Nuclear Proliferation and Civilian Nuclear Power

Report of the Nonproliferation Alternative Systems Assessment Program

Volume VI: Safety and Environmental Considerations For Licensing

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1. INTRODUCTION

1.1 PURPOSE, APPROACH, AND SCOPE OF THE ASSESSMENT

This volume of the Nonproliferation Alternative Systems Assessment Program report addresses safety and environmental considerations in licensing the principal alternative nuclear reactors and fuel cycles in the United States for large-scale commercial nuclear power plants. In addition, this volume examines the safety and environmental considerations for licensing fuel service centers. These centers, which have been proposed for controlling sensitive fuel-cycle facilities and special nuclear materials, would contain a combination of such facilities as reprocessing plants, fabrication plants, and reactors. For this analysis, two fuel service center concepts were selected--one with power-generating capability and one without.

This volume also provides estimates of the time required for development of large-scale commercial reactor systems to reach the construction permit application stage and for fuel-cycle facilities to reach the operating license application stage, which is a measure of the relative technical status of alternative nuclear systems.

1.1.1 Purpose

The purposes of this assessment were as follows:

- o To define key safety and environmental issues applicable to the licensing of alternative nuclear reactors and fuel cycles.
- o To assess these issues from a technical viewpoint for determining whether or not a safety and/or environmental issue could restrict or preclude commercializing an alternative nuclear reactor and/or fuel cycle.
- o To evaluate the relative status of these alternatives and to estimate the time required for resolving technical issues necessary to achieve licensing of large commercial reactors and fuel-cycle facilities.

1.1.2 Approach

The approach used in this assessment comprised the following steps:

1. Selecting the reactors and fuel cycles for assessment.
2. Evaluating past reviews of these alternative reactors and fuel cycles.
3. Exchanging information between the Nuclear Regulatory Commission staff and the Department of Energy staff.
 - a. Nuclear Regulatory Commission staff members were provided with descriptions and analyses of these alternatives.
 - b. The Nuclear Regulatory Commission staff members then provided their views on the issues and problems.
4. Defining the key issues and problems revealed by past reviews and by discussions with the Nuclear Regulatory Commission staff.
5. Identifying research, development, and demonstration needs for resolving these issues.
6. Determining what effect safety or environmental issues have on commercialization of an alternative reactor and/or fuel cycle.
7. Estimating the time required for development to reach the licensing application stage for large commercial units.

1.1.3 Scope

The scope of the assessment was restricted to safety and environmental issues only. The assessment did not include the need for power, economics, antitrust issues, proliferation risks, safeguards, and institutional and social factors, all of which are considered in the licensing process. Proliferation risks and safeguards considerations are reviewed in Volume II of the NASAP Report. The principal reactor and fuel-cycle concepts assessed are presented in Table 1. Detailed descriptions of these reactors and fuel cycles can be found in Volume IX. These nuclear power systems were chosen for assessment because they are representative of the principal alternatives in operation or under development in the world. Other less-developed reactor and fuel-cycle concepts, which are not as well characterized, are not included in this assessment but are described in Volume VIII.

Table 1. Principal Reactor and Fuel Cycle Concepts

Reactor	Fuel Cycle	
	Once-Through	Recycle
1. Light-Water Reactor	<ul style="list-style-type: none"> o Low-enriched uranium, 30 MWd/kg burnup o Low-enriched uranium 15% improved uranium utilization 	<ul style="list-style-type: none"> o Uranium/plutonium self-generated recycle o Denatured uranium/thorium, with uranium enriched to 12% U-233
2. Heavy-Water Reactor	<ul style="list-style-type: none"> o Uranium enriched to 1.2% U-235 	
3. Water-Cooled Breeder Reactor and Prebreeder*		
Prebreeder		<ul style="list-style-type: none"> o Uranium/thorium with uranium enriched to 15-20% U-235 o Uranium/thorium with uranium enriched to 93% U-235 o Plutonium/thorium
Light-Water Breeder Reactor		<ul style="list-style-type: none"> o U-233/thorium
Light-Water Breeder Reactor High-Gain Converter		<ul style="list-style-type: none"> o U-233/thorium

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*Several variations of these reactors were reviewed.

Table 1. (Continued)

Reactor	Fuel Cycle	
	Once-Through	Recycle
4. High-Temperature Gas-Cooled Reactor		
High-Temperature Gas-Cooled Reactor, Steam Cycle	o Uranium/thorium with uranium enriched to 20% U-235	o Uranium/thorium, with uranium enriched to 20% U-235 o Uranium/thorium with uranium enriched to 93% U-235; denatured uranium/thorium; uranium/thorium, with uranium enriched to 12% U-233
High-Temperature Gas-Cooled Reactor, Direct Cycle	o Same fuel cycle as the high-temperature gas-cooled reactor, steam cycle	o Same fuel cycles as the high-temperature gas-cooled reactor, steam cycle
5. Gas-Cooled Fast Reactor		o Uranium-plutonium core with thorium blankets
6. Liquid-Metal Fast-Breeder Reactor**		o Uranium-plutonium core with uranium blankets o Uranium-plutonium core with thorium blankets o Plutonium-thorium core with thorium blankets o Denatured U-233/thorium core with thorium blankets

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**Several core design variations (homogeneous, heterogeneous) and fuel type variants (oxide, carbide, metal) were reviewed.

1.2 THE LICENSING PROCESS

The Federal agency responsible for licensing nuclear power plants and fuel cycle facilities is the Nuclear Regulatory Commission. Its functions are to provide the basic rules for the licensing process through the Code of Federal Regulations and the Regulatory Guides, to review license applications for compliance with these regulations, and to issue the licenses for construction and operation of all commercial nuclear facilities. A summary of the principal actions in the Federal licensing of nuclear power plants is presented in Table 2. Before a nuclear power plant or fuel-cycle facility can be built at a particular site, the applicant must obtain a construction permit from the Nuclear Regulatory Commission. As a major part of the application for a construction permit, an applicant must file a Preliminary Safety Analysis Report presenting the design criteria and preliminary design information for the proposed plant and comprehensive data on the proposed site. The report also discusses various hypothetical accident situations and the safety features which will be provided to prevent accidents or, if they should occur, to mitigate their effects on both the public and the facility's employees. In addition, the applicant must submit a comprehensive Environmental Report providing a basis for the evaluation of the environmental impact of the proposed plant.

The application is reviewed by members of the Nuclear Regulatory Commission staff and its consultants over a period of about 2 years to determine that the plant design is consistent with the Nuclear Regulatory Commission requirements. When the staff concludes that acceptable criteria, preliminary design information and financial information are documented adequately in the application, a Safety Evaluation Report is prepared. This report represents a summary of the review and evaluation of the application, relative to the anticipated effect of the proposed facility on the public health and safety.

The Advisory Committee on Reactor Safeguards, an independent statutory committee established to provide advice to the Nuclear Regulatory Commission on reactor safety, reviews each application for a construction permit. Then, the Nuclear Regulatory Commission prepares one or more supplements to the Safety Evaluation Report to address the safety issues raised by the Advisory Committee on Reactor Safeguards in its report and to include any other information made available since issuance of the original Safety Evaluation Report. After receipt of all comments and resolution of any outstanding issues, a Final Environmental Statement is issued and also is made public. The Safety Evaluation Report and its supplements and the Final Environmental Statement constitute the staff's primary evidence at the subsequent public hearings.

