single, protected site, thus eliminating the need for transporting sensitive material outside, in contrast to the extensive shipment requirements of dispersed power reactors and fuel-cycle-support plants. The IFSCs clearly alleviate the important regulatory concern over the shipping of large quantities of fissile material.

On the other hand, a power-generating IFSC requires extensive and multiple physical-protection barriers, around facilities, plants, modules, and the whole IFSC, as well as multiple and diverse material-accountability installations throughout vital areas inside the site; the latter must also be integrated with a centralized, massive control system for the safeguards to be effective against potential threats from both outside and inside. This implies two primary concerns, one for the assignment of safeguards responsibility in a center operated under international sponsorship, and the other for the division of physical-protection responsibility between the host country and the IFSC personnel. Application of full-scope IAEA safeguards to all IFSC activities has been proposed as the most sensible institutional solution, and it is recognized that specific institutional arrangements or international agreements among the participating countries will be required to insure effective safeguards in IFSCs.

10.2.5 DECOMMISSIONING AND DECONTAMINATION

Power-generating IFSCs could have advantages in regard to decommissioning, but they could also present more extensive problems after radioactivity contamination.

The decommissioning of power plants would be less disruptive in a large IFSC because it would involve a small fraction of the power generated, and the required technical support (manpower and machinery) could be continuously maintained at the site. The IFSC with resident technical personnel would hold a similar advantage with regard to the renovation or modification of fuel-cycle plants and power plants.

In an IFSC an accidental release of radioactivity could potentially contaminate many reactor areas. The contamination would generally be limited to the reactors in one module because of the large exclusion distance between modules, but many reactors in other modules could also be affected, depending on the severity of the accident and the prevailing weather conditions. In contrast, only a few reactors would be contaminated as a consequence of a similar accident at a dispersed site. In IFSCs, an accident in the fuel-cycle-facilities module (the reprocessing plant primarily) could also contaminate power-plant areas, in addition to the fuel-cycle plants themselves, although the potential contamination effect here is much smaller than it is for power reactors. Clearly then, clustering many power reactors at a single site could potentially be disadvantageous because large-scale loss of power generation could occur and extensive decontamination could be required.

10.2.6 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

Significant research, development, and demonstration needs arise for power-generating IFSCs because of their large impacts; they concern primarily the need to predict with acceptable certainty the climatic effects of concentrated heat rejection and technological ways to reduce the cost and ecological penalties from the transmission of large amounts of power to distant regions. The research and development in both areas is expected to take place in conjunction with current sites of three or four large power units.

In the area of heat rejection, initial emphasis will be on the collection of data from observations of weather and establishment of wildlife habitats and migration patterns. Specific technical areas for high-priority research and development were indicated in the NRC study (Ref. 5, Vol. III, p. 3-115) and include:

1. Vorticity and turbulence
2. Precipitation enhancement
3. Interaction of vapor plumes with pollutants
4. Optimization of heat-rejection systems
5. Criteria for power-plant layout

In the area of power transmission, the same study (Ref. 5, Vol. III, p. 4-69) indicates that the power-generating IFSC would benefit more than the dispersed sites from major advances in higher capacity (higher voltage) power-transmission technology or improved system reliability, although such advances would not provide significant advantages to IFSCs alone. Higher capacity lines and advanced switching and control would alleviate the excessive right-of-way land requirements and thus make the large IFSC more practical for both compact regions and remote sites. Improved system reliability "could make remote siting more attractive, which would favor energy center siting" (Ref. 5, vol. I, p. 4-16).

10.3 NON-POWER-GENERATING IFSCs (TYPE C)

10.3.1 DESCRIPTION

The type C IFSC contains only the safeguarded fuel-cycle service module described above and no power reactors. All power reactors, LWRs on the uranium cycle, are located outside the center, possibly at dispersed sites. The only weapons material available is plutonium, and thermal recycle is practiced under the constraint that the plutonium/uranium dioxide fuel be appropriately spiked with radioactive elements to inhibit diversion. The startup schedule of the plutonium-burning reactors would be one reactor per year with parallel LWR (low-enrichment uranium) deployment until an equilibrium situation is reached, as shown in Table 10-1.

A type C IFSC would be much like the 1,500-MTHM/yr reprocessing plant that Allied-General Nuclear Services built at Barnwell, South Carolina, except that the plant complex area would be larger, with a larger exclusion perimeter to accommodate the other fuel-cycle service plants and the safeguards requirements. The co-location of fuel-cycle plants should not entail significant safety considerations and environmental impacts compared to the reprocessing plant by itself in a dispersed site of similar size; it is believed, on the contrary, that some benefits would accrue from the IFSC concept. Last, compared to power-producing IFSCs, the impacts here are much lower and more predictable since they are not related to either heat dissipation or power transmission. Non-power-generating IFSCs of similar and much larger sizes were examined extensively in the 1977 study by IAEA (Ref. 6).

10.3.2 SAFETY CONSIDERATIONS

The potential radiological impact from the reprocessing plant would be the limiting factor in non-power-generating IFSCs, and it is assumed that the site is suitably large and/or the reprocessing plant stack is appropriately tall so that the long-term risk to the population at great distances would be no worse than with dispersed siting of fuel-cycle plants or power-generating IFSCs for both routine and accidental radioactivity releases. The onsite effect of radioactivity release in a severe design-basis accident (e.g., a criticality accident in a high-level-waste concentration) or in a natural disaster is somewhat worse than that at dispersed sites because personnel in other facilities could be exposed along with those at the reprocessing plant. However, the impact would be less than with power-generating IFSCs. Last, with IFSCs there is the potential for fewer transportation accidents because conceivably there would be fewer spent-fuel shipments, they would be over shorter distances, and there would be no need for the transport of special nuclear material outside the center. Plutonium fuel shipments to off-center reactors are envisioned, but the associated risk from transportation accidents is similar to that with power-generating IFSCs or dispersed facilities.

10.3.3 ENVIRONMENTAL CONSIDERATIONS

Non-power-generating IFSCs present virtually no heat-dissipation problems; the IFSC site is large, and the dispersion of the small quantity of waste heat involved is as inconsequential as with dispersed facility siting. The land required by co-located fuel-cycle plants, on the other hand, could be less than that needed by equivalent facilities at dispersed sites because in the IFSC concept all plants would share a common exclusion area. Thus, less land would be affected ecologically with the IFSC, and, in addition, the effect would be on a single region instead of several dispersed regions.

It is clear that the siting of non-power-generating IFSCs is not significantly constrained, in contrast to the situation with power-generating IFSCs, and it should also present fewer difficulties than does dispersed siting.

Construction-cost savings attributed to construction-force stability and labor-saving machinery in co-located plant sites should be, if realizable, similar to those for power-reactor sites because in both cases the manpower level (about 4,000 workers at peak) and the duration of the construction (7 to 10 years) are similar (Ref. 8, Vol. II, Section 9.4). While the permanence aspect of the construction period is imperceptible with co-located facility centers, the fact that the operating work force is approximately the size of the construction work force implies a permanent socioeconomic impact, justifying more efficient community planning and permanent services to a greater extent than with dispersed plants or power-reactor sites in particular. Therefore, considering that the impacts are also manageable, non-power-generating IFSCs could be used as "growth poles" and located in more remote and rural areas with greater ease than would be the case with power-generating IFSCs. However, the local property-taxation problem would still have potential political implications in the United States, as with power-generating IFSCs.

10.3.4 LICENSING CONSIDERATIONS

The safety and environmental implications of non-power-generating IFSCs are not unlike those of dispersed facilities and should not raise technical licensability issues; it is assumed, however, that a suitably large site would be chosen to accommodate security measures, thus alleviating also any radiological concerns. The safeguards system at non-power-generating IFSCs would not be as massive and complex as in power-generating centers, but the multibarrier physical security and material-accountability and control installations would be extensive and more so compared with currently existing facilities anywhere in the world. Thus, safeguards would present licensing concerns similar to those in power-generating IFSCs at the worst.

10.3.5 DECOMMISSIONING AND DECONTAMINATION

The decommissioning of co-located fuel-cycle facilities would be similar to that for dispersed facilities but less disruptive socioeconomically if it is done gradually, or if it involves the renovation or modification of selected facilities.

As with power-generating IFSCs, facility co-location causes concern that more than one fuel-service area would be contaminated after a significant release of radioactivity, thus requiring facility shutdown and cleanup on a larger scale and over a longer period of time than in the case of dispersed facilities. This would cause fuel-supply disruptions, but operation of the associated reactors could be affected minimally provided that fuel is available from other sources for a while. In power-generating IFSCs, in contrast, the effect could be severe, as power reactors at the site could be shut down for decontamination.

10.3.6 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

Non-power-generating IFSCs do not require significant research and development for deployment, compared with large power-generating centers where environmental uncertainties exist. Fuel-cycle plant co-location is attractive for logistical reasons (e.g., better security) and is already practiced on a small scale either by choice or necessity. For example, the Windscale works (England) and the complex at La Hague

(France) could be viewed as prototype non-power-generating IFSCs; thus, the deployment of co-located fuel-cycle service plants could possibly take place as a result of current practices in large countries.