Table 2. Principal Steps in the Federal Licensing of Nuclear Power Plants

Phase	Applicant Actions	Nuclear Regulatory Commission Actions	Advisory Committee on Reactor Safeguards Actions
<u>Phase 1</u> Construction Permit Application	<u>Step 1:</u> Documents submitted to the Nuclear Regulatory Commission	<u>Step 1:</u> Reviews and comments on applicant's documents	<u>Step 1:</u> Reviews the Preliminary Safety Analysis Report and the Safety Evaluation Report
	o Preliminary Safety Analysis Report	<u>Step 2:</u> Prepares response documents	<u>Step 2:</u> Issues a report to the Nuclear Regulatory Commission
	o Environmental Report	o Safety Evaluation Report	
	o Financial Information	o Draft and Final Environmental Statement	
	<u>Step 2:</u> Responds to Nuclear Regulatory Commission's comments	<u>Step 3:</u> Conducts public hearings through the Atomic Safety and Licensing Board	
		<u>Step 4:</u> Issues construction permit	
<u>Phase 2</u> Operating License Application	<u>Step 1:</u> Submits Final Safety Analysis Report to the Nuclear Regulatory Commission	<u>Step 1:</u> Reviews and comments on Final Safety Analysis Report	<u>Step 1:</u> Reviews and evaluates the Final Safety Analysis Report
	<u>Step 2:</u> Responds to Nuclear Regulatory Commission's comments	<u>Step 2:</u> Prepares Safety Evaluation Report	<u>Step 2:</u> Advises the Nuclear Regulatory Commission
		<u>Step 3:</u> Holds public hearings (upon petition)	
		<u>Step 4:</u> Issues operating license containing the technical specifications for the plant	

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The law requires that a public hearing be held before a construction permit may be issued for a nuclear power plant. Soon after an application is approved, the Nuclear Regulatory Commission issues a notice of the hearing which will be held after completion of the Nuclear Regulatory Commission's safety and environmental reviews. The public hearing is conducted by a three-member Atomic Safety and Licensing Board appointed from the Nuclear Regulatory Commission's Atomic Safety and Licensing Board Panel. If the initial decision regarding the National Environmental Policy Act and safety matters is favorable, a construction permit is issued to the applicant by the Director of Nuclear Reactor Regulation. The board's initial decision is subject to review by an Atomic Safety and Licensing Appeal Board and the Commission.

When the construction of the nuclear plant has progressed to the point where final design information and plans for operation are ready, the applicant submits the Final Safety Analysis Report in support of an application for an operating license. This document sets forth the pertinent details on the final design of the facility, including final containment design, design of the nuclear core, and waste handling system. It also provides plans for operation and emergency procedures. Again the Commission makes a detailed review of the information. Amendments to the application and reports may be submitted from time to time. The staff again prepares a Safety Evaluation Report and, as during the construction permit stage, the Advisory Committee on Reactor Safeguards makes an independent evaluation and presents its advice to the Commission.

A public hearing is not mandatory for an operating license application. However, soon after acceptance for review of the operating license application, the Commission publishes notice that it is considering issuance of the license. The notice provides that any person whose interest might be affected by the proceeding may petition the Nuclear Regulatory Commission for a hearing. If a public hearing is held, the same decision process described for the construction permit hearing is applicable.

Each license for operation of a nuclear reactor contains Technical Specifications, which set forth the particular safety and environmental protection measures to be imposed upon the plant, and the conditions of its operation that are to be met in order to assure protection of the health and safety of the public and of the surrounding environment.

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1.3 GENERIC SAFETY AND ENVIRONMENTAL ISSUES

Safety issues common to all of the alternative reactors and fuel cycles relate to protecting the nuclear facility operating staff and the general public from radiation exposures exceeding legal standards and guidelines that might result from unplanned events or accidents (equipment failures, operator errors, or external forces such as earthquakes). The primary assurance of safety depends on a high degree of reliability of plant equipment obtained through conservative design, adequate testing, and application of rigorous standards and extensive quality control. Redundant components are provided to increase system reliability. In accordance with the "defense-in-depth" concept, which provides multiple safety protection barriers, additional safety features and engineered safety systems are provided to mitigate the consequences of plant equipment failures and accidents postulated to occur despite precautions. Also, nuclear facilities are sited in areas of low population to keep the number of persons exposed to radiation low, in the event of a radioactivity release.

Environmental issues common to all of the assessed nuclear alternatives include a broad spectrum of physical and social impacts. These include commitment of land and water resources; heat disposal; radiation exposures, both to the operating staff and to the general public; proliferation risks; and social and direct costs.

Underlying the licensing process are public concerns that are common to all nuclear power alternatives. The licensing process formalizes the assessment of nuclear facilities and ensures that these public concerns are adequately addressed. The public concerns that must be resolved for all nuclear power alternatives are discussed below.

- o Possibility of Accidents. How improbable is an accident? What frequency of occurrence and magnitude of accidental radioactivity release are acceptable for each type of accident? Can acceptable emergency plans be put in place to handle such accidents?
- o Disposal of Highly Radioactive Wastes. What should be the criteria and standards for encapsulation and long-term storage of highly radioactive nuclear wastes? Can these standards be achieved?
- o Low-Level Radiation Exposure. What level of general population exposure is acceptable from normal operation of nuclear facilities?

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- o Nuclear Proliferation. What level of safeguards and proliferation countermeasures would be required for sensitive operations such as reprocessing?
- o Trust in Institutions. Is the Nuclear Regulatory Commission prepared to deal with an emergency? Can the Nuclear Regulatory Commission and the Environmental Protection Agency be trusted to be unbiased and competent regulators of the nuclear industry? Can utilities be trusted to operate nuclear reactors?

The granting of a license to construct and operate a nuclear facility requires comprehensive analyses of safety and environmental factors; these analyses are accomplished by the applicant and by regulatory agency reviews. Public concerns are addressed in hearings to allow the public to question the basis for granting the license.

At the present time, although safety standards do exist, standards for acceptable levels of probability and frequency of occurrence for severe accidents have not been firmly established by the Nuclear Regulatory Commission. Maximum radioactivity releases during accidents are specified in Title 10 of the Code of Federal Regulations, Part 100. At the present time, the general approach taken by the Nuclear Regulatory Commission is that the more severe the consequences of an accident, the lower its probability should be, with the most severe accidents expected to have a frequency of occurrence of less than 10⁻⁷ per year of reactor operation. In the licensing process, accidents must be shown, through analysis and verification tests of less severe accidents, to have acceptably low potential frequencies of occurrence. An adequate basis of experience must be obtained through research, development, and demonstration to judge both the probability of equipment failure and thus the frequency of accident occurrence, and the consequences of the occurrence. This basis must be documented in the form of design criteria, standards, analyses, and test reports before commercial use can be licensed.

Disposal of highly radioactive wastes is a concern for all nuclear reactors and fuel cycles because these wastes are generated from the fission products, activation products, and transuranics elements created by nuclear reactions in nuclear power plants. Whether or not the spent-fuel elements from a reactor are reprocessed, these radioactive materials must eventually be safely disposed of for extremely long time periods. A draft regulation (Federal Register 43:53262) providing criteria for waste disposal has been published by the Environmental Protection Agency. Development of standards and guidelines is under way at the Nuclear Regulatory Commission for geologic disposal of high-level wastes from commercial nuclear operations.

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The maximum permissible concentrations of radioactive materials that may be emitted to unrestricted or restricted areas during normal operation of commercial nuclear facilities are specified in the regulation Title 10 of the Code of Federal Regulations, Part 20. The Nuclear Regulatory Commission has implemented a policy providing that emissions should be substantially below these levels and should be "as low as reasonably achievable," which for current plants is about 1 percent of the maximum permissible amount for unrestricted areas, but is close to the maximum permissible for occupational exposure in restricted areas. A key consideration in establishing the maximum permitted concentrations of radioactive materials is the relationship between exposure and adverse health effects. Currently, this relationship is estimated for the low levels of emissions that occur in nuclear reactors and fuel cycles by linear extrapolation of health effects data from higher-level exposures and assumes that no threshold exists below which radiation ceases to have health effects on humans. This linear extrapolation basis was reviewed recently by the National Academy of Sciences, Committee on the Biological Effects of Ionizing Radiation, who judged the linear extrapolation to be the most prudent assumption at the present time, based on current scientific knowledge; although according to a substantial minority of the Committee, this basis may ultimately prove to be overly conservative.

A review of the process and institutions for regulating the nuclear industry is part of the work of the presidentially-appointed Kemeny Commission, formed after the accident at the Three Mile Island plant. The Commission's report was recently issued. Congressional committees are also reexamining the regulatory process in areas such as emergency planning, inspection, and enforcement.

2. FINDINGS

2.1 REACTORS

2.1.1 Light-Water Reactors--Current Designs

The light-water reactor is the only reactor licensed by the Nuclear Regulatory Commission for commercial use in the United States. A total of about 70 reactors (51,500 MW) have been licensed to operate, and 91 reactors (99,700 MW) were licensed for construction as of August 1979. After more than 25 years of evolution, licensing requirements for light-water reactors are considered to be comparatively well-defined in the form of regulatory criteria and guides. However, there are still some unresolved safety issues which were identified before the accident at Three Mile Island and are subjects for ongoing industry and Nuclear Regulatory Commission actions to enhance safety. In addition, the accident at Three Mile Island in March 1979 has led to major reviews of the licensing criteria, design requirements, operating procedures, and operator qualifications by the Nuclear Regulatory Commission, the Kemeny Commission, several congressional committees, and industry. Significant changes have been identified in a preliminary report and statements by the Nuclear Regulatory Commission in their Lessons-Learned study. The most important of these changes may be the modification or extension of the present approach to defining and analyzing design basis events or the establishment of an entirely new approach to defining these events.

Furthermore, requirements for some systems and components which are currently nonsafety-grade may be upgraded. The current approach to operator-system interaction, which ignores the possibility of an operator error, will be reexamined, as will the design requirements for handling post-accident operations. The full impact of this accident on reactor licensing cannot be known until reviews are completed, some in late 1979 and others in 1980.

2.1.2 Alternative Reactors

Fulfilling safety and environmental requirements will be more difficult to demonstrate for some alternative reactor concepts than for others, and the risk of serious delays or setbacks will vary with the extent to which existing technology can be extrapolated. Changes in the General Design Criteria and design requirements for light-water reactors will also affect the alternative

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reactors. However, no safety or environmental issues have been identified for the reactors shown in Table 1 that would prevent eventual licensing.

As a measure of the difficulty each alternative would have in meeting safety and environmental requirements, estimates were made of the minimum time it would take to resolve technical issues which are currently preventing commercial license application. Reactor plant concepts that have a demonstrated safety basis to proceed with a definitive commercial design for wide employment would need a minimum of 2 to 3 years for design completion and Preliminary Safety Analysis Report preparation. Reactor plant concepts that still require a demonstration reactor to be operated so that the safety basis can be established, would require a minimum of 16 to 18 years before a commercial-size plant Preliminary Safety Analysis Report could be submitted. Some fuel concepts would need qualification before loading into commercial reactors; the minimum time for them to reach this stage would be 7 to 12 years.

For each of the alternative reactor and fuel concepts selected for this analysis, several issues remain to be resolved through research, development, and demonstration programs. The most important of these issues are summarized in the following discussions; estimates of the minimum time to resolve these issues and submit a Preliminary Safety Analysis Report are summarized in Table 3.

For light-water reactor fuel which is modified to be 15 percent more efficient in the use of uranium, response to transients will have to be essentially equivalent to that of current design fuel, and the fuel failure rate during normal operation is designed to be within current standards. A Preliminary Safety Analysis Report for commercial loadings could be submitted in 8 years. Fuel designed for less than 15 percent improvements could reach the same stage earlier, as irradiation data became available.

Fuel qualification is the major issue for the light-water breeder-reactor prebreeder and breeder concepts. The response to transients and fuel failure rate during normal operation must be within the range of current light-water reactor fuel performance for both the prebreeder and breeder fuels. In addition, it must be demonstrated that loss-of-coolant accidents can be accommodated safely in the tight-lattice core of the breeder for large-scale commercial plants. A Preliminary Safety Analysis Report for commercial loading of prebreeder fuel could be submitted in 12 years. A Preliminary Safety Analysis Report for commercial loading of breeder fuel could be submitted in 7 years; the time for the breeder is shorter because the breeder fuel is now being demonstrated in the Shippingport Nuclear Power Station.

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However, because the light-water breeder reactor requires a large quantity of U-233 for the initial core, which must be obtained from reprocessing of prebreeder cores, the breeder cannot proceed commercially until after prebreeders have been operated and their fuel reprocessed -- this will be 20 years or more.

Table 3. Preliminary Safety Analysis Report Time Schedule

Concept	Demonstration Needed		Minimum Time to Submit Preliminary Safety Analysis Report for Commercial Employment ^b			
	Plant	Fuel	(years)			
			<5	5-10	10-15	15-20
Light-water reactor 15% improved		X		X		
Light-water breeder reactor prebreeder		X			X	
Light-water breeder reactor breeder		X		X		
Heavy-water reactor			X			
High-temperature gas- cooled reactor steam cycle			X			
High-temperature gas- cooled reactor direct cycle	X					X
Liquid-metal fast breeder reactor (loop-type)	X	X				X
Liquid-metal fast breeder reactor (pool-type)	X	X		X ^a		
Gas-cooled fast reactor	X	X				X

^a Based on Superphenix demonstration of plant and fuel by the end of 1983

^b From end of 1979.

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For the heavy-water reactor concept, standards for licensing pressure tubes as nuclear pressure boundaries, the safety aspects of onstream refueling systems, and standards for tritium containment must be developed. Using the Canadian safety data for the plant and fuel concepts, which is routinely made available as a condition of Canadian Deuterium Uranium reactor commercial licensing agreements, a Preliminary Safety Analysis Report for a definitive commercial design can be submitted in 3 years. The fuels enriched to 1.2 percent U-235, with a burnup of about 20 MWd/kg used in the NASAP conceptual heavy-water design would require further qualification tests which could be done in parallel with the design and licensing of the plant.

Issues yet to be resolved for the steam-cycle high-temperature gas-cooled reactor design include the system response to some accidents, the acceptability of graphite as a core structural material, the response of the reactor core to earthquakes, in-service inspectability requirements for the reactor components, provisions for emergency core cooling, primary system integrity, and low-probability accidents. A Preliminary Safety Analysis Report for a definitive commercial design could be submitted in 2 years.