REFERENCES FOR CHAPTER 10

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2. U.S. Atomic Energy Commission, Evaluation of Nuclear Energy Centers, WASH-1288, January 1974.
3. Battelle Northwest Laboratories, The Hanford Nuclear Energy Center--An Interim Conceptual Study, BNWL-1925, June 1975.
4. General Electric Company, Assessment of Energy vs. Dispersed Electric Power Generating Facilities, May 1975.
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6. International Atomic Energy Agency, Regional Nuclear Fuel Cycle Centers, Vols. I and II, 1977.
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8. International Fuel Service Center Study, draft, August 1978. Prepared by Burns and Roe Industrial Services Corporation for DOE.
9. Interim Report for Evaluations of Reprocessing Configurations for a Nuclear Energy Center, XN-FR-DE, 1, 2, & 3, October 1978.
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APPENDIX A

U.S. Nuclear Regulatory Commission Review of Safeguards Systems for the Nonproliferation Alternative Systems Assessment Program Alternative Fuel-Cycle Materials

BACKGROUND

The procedures and criteria for the issuance of domestic licenses for possession, use, transport, import, and export of special nuclear material are defined in 10 CFR 70, which also includes requirements for nuclear material control and accounting. Requirements for the physical protection of plants and special nuclear materials are described in 10 CFR 73, including protection at domestic fixed sites and in transit against attack, acts of sabotage, and theft. The U.S. Nuclear Regulatory Commission (NRC) has considered whether strengthened physical protection may be required as a matter of prudence (Ref. 1). Proposed upgraded regulatory requirements to 10 CFR 73 have been published for comment in the Federal Register (43 FR 35321). A reference system described in the proposed upgraded rules is considered as but one representative approach for meeting upgraded regulatory requirements. Other systems might be designed to meet safeguards performance criteria for a particular site.

NONPROLIFERATION ALTERNATIVE SYSTEMS ASSESSMENT PROGRAM SAFEGUARDS BASIS

The desired basis for the NRC review of safeguards systems for the Nonproliferation Alternative Systems Assessment Program (NASAP) alternative fuel-cycle materials containing significant quantities of strategic special nuclear material (SSNM),[a] greater than 5 formula kilograms,[b] during domestic use, transport, import, and export to the port of entry of a foreign country is the reference system described in the current regulations and the proposed revisions cited above. The final version of the proposed physical protection upgrade rule for Category I[c] material is scheduled for Commission review and consideration in mid-April. This proposed rule is close to being published in effective form and, together with existing regulations, will provide a sound basis for identification of possible licensing issues associated with NASAP alternative fuel cycles. This regulatory base should be applied to evaluate the relative effectiveness of a spectrum of safeguards approaches (added physical protection, improved material control and accounting, etc.) to enhance safeguards for fuel material types ranging from unadulterated to those to which radioactivity has been added.

To maintain safeguards protection beyond the port of entry into a country whose safeguards system is not subject to U.S. authority, and where diversion by national or subnational forces may occur, proposals have been made to increase radioactivity of strategic special nuclear materials (SSNMs) that are employed in NASAP alternative fuel cycles. Sufficient radioactivity would be added to the fresh-fuel material to require that, during the period after export from the United States and loading into the foreign reactor, remote reprocessing through the decontamination step would be necessary to recover low-radioactivity SSNM from diverted fuel. It is believed that with sufficient radioactivity to require remote reprocessing, the difficulty and time required in obtaining material for weapons purposes by a foreign country would be essentially the same as for spent fuel. In addition, the institutional requirements imposed by the Nuclear Non-Proliferation Act of 1978 include application of International Atomic Energy Authority (IAEA) material accountability

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[a] ≥20% U-235 in uranium, ≥12% U-233 in uranium, or plutonium.

[b] Formula grams = (grams contained U-235) + 2.5 (grams U-233 + grams plutonium); Ref. 10 CFR 73.30.

[c] IAEA definitions of highly enriched uranium (>20%).

requirements to nuclear-related exports. A proposed additional institutional requirement would be that verification of fuel loading into a reactor would be necessary by the IAEA prior to approval of a subsequent fuel export containing SSNM.

Another proposed alternative that could be used to provide additional safeguards protection against diversion of shipments of SSNM by subnational groups would be to mechanically attach and lock in place a highly radioactive sleeve over the SSNM container or fuel assembly.