The direct-cycle high-temperature gas-cooled reactor concept faces the same plant design issues as the steam-cycle concept and, in addition, system responses to transients such as loss-of-load turbine-generator overspeed, turbine/generator shaft seal failure, internal pressure equilibration accidents, and the turbomachinery (turbine-generator-compressor) failure consequences, maintenance capability, and acoustic vibration from turbomachinery. A demonstration plant would be necessary before commercial deployment. Assuming successful demonstration, a Preliminary Safety Analysis Report for a definitive commercial design could be submitted in 16 years.

Unresolved issues for the liquid-metal fast-breeder reactor are the effect of the high power density, energy-releasing accidents following loss of coolant, recriticality after core disruption accident, core meltdown progression after a core disruption accident, and sodium interactions with air or water. Two possible licensing alternatives are offered in terms of commercial deployment. One would be to proceed with a commercial design similar to the pool-type Superphenix design. In this case, assuming that the French safety data base would be available, a Preliminary Safety Analysis Report for a commercial-size liquid-metal fast-breeder reactor in the United States could be submitted in about 6 years (2 years after the Superphenix scheduled starting date of operation). The other alternative would be to proceed with a design significantly different from the French design such as a loop-type design. A demonstration plant would then be required, and on the basis of its successful operation, a Preliminary Safety Analysis Report for a commercial-size plant could be submitted in 16 years. This estimate includes 2 years

(until the end of 1981), when it is assumed that a decision will have been made on the choice and type of breeder reactor plant that is to be pursued in the United States.

Issues yet to be resolved for the gas-cooled fast reactor are reliability of core cooling, definition of and response to depressurization accidents, and recriticality accidents. Unlike the other concepts, a prototypical small-scale reactor has not been operated. From a safety viewpoint, it is possible, however, that such a prototype would not be needed, before a full-size demonstration plant could be licensed because of the similarity of its reactor plant components to those of the high-temperature gas-cooled reactor steam cycle and its fuel to that of the liquid-metal fast breeder reactor. The components of the high-temperature gas-cooled reactor have been adequately demonstrated from a safety viewpoint, and the fast-breeder fuel will undergo development tests in the Fast Flux Test Facility and in transient test reactors. In addition, aspects of the gas-cooled fast reactor which will need prototypical demonstration can be addressed in non-nuclear experiments and test loops, existing fast-breeder reactor facilities, the Experimental Breeder Reactor-2 and the Fast Flux Test Facility. Based on the successful operation of a full-size demonstration plant, a Preliminary Safety Analysis Report for a definitive commercial design could be submitted in 18 years. This estimate includes 2 years (until the end of 1981) when it is assumed that a decision will have been made on the choice and type of breeder reactor that is to be pursued in the United States.

2.2 FUEL CYCLES

2.2.1 Once-Through Light-Water Reactor Fuel Cycle, Interim Spent-Fuel Storage

The only fuel cycle licensed by the Nuclear Regulatory Commission is the once-through fuel cycle using low-enriched uranium oxide pellet fuel in light-water reactors with interim storage of spent fuel above ground. Licensing of uranium ore mills, uranium hexafluoride conversion facilities, fuel fabrication plants, and the fresh fuel transportation system is well established.

Licensing of new interim above-ground spent-fuel storage facilities away from the reactors has not yet been accomplished, although limited storage has been licensed at the shut-down reprocessing facilities at West Valley, New York and Morris, Illinois. A new away-from-reactor facility could be available in about 7 years after a decision to proceed. New spent-fuel storage facilities either at-reactor or away-from-reactor will be needed for some reactors by mid-1980's.

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At least for the near term, the licensing of enrichment facilities is not an issue, since such facilities are Federally-owned and operated. Although a Safety Analysis Report guide has been issued by the Nuclear Regulatory Commission, the licensing requirements for commercial enrichment facilities have not yet been fully defined. There are no commercial license applications active at the present time.

Licensing requirements for shipping spent-fuel shipping casks and transportation are well established. However, proposed changes to the cask shipping regulations, which are intended to make shipments more secure and to reduce the consequences of transportation accidents, will make spent-fuel shipping more difficult and costly.

2.2.2 Alternative Fuel Cycles

No safety or environmental issues have been identified that would prevent eventual licensing of any of the fuel cycles shown in Table 1. It is assumed that these alternative fuel cycles will have to meet safety and environmental criteria essentially the same in basic intent as the criteria for the once-through light-water reactor cycle.

The minimum times from the decision to proceed with an alternative fuel cycle to the commercial operation of large-scale facilities are as follows:

- o From 10 to 14 years would be required for the once-through light-water reactor cycle with a Federally-owned facility for the retrievable geologic disposal of spent fuel.
- o From 10 to 20 years would be required for the recycle of uranium/plutonium fuel in light-water reactors if the Allied General Nuclear Services reprocessing facility at Barnwell, South Carolina, were used; from 15 to 20 years if a new reprocessing facility would have to be built.
- o At least 20 years would be required for the use of uranium/plutonium or uranium/thorium fuels to be used in breeder reactors.

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For each of these alternative fuel cycles several issues remain to be resolved through research, development, and demonstration efforts. The most important of these issues are presented below.

Geologic disposal of spent fuel from light-water reactors on the once-through fuel cycle has not yet been demonstrated. Because of uncertainties in the demonstration, it is likely that retrievability of the spent-fuel disposal packages will be a requirement. A program is underway to develop the technology and evaluate geologic media. The program is planned to lead to operation of a Federally-owned repository licensed by the Nuclear Regulatory Commission beginning between 1990 and 1994. The issue of storage of high-level wastes containing large quantities of very long-lived transuranic elements must be addressed in accordance with the Environmental Protection Agency's proposed criteria for radioactive wastes and the Nuclear Regulatory Commission standards being prepared under Title 10 of the Code of Federal Regulations, Part 60. This is a potentially significant difference between once-through and recycle systems. A technical resolution of this issue appears feasible but there are significant technical and institutional barriers to be overcome.

The uranium/plutonium cycle in light-water reactors has had the process steps developed and demonstrated for many of its parts using pilot-scale facilities. The remaining issues involve worker exposure, criticality safety, containment of radioactivity during normal and abnormal operation and during natural catastrophes, high-level waste disposal, and material safeguards. These concerns apply principally to the reprocessing and refabrication operations. Commercial-scale recycle of recovered plutonium and uranium could begin about 10 years following a decision to recycle if the Allied General Nuclear Services reprocessing facility at Barnwell, South Carolina were completed without major design revision and operation was not contingent on availability of a geologic high-level waste disposal facility. Licensing and construction of a recycle fuel fabrication facility could commence concurrently. If a variation of the basic PUREX process were needed or if spiking were required, an additional 5 or more years may be added to the schedule to conduct the necessary research and development leading to the final design. Demonstrations of high-level waste disposal in geologic repositories will be required. As with the spent-fuel disposal demonstration, retrievability of the high-level waste disposal package will be a requirement.

The issues to be resolved for the uranium/plutonium fuel cycle in breeders are similar to those for the same cycle in light-water reactors. Those issues unique to breeders, such as criticality safety with high plutonium concentration, and higher heat loads and radiation fields from high-burnup fuel, may require that more stringent design requirements be met by reprocessing plants

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and recycle fuel fabrication plants. In addition, reprocessing and refabrication of breeder fuel that has been allowed to cool for only a short period of time remains to be demonstrated. Remote fabrication and special in-process safeguards accounting capability will be required because of the high gamma-ray and neutron radioactivity of these fuels. It is expected that research, development, and demonstration programs on light-water reactor recycle fuels and on the unique aspects of breeder fuels will resolve these issues. Commercialization of recycling light-water reactor fuel would provide additional licensing experience which would reduce the uncertainties of recycle in breeders, but would not significantly shorten the time it would take to achieve licensing of breeder recycle. Twenty years or more would be required from a decision to proceed with this fuel cycle to operation of licensed large-scale commercial breeder recycle fuel plants (including operation of demonstration facilities).

The uranium/thorium fuel cycle alternatives are less developed than the uranium/plutonium fuel cycle and need substantial technology development and demonstration. Licensing of thorium/U-233 fuels would have issues similar to those of uranium/plutonium fuels, but more research and development would be required. Remote fabrication and special in-process safeguards accounting capability must be developed since these recycle fuels contain high gamma-ray activity. Twenty years or more would be required to commence operation of licensed commercial large-size plants from the time that a decision is made to proceed with the uranium/thorium fuel cycle. This time will apply to the high-temperature gas-cooled reactor, the light-water breeder or fast-breeder reactor fuel cycles employing uranium/thorium fuels with recycle of U-233.

2.3 COLOCATED FACILITIES

In this assessment, two types of fuel service centers are considered for the handling and use of sensitive fuel materials to improve safeguards and control. They are:

1. Fuel service centers without power reactors, in which fuel-recycle facilities are colocated.
2. Power-generating fuel service centers, or energy centers, in which power reactors are located with fuel-recycle facilities.

These two options have been compared with equivalent types and numbers of facilities in dispersed locations. This analysis has found that the safety

issues and environmental impact of fuel service centers without power generation are similar to those for dispersed facilities. Total licensing time would not be significantly affected by colocation, although colocated facilities will have some safeguards advantages which may simplify licensing.

Safety issues of a power-generating fuel service center compared to those for dispersed sites depend on the number of reactors located at the center, the site locations, and site characteristics. A general conclusion concerning relative safety cannot be made except on a site-specific basis, but some intrinsic differences can be identified, as follows:

- o An on-site radioactive release might have greater impact on facilities and workers at the center than the same release at a dispersed facility.
- o Safeguarding of sensitive materials in transit at colocated facilities is simpler and subject to greater control than at dispersed facilities.
- o The potential for transportation accidents and injuries is less for colocated facilities because there is a significant reduction in shipping distances for spent fuel, recycle fuel, and high-specific activity waste, and because specially-dedicated transportation facilities would be used to move these materials on site.

Comparing the environmental impacts of power-generating fuel service centers to those of dispersed facilities is highly dependent upon the number of colocated reactors. For a large center (25,000 MW of electrical generating capacity and 1,500 t/yr reprocessing plant), environmental and socioeconomic impacts are large and localized. For example:

- o Land and water commitments would be very large (roughly 69 square miles for the total site and 220,000 gallons per minute of water use for reactor cooling alone).
- o Heat rejection to the atmosphere would be very large (50 GW of heat) and could cause local weather modifications that would adversely affect the local population and ecology.
- o A large work force (a peak level of 10,000 to 20,000) would be needed during construction; however, the operating staff would approach the lower range of these figures lending some long-term stability to area employment.

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- o Financial impacts would be large and could affect regional tax bases, labor rates, and capital investment in other activities.

It is unlikely that new, large power-generating fuel service centers will be fully implemented before the year 2000 because of the major technical, safety, environmental, and institutional barriers that must be overcome. However, implementation of such centers might be accelerated through the use of some similar facilities that already exist, e.g., Hanford, Washington; Idaho Falls, Idaho; Oak Ridge, Tennessee; and Savannah River, Aiken, South Carolina.

3. REACTOR LICENSING CONSIDERATIONS

Most of the world's nuclear power stations (operational or planned) are either pressurized-water or boiling-water light-water reactors. Seventy light-water reactors (51,500 MW) are currently licensed for operation in the United States, and an additional 91 reactors (99,700 MW) are licensed for construction as of August 1979; another 4 reactors (4,100 MW) have authorization for limited work. A similar number of light-water reactors are in operation, under construction, or planned in other countries.

Safety and environmental standards, regulatory requirements, and licensing procedures for light-water reactors have evolved, with increasing levels of technical understanding achieved through design studies, research and operating experience, and regulatory experience. This evolution is expected to continue, with increasing emphasis on risk-benefit evaluations of the safety and environmental aspects of nuclear reactor operation. Any meaningful licensing assessment of the principal alternative reactor concepts in the United States must be based on current light-water reactor safety and environmental requirements.

The safety and environmental aspects of each alternative reactor concept are discussed in the following sections. For each alternative, estimates were made of the time required to reach a stage of development at which a definitive commercial plant design could be prepared and a Preliminary Safety Analysis Report could be submitted to the Nuclear Regulatory Commission for review (see Table 3). This assessment is based solely on the likely effects that safety and environmental issues will have on licensing for the reactor concepts; the assessment of reactors is disassociated from licensing and construction of fuel-cycle facilities, economic considerations, market conditions, commercial risk considerations, and means of funding.

3.1 LIGHT-WATER REACTORS

3.1.1 Safety Aspects

Light-water reactors of the boiling-water reactor and the pressurized-water reactor types are the only plants the Nuclear Regulatory Commission has licensed in commercial-size units, with some plants exceeding 1200-megawatt (MW) electrical output. These plants have been licensed to use low-enriched uranium (3 to 5 percent U-235) fuel operated to an average burnup of about 30

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megawatt days per kilogram (MWd/kg)* on the once-through cycle. Some batches have been operated up to about 38 MWd/kg in licensed facilities.

Light-water reactor licensing regulations are codified in the Code of Federal Regulations, principally in Title 10. Many regulatory guides for meeting these criteria have been issued, and, in addition, Branch Technical Positions issued by the Nuclear Regulatory Commission further implement application of the regulations. However, even for this relatively mature technology, some unresolved safety issues remain. An interim report has been completed by the Nuclear Regulatory Commission and additional study is underway.

Unresolved Light-Water Reactor Safety Issues

Some aspects of light-water reactor design, construction, and operation are still under review by the Nuclear Regulatory Commission and by the Advisory Committee on Reactor Safeguards and are subjects of ongoing Nuclear Regulatory Commission and industry safety research projects. These aspects were reviewed by the Nuclear Regulatory Commission in a report to Congress prior to the Three Mile Island incident (Reference 1). These safety issues were not considered significant enough by the Commission to require plant shutdown for immediate modification, however, except for some plants which were shutdown for seismic design modifications. These issues are the types that may arise for other reactor concepts after they have achieved a comparable degree of operating experience. Some of the issues pertain only to the early light-water reactors and have arisen as a result of increased experience and more stringent interpretation of safety criteria. The issues pending completed Nuclear Regulatory Commission action are discussed below. It is expected that most of these remaining issues will be resolved before the end of 1981.

Water hammer--The concern is that water hammer, which has occurred in many light-water reactor fluid systems, could cause failure of a pipe in the reactor coolant system or could disable a system required to cool the plant after reactor shutdown.