NRC REVIEW

It is requested that NRC perform an evaluation of a spectrum of safeguards measures and deterrents that could be utilized to protect the candidate alternative fuel cycles. For the fuel cycles under review, consideration should be given to both unadulterated fuel materials and those to which added radioactive material purposely has been added. The relative effectiveness of various safeguards approaches (such as upgraded physical protection, improved material control and accountancy, dilution of SSNM, decreased transportation requirements, few sites handling SSNM, and increased material-handling requirements as applied to each fuel material type) should be assessed. The evaluation should consider, but not be limited to, such issues as the degree to which added radioactive contaminants provide protection against theft for bomb-making purposes; the relative impacts on domestic and on international safeguards; the impact of radioactive contaminants on detection for material control and accountability, measurement, and accuracy; the availability and process requirements of such contaminants; the vulnerability of radioactive sleeves to tampering or breaching; the increased public exposure to health and safety risk from acts of sabotage; and the increased radiation exposure to plant and transport personnel. Finally, in conducting these assessments, the NRC must consider the export and import of SSNM as well as its domestic use.

As part of this evaluation, we request that the NRC assess the differences in the licensing requirements for the domestic facilities, transportation systems to the port of entry of the importer, and other export regulations for those unadulterated and adulterated fuel-cycle materials having associated radioactivity as compared to SSNM that does not have added radioactivity. The potential impacts of added radioactivity on U.S. domestic safeguards, and on the international and national safeguards systems of typical importers for protecting exported sensitive fuel cycle materials from diversion should be specifically addressed. Aspects which could adversely affect safeguards, such as more limited access for inspection and degraded material accountability, as well as the potential advantages in detection or deterrence should be described in detail. The potential role, if any, that added radioactivity could or should play should be clearly identified, particularly with regard to its cost effectiveness in comparison with other available techniques, and with consideration of the view that the radioactivity in spent fuel is an important barrier to its acquisition by foreign countries for weapons purposes. Licensability issues that must be addressed by research, development, and demonstration programs also should be identified.

Table A-1 presents a listing of unadulterated fuel materials and a candidate set of associated radiation levels for each that should be evaluated in terms of domestic use, import, and export:

Table A-1. Minimum radiation levels for various fuel material types

Fuel Material Type	Minimum radiation level during 2-year period, rem/hr at 1 meter (Ref. 6)	
	Mixed[a]	Mechanically attached[b]
PuO_2, $HEUO_2$ powder or pellets[c]	1,000/kgHM	10,000/kgHM
PuO_2-UO_2 and $HEUO_2$-ThO_2 powder or pellets[c]	100/kgHM	10,000/kgHM
LWR, LWBR, or HTGR recycle fuel assembly (including type b fuels)	10/assembly	1,000/assembly
LMFBR or GCFR fuel assembly (including type b fuels)	10/assembly	1,000/assembly

[a] Radioactivity intimately mixed in the fuel powder or in each fuel pellet.

[b] Mechanically attached sleeve containing Co-60 is fitted over the material container or fuel element and locked in place (hardened steel collar and several locks).

[c] HEU is defined as containing 20% or more U-235 in uranium, 12% or more of U-233 in uranium, or mixtures of U-235 and U-233 in uranium of equivalent concentrations.

The methods selected for incorporating necessary radioactivity into the fuel material will depend on the radioactivity level and duration, as well as other factors such as cost. Candidate methods and radiation levels are indicated in the following table and references.

Table A-2. Candidate methods and radiation levels for spiking fuel materials

Fuel material type	Minimum 2-year radiation level, (rem/hr at 1 m)	Process	Minimum initial radiation level, (rem/hr at 1 m)	References
PuO_2, $HEUO_2$ powder or pellets	1,000/kgHM	Co-60 addition	1,300/kgHM	2, 3, 5, 6
PuO_2-UO_2 and $HEUO_2$-ThO_2 powder or pellets	100/kgHM	Co-60 addition	130/kgHM	2, 3, 5, 6
		Fission product addition (Ru-106)	400/kgHM	2, 3, 5, 6
LWR, LWBR, or HTGR recycle fuel assembly	10/assembly	Co-60 addition	13/assembly	2, 3, 5, 6
		Fission-product addition (Ru-106)	40/assembly	2, 3, 5, 6
		Pre-irradiation (40 MWd/MT)	1,000 (30 day)/ assembly	4
LMFBR or GCFR fuel assembly	10/assembly	Co-60 addition	13/assembly	2, 3, 5, 6
		Fission-product addition (Ru-106)	40/assembly	2, 3, 5, 6
		Pre-irradiation (40 MWd/MT)	1,000 (30 day)/ assembly	4

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6. E. A. Straker, Material Radiation Criteria and Nonproliferation, SAI-01379-50765, Science Application, Inc., January 8, 1979.