Asymmetric Blowdown Loads on the Reactor Coolant System--Blowdown caused by events such as a primary coolant piping rupture would impose large, unevenly distributed loads on the reactor vessel, on reactor vessel internals, and on other components in the reactor coolant system. The potential for such asymmetric loads was identified only a few years ago and was not considered

*Megawatt-days of thermal power per kilogram of uranium fuel.

in the original design of some pressurized-water reactors. The Nuclear Regulatory Commission has requested all operating pressurized-water reactor licensees to assess the adequacy of the reactor vessel supports for these loads. The Nuclear Regulatory Commission will review these analyses and will conduct a pipe-break probability study. Boiling-water reactors are also susceptible to these asymmetric loads, and plans are being made to resolve this issue for boiling-water reactors as well.

Pressurized-Water Reactor Steam Generator Tube Integrity--The complex, corrosion-related phenomena of tube denting, stress corrosion, interactions between the tube and support plate, and tube cracking have severely affected several pressurized-water reactor plants. While the Nuclear Regulatory Commission reviews specific repairs, it is conducting generic studies to evaluate inspection results, failure consequences, coolant monitoring requirements, and design improvements. These studies will be used to revise current Nuclear Regulatory Commission requirements and guidelines.

Boiling-Water Reactor Mark I and Mark II Pressure Suppression Containments--Suppression pool hydrodynamic loads resulting from the drywell air and steam being rapidly forced into the pool during a loss-of-coolant accident and from various modes of safety/relief valve operation were identified; these loads had not been explicitly included in the design bases for plants with the Mark I and Mark II containment designs. The Nuclear Regulatory Commission determined that a detailed reevaluation of these containment system designs was required. Analyses made by industry and accepted by the Nuclear Regulatory Commission for the Mark I design have established that the plants could continue to operate safely, pending decisions on longer-term approaches.

The analysis of the Mark II plants was completed in October 1978. The Nuclear Regulatory Commission is evaluating confirmatory experimental and analytical programs to assess the margin for selected loads.

Anticipated Transients Without Scram--This issue concerns the possible failure of the reactor control rod shutdown system to scram after an anticipated transient and the probability that these failures will occur so infrequently that they can continue to be excluded from the design basis.

In April 1978, the Nuclear Regulatory Commission concluded that the reliability of current reactor scram systems was insufficient to preclude the occurrence of these failures and that means of mitigating their consequences would have to be provided in plant designs. Alternative means of reducing

the probability or the consequences of anticipated transients without scram events are being evaluated.

Boiling-Water Reactor Nozzle Cracking--The issue is whether or not the cracking that has occurred in the feedwater nozzles of 21 of the 23 boiling-water reactors and in the nozzles of control rod drives of two additional reactors has been arrested by the repairs that have been made. Excessive crack growth could impair pressure vessel safety margins, and the ensuing necessary repairs could expose personnel to additional radiation. The Nuclear Regulatory Commission issued interim guidance to operating plants in July 1977 and is continuing to review development of design modifications and crack-detection techniques.

Reactor Vessel Materials Toughness--Results from a reactor vessel surveillance program indicate that up to 20 older operating pressurized-water reactors were fabricated with materials that may have marginal toughness after comparatively short periods of operation. A Nuclear Regulatory Commission task scheduled for completion in 1979 was to evaluate material degradation mechanisms resulting from neutron irradiation and to determine the appropriate licensing criteria and corrective action for the low toughness of reactor vessels in these older plants.

Fracture Toughness and Potential Lamellar Tearing of Pressurized-Water Reactor Steam Generator and Reactor Coolant Pump Supports--Although these supports are designed for worst-case accident conditions, poor fracture toughness could cause the supports to fail during accidents. The Nuclear Regulatory Commission has completed a generic study of fracture toughness and a study of lamellar tearing is underway.

Systems Interactions in Nuclear Plants--The Advisory Committee on Reactor Safeguards requested that the Nuclear Regulatory Commission evaluate the reactor safety systems from a multidisciplinary point of view to identify undesirable interactions between plant systems. These interactions may be overlooked because the plant design and analysis are sometimes fragmented and assigned to teams of functional engineering specialists without adequate design integration. Sandia Laboratories is evaluating current Nuclear Regulatory Commission review procedures for adequacy in this regard.

Environmental Qualification of Safety-Related Electrical Equipment--Electrical equipment in safety systems (e.g., control circuits, instruments, and motors) must be qualified to work under accident conditions. Both the capability of equipment in older plants to perform under accident conditions and the

adequacy of tests and analyses of equipment in newer plants to qualify their equipment are questionable. The Nuclear Regulatory Commission has initiated an augmented inspection effort in the older plants, which concentrates on inspection of installed safety-related electrical equipment and on audits of the records of environmental qualifications under accident conditions. For newer plants, the Nuclear Regulatory Commission and industry are developing guidelines for implementing the applicable standards.

Residual Heat Removal Shutdown Requirements--A power plant must be able to go from hot-standby to cold-shutdown under all conditions, when that course of action is deemed best for safety. Accomplishment of this action in some plants is uncertain under all accident conditions and using the plant safety systems--as the recent experience with the Three Mile Island plant has shown. The Nuclear Regulatory Commission staff reviewed this issue and made changes to the Nuclear Regulatory Commission Standard Review Plan for plants. In addition, guidelines for residual heat removal system design requirements were incorporated into Regulatory Guide 1139, although these guidelines expected to be have not yet been issued in final form.

Control of Loads Near Spent Fuel--Plant operating procedures and designs may not protect adequately against the dropping of a heavy object, such as a spent fuel shipping cask, onto recently discharged spent fuel assemblies. An accident of this kind could result in overexposure of plant personnel or could cause offsite releases to exceed the guideline values in Title 10 of the Code of Federal Regulations, Part 100. The Nuclear Regulatory Commission is reevaluating current requirements and procedures to develop a revision to the Standard Review Plan, which can then be used to implement changes both in currently operating plants and in new plants.

Seismic Design Criteria--A number of plants have construction permits and operating licenses that were issued before the Nuclear Regulatory Commission's current regulations and regulatory guidance were in place. To ensure that these plants do not present an undue risk to the public, the Nuclear Regulatory Commission is reviewing the seismic design of these plants. It is also reviewing the seismic design sequence to determine its conservatism for all types of sites, to investigate alternative approaches to parts of the design sequence, and to quantify the overall conservatism of the design sequence. In April 1979, five plants were shut down pending validation of the seismic design basis and strengthening of seismic supports for equipment and piping. These actions have been completed and the plants have resumed operation. Additional reviews and a study on earthquake ground motion near the earthquake source are underway.

Pipe Cracks at Boiling-Water Reactors--Pipe cracking has occurred in the heat-affected zones of welds in the primary system piping in boiling-water reactors since the mid-1960's. The phenomenon is believed to be understood, and adequate repairs or design modifications have been made. Nevertheless, some recent occurrences in large-diameter pipes and in pipes of several materials have led the Nuclear Regulatory Commission to reestablish a pipe crack study group to address the pertinent issues: the significance of the recent occurrences relative to conclusions in NUREG-0313 on this subject, the adequacy of detection and inspection techniques, and the potential for stress-corrosion cracking in pressurized-water reactors. A report on these issues has been completed. The Nuclear Regulatory Commission is also conducting several generic technical reviews that focus on improving piping inspection techniques and requirements.

Containment Emergency Sump Reliability--After a major break in the reactor coolant system piping, the containment emergency sump would collect the water flowing from the break, and the emergency core-cooling system pumps would recirculate this water to the containment spray system. The concern is that debris in the sump or abnormal conditions (e.g., air entrainment, vortices, or excessive pressure drop) would inhibit or prevent the pumps from providing adequate recirculation. Regulatory guide 1.82 provides guidance on pump protection from debris and required pump redundancy. However, the Nuclear Regulatory Commission is continuing to study the behavior of pipe insulation debris under accident conditions. Regulatory Guide 1.79 addresses testing of the recirculation function, and the Nuclear Regulatory Commission staff believes that pumps tested in accordance with this guide resolve the issue. Study of the issue to provide further guidance on hydraulic design and review of emergency sumps is continuing.

Station Blackout--The issue is whether or not nuclear power plants should be designed to accommodate a complete loss of all alternating-current (a.c.) power including offsite sources and onsite emergency diesel generators. Long-term loss of a.c. power at pressurized-water reactors accompanied by loss of auxiliary feedwater pumps could result in an inability to cool the reactor core. Current Nuclear Regulatory Commission requirements demand that diverse power drives be provided for the redundant auxiliary feedwater pumps, normally accomplished by an a.c.-powered electric motor and a redundant steam-turbine-driven pump. Adequacy of the design of the plants licensed prior to adoption of the current requirements is of concern. An initial survey of operating plants shows that all plants have some capability for accommodating an extended loss of a.c. power. The Nuclear Regulatory Commission is continuing its further review of older plants to determine if additional requirements are needed, for example, the time for retaining this core-cooling capability.

Reviews of the Incident at Three Mile Island

The major incident at the Three Mile Island Nuclear Power Station occurred after the Nuclear Regulatory Commission issued its report to Congress in January 1979, which described the currently unresolved safety issues discussed above. This incident led to major reviews of the licensing criteria, design requirements, operating procedures, and operator qualifications by the Nuclear Regulatory Commission, the Kemeny Commission, several congressional committees, and industry. The full impact of this incident on reactor licensing cannot be known until these reviews are completed. The principal Nuclear Regulatory Commission reviews are the Lessons-Learned Task Force and the Three Mile Island Special Inquiry which is an independent evaluation headed by M. Rogovin, due to be completed by January 1980. The Kemeny Commission report was issued in late October and its recommendations are now being evaluated. The Congressional reviews being conducted in the House of Representatives and the Senate are expected to extend into 1980 and to result in significant changes to the licensing process and requirements.

The initial report (Reference 2) by the Nuclear Regulatory Commission Lessons-Learned Task Force in July 1979 disclosed a number of possible improvements in areas of design, analysis, and plant operations that the Task Force recommends to be required in the short term in order to provide additional protection which is required for the public health and safety. Most of these short term recommendations are consistent with existing Nuclear Regulatory Commission regulations, regulatory guides, and the staff's standard review plans. Out of the 23 recommendations there are only three that would require revisions of present regulations. These three are:

- o Making inert all boiling-water reactor containments,
- o Providing the capability to install external hydrogen-oxygen recombiner at each light-water reactor facility, and
- o Revising the limiting conditions for operation based on the total loss of safety system availability through operator error.

Beyond these short term issues, however, the accident at Three Mile Island has raised a number of other broader and more fundamental questions concerning the design of nuclear power plants. Subsequent to the initial report, the Lessons-Learned Task Force concentrated on fundamental issues which are associated with the following areas of the design:

General Safety Criteria--The underlying philosophy of nuclear reactor safety has been the concept of "defense in depth." The Task Force has concluded that although the concept is sound and it was not challenged by the occurrence of the accident, there is a need to improve the implementation of the concept in determining safety requirements. Specifically, there will be consideration of whether revisions or additions to the General Design Criteria which implement the "defense-in-depth" concept, are necessary in light of some of the occurrences during the accident. A central issue that will be considered is whether to modify or extend the current deterministic approach to defining design basis events or to depart from this approach and make more extensive use of fault-tree, event-tree analysis. For example, analysis of design-basis accidents could be modified to include multiple equipment failures and more explicit consideration of operator action or inaction rather than employing the conventional single-failure criterion. Alternatively, analyses of design basis accidents could be extended to include core uncovering or core-melting scenarios. Risk assessment and explicit consideration of accident probabilities and consequences might also be used.

System Design Requirements--The adequacy of system design requirements has been questioned in view of demonstrated disparities between description and evaluation of accidents in the licensing review of the Safety Analysis Report and the actual response of the plant and its operators. Specifically, the system design requirements judged to be the most important and selected for future study are the classification and requirements for non-safety-grade systems and components, operator interactions, and post-accident design requirements. These are described below.

- o Non-Safety System Classification and Requirements. These systems are currently assumed to be non-functional for mitigation of accidents, and no special criteria are applied with the exception of control systems. This approach will be reassessed in terms of possible reclassification of safety and non-safety systems and added special requirements for some non-safety systems.
- o Operator Interactions. The current regulation covering the operator-systems interaction immediately following the initiation of a transient or an accident will also have to be reassessed. At present, no credit is assumed for operator action during the period of time immediately after the accident initiation, and all required steps are assumed to be automatic. This assumption ignores the possibility of operator errors which occurred several times during the Three Mile Island accident, and will be re-examined.

- o Post-Accident Design Requirements. The ongoing reexamination of design requirements for post-accident operations includes availability of post-accident monitoring instrumentation, provisions for storage and treatment of large quantities of radioactive liquid and gaseous wastes and procedures for handling other anticipated post-accident problems on site.

In addition to the plant design issues, the Lessons-Learned Task Force addresses concerns raised with regard to the criteria for the organization, qualification and training of the utility staff who operate the plant and it is studying means of improving the quality of the licensing review process.

Issue identification and resolution is a continuous experience as operating experience increases. Because each of the alternatives to the current light-water reactor has not yet reached the same level of maturity, many of the concerns surrounding them are more general in nature. Undoubtedly, additional issues, which are currently unforeseen, will arise and will need to be resolved by further research, development, and demonstration. Thus, the reader is cautioned not to make summary judgments of the licenseability of alternative systems based merely on the number of issues identified in the following pages or in the space devoted to each system. Resolution of many of these light-water reactor unresolved safety issues will impact the other concepts.

Light-Water Reactor, 15 Percent-Improved Low-Enriched Uranium, Once-Through Cycle

The principal modifications will be an increase in fuel enrichment to about 4.3 percent; an increase in burnup to 50 MWd/kg; use of a five-batch, annually refueled core; and revised management of fuel during residence in core. These improvements are directed toward a 15 percent reduction in annual uranium consumption.