APPENDIX B

Responses to Comments by the U.S. Nuclear Regulatory Commission PSEID, Volume VII, Fuel-Cycle Facilities

Preface

This appendix contains comments and responses resulting from the U.S. Nuclear Regulatory Commission (NRC) review of the preliminary safety and environmental submittal of August 1978. It should be noted that the NRC comments are the result of reviews by individual staff members and do not necessarily reflect the position of the Commission as a whole.

RESPONSES TO GENERAL COMMENTS

1. Regarding the NRC request to reduce the number of reactor concepts and fuel-cycle variations, the Nonproliferation Alternative Systems Assessment Program (NASAP) set out to look at a wide variety of reactor concepts and fuel cycles with potential nonproliferation advantages. These various concepts have differing performance characteristics in other important respects, such as economics, resource efficiency, commercial potential, and safety and environmental features. The relative importance of these other characteristics and trade-offs has been determined and the findings are incorporated in the NASAP final report.

2. Regarding the comment on the need to address safeguards concepts and issues, some concepts for providing protection by increasing the level of radioactivity for weapons-usable materials have been described in Appendix A to each preliminary safety and environmental information document (PSEID). Appendix A has been revised to reflect NRC comments.

   An overall assessment of nonproliferation issues and alternatives for increasing proliferation resistance is provided in Volume II of the NASAP final report and reference classified contractor reports.

NRC Comment on Mining and Milling

Section 1.1, Uranium, contains data on radon releases from uranium mining taken from a U.S. Department of Energy (DOE) document dated 1975. The NRC has presented data on radon releases both during mine operation and after shutdown to its licensing boards in 1978 and is in the process of updating the radon value in 10 CFR 51, Table S-3. We suggest that the latest NRC material be reviewed to determine whether or not these later data should be used in this PSEID.

Response

The NRC has inferred that the radon release data in Section 1.1 were taken from a DOE document dated 1975 and suggests that more recent NRC data are available and should be used. Actually, as noted in Section 1.1.3 of the PSEID (footnote on page 1-5 of Vol. VII), the radon release rates used in the PSEID (16.6 Ci/MT U_3O_8) were based on NRC data presented in 1978 (Reference 5 in Chapter 1 of the PSEID).

A more recent study[a] performed for the NRC indicates a slightly higher value for underground mines (20.2 Ci/MT U_3O_8). A similar study[b] on open pit mines arrived at an emission rate of 3.3 Ci/MT U_3O_8. The value used in the PSEID (16.6 Ci/MT U_3O_8) appears to be reasonable and conservative for the mix of underground and surface mines expected over the near term future.

NRC Comment on Fuel Fabrication

Section 4.5, Research, Development, and Demonstration, contains statements that more properly should be contained on Volumes 1-6. Research, development, and demonstration (RD&D) work necessary to demonstrate fuel performance (p. 4-20, paragraphs 4 and 9, and p. 4-21, paragraph 8, for example) have more to do with licensability of the reactor concept than the fuel fabrication step.

Response

The statement paragraph 4, page 4-20 that sphere-pac loading of commercial-length rods remains to be demonstrated is intended to relate to process development rather than fuel performance and licensability. Research and development (R&D) work has shown (demonstrated) that short-length fuel rods can be fabricated but work on full-length rods has not been accomplished on a reasonable basis. An irradiation test program (paragraph 8) is needed to prove the fabrication concept--not the reactor concept. The comment on page 4-21, paragraph 8, also appears to be a misunderstanding since the discussion is on the status of RD&D of recycle U-233 fuels including scale-up, prototype fabrication, and irradiation proof-test and does not relate to the licensability of a reactor concept.

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[a] P. O. Jackson et al., Interim Report - Radon-222 Emissions in Ventilation Air Exhausted from Underground Uranium Mines, NUREG/CR-0627 (PNL-2888), March 1979.

[b] K. K. Nielson et al., Prediction of the Net Radon Emission from a Model Open Pit Uranium Mine, NUREG/CR-0628 (PNL-2889), April 1979.

NRC Comment on Reprocessing

Section 5.7.5, Research, Development, and Demonstration, cites the requirement for radiation experiments. These experiments bear more on the licensability of the reactor concepts than the licensability of the fuel-reprocessing concept, and should therefore be a part of the reactor PSEIDs.

Response

Radiation experiments are believed necessary to prove the feasibility of use of recycled coprocessed thorium/uranium as a potential reactor feed material and to provide material for testing under reprocessing conditions. This is necessary to determine environmental release potential, fission product behavior, fuel dissolution, and other information related to process economics, safety, and technology. These experiments may have a bearing on reactor concept licensability in the future but would be primarily a materials testing program, directly related to the reprocessing and fabrication concepts.

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