The evaluation of selected safety parameters indicates that the reactor design with five-batch fuel management and with the 50 MWd/kg burnup has a high probability of being licensed. A fuel element test program is under way in commercial reactors with current-design light-water reactor fuel assemblies being irradiated to extended burnups (Oconee 1 for pressurized-water reactor fuel; Big Rock Point and Oyster Creek for boiling-water reactor fuel). Later, demonstration assemblies, designed specifically for higher burnup, will be irradiated in other reactors. Although not required for safety reasons, batch reloads will be irradiated as the final step in the program. Ramp transient

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tests will be performed in test reactors for safety reasons. Thus, the research, development, and demonstration program will demonstrate that the fuel/cladding integrity, both during normal operation and during ramp transients, remains in an acceptable range up to an average burnup of 50 MWd/kg and peak burnups of 60 to 75 MWd/kg. Although no fundamental problems are expected in the successful completion of this program, some safety considerations exist that must be addressed, as follows:

- o The shutdown control-rod worth may be smaller (about 10 percent lower); the result of this lower control-rod worth is a tendency for reduced reactivity margins for certain accidents (i.e., steam-line break). The reduced margin may lead to difficulty in licensing, depending on the amount of rod worth margin that is available for a particular reactor design.
- o Delayed neutron fraction at the beginning and end of an equilibrium cycle is expected to be somewhat smaller but should not cause problems in the safety evaluation of the design.
- o The moderator temperature coefficients in five-batch fuel management are expected to be somewhat more negative than those in three-batch fuel management. For cold-water incidents, such as the steam-line-break incident or the idle-loop-startup incident, the more negative coefficient increases the reactivity inserted by the cold water and results in a more limiting accident. The acceptability of the more negative moderator temperature coefficients requires analysis of an actual design with detailed design methods. The more negative coefficients are not expected to lead to unacceptable results, but verification is needed. Lattice changes, such as reduced pin diameter, may also be implemented at the same time as the extended burnup to reduce U_3O_8 consumption and will cause the moderator temperature coefficients to be slightly less negative, tending to counteract the effect of going to the five-batch fuel management.
- o Small perturbations will be introduced in decay-heat rates because of the larger inventories of fission products that result from higher burnups. These changes will be very small for short-term decay heat, which is of primary concern for accident analysis. Thus, from a safety analysis point of view, these perturbations in decay heat should be insignificant.

- o The larger fission product inventories have the potential for higher offsite radiation doses. However, the I-131 inventory, which is the most important short-term hazard, is unchanged. A design goal believed to be achievable is that the same reactor primary coolant radioactivity limit as that in the low burnup fuel will be maintained; this limit is now substantially higher than operative experience with current fuels that have reached about 38 MWd/kg burnup. The current high burnup fuel development programs are expected to demonstrate fuel failure rates no higher than current fuel designs.

Light-Water Reactor, Low-Enriched Uranium, Uranium/Plutonium, Self-Generated (Spiked) Recycle

Extensive safety and licensing reviews have been performed for plutonium-uranium mixed-oxide fuels, and extensive fuel irradiation programs have been conducted. The Big Rock Point, Dresden Unit 1, and Quad Cities Unit 1 boiling-water reactors were licensed to operate with mixed-oxide fuel assemblies. No fundamental safety-related problems have been identified for the recycling of self-generated plutonium in light-water reactors. Assuming that the reactor plant design is basically the same as that for the currently licensed light-water reactor, the reactor-fuel combination concept should be fundamentally licensable from a safety viewpoint. The differences in neutronic and physical characteristics of the fuel, although small, will nevertheless require detailed analysis of transient and accident behavior. Additional testing may be required if future performance standards are upgraded from those in effect at the time of earlier tests.

Some basic differences of the mixed-oxide core compared with the uranium dioxide core are the more negative Doppler, moderator temperature, and void-reactivity coefficients; smaller delayed neutron fraction; and shorter prompt-neutron lifetime. The combined effect of these differences may increase or decrease the severity of some events, depending on the type of event or on the core design. This difference can be accommodated by design modifications, if necessary. Changes in reload fuel equipment and handling procedures would be required to protect the public and the reactor plant workers if the plutonium recycle fuel were spiked with high levels of gamma-ray activity for safeguards reasons. These changes would include the use of shielded casks for offsite shipping and inplant transfer and shielded facilities for storage of fresh fuel; the changes would be demonstrated during the development program.

Light-Water Reactor, Denatured U-233/Thorium Cycle

Preliminary evaluations of the characteristics of thorium-containing cores indicate that the core response during postulated accidents is satisfactory (Reference 3). Assuming that the reactor plant design is basically the same as that of the currently licensed light-water reactor, this reactor-fuel combination concept should be fundamentally licensable. Thorium/uranium cores were licensed and operated at the Indian Point 1 pressurized-water reactor and the Elk River boiling-water reactor; a partial loading of ThO_2 fuel was licensed and operated at the Dresden-1 boiling-water reactor.

The physical properties of thorium-bearing fuels and core properties (such as coefficients of reactivity and control-rod worth) are somewhat different from those of cores fuelled with uranium dioxide. An analysis of the behavior of thorium dioxide-based cores indicates that all postulated events can be expected to be in the range predictable for uranium dioxide or mixed uranium-plutonium oxide cores. It will still be necessary to demonstrate by further detailed analysis and experiment that the safety performance of the light-water reactor fueled with denatured uranium/thorium falls within the criteria established for uranium dioxide-fueled operations.

In addition, appropriate safety criteria, such as limits on acceptable fuel design and on maximum energy deposition in the fuel, must be determined. Changes in core physics parameters that could result in altered fuel loadings and the implications of these changes for reactor design and safety need to be quantified. As with the spiked plutonium recycle, the denatured U-233 fuel will have to be shielded during shipping, transfer, and storage to protect the public and inplant workers from the high gamma-radioactivity associated with U-233 fuels.

The development program should consist of extensive thorium dioxide fuel irradiations to demonstrate that these fuels perform safely under reactor conditions at high burnups. The performance of the fuels during severe transients must also be demonstrated to be satisfactory in independent ramp tests. An experimental physics verification program should be conducted. Procedures for safely shipping and loading radioactive fuels must be demonstrated.

Light-Water Reactor Prebreeder, Plutonium/Thorium (Spiked) Recycle

The comments made for the previous reactor-fuel-cycle combination are valid for this concept. Fundamentally, the concept is licensable, but differences in neutronic and physical characteristics will require detailed analysis of transient and accident behavior and demonstration of acceptable performance in reactors. As with the spiked plutonium/uranium recycle, shielding will be needed during shipping, transport, and storage to protect inplant workers and the public from gamma-radioactivity.

3.1.2 Environmental Aspects

In this part of the assessment, environment effects are assessed for the light-water reactor only, using the current pressurized-water reactor design with low-enriched uranium fuel on the once-through fuel cycle as a reference, and assuming a similar site for all the alternatives. Often, environmental effects (those resulting from radioactive releases in particular) can be presented only qualitatively because irradiation performance data on fuel types being considered are not available.

Environmental issues considered in licensing a light-water reactor plant include land use; water use; thermal and chemical discharges; radioactive effluents in gaseous, liquid, and solid forms; and occupational exposure to radiation during normal reactor operation. The licensing and regulatory policy and procedures for environmental protection are described in Title 10 of the Code of Federal Regulations, Part 51. Changes to this regulation will be required for all the alternative fuel cycles, particularly for Tables S-3 and S-4 which are limited to the uranium fuel cycle with initial fuel enrichment of up to 4 percent and fuel burnups up to 33 MWd/kg.

Land Use

The land use at the reactor site will be changed very little for any of the light-water reactor fuel-cycle alternatives. For the once-through cycle, the spent fuel would be shipped either offsite to a centralized interim storage facility or to a geologic repository, or it would be held onsite in an interim storage facility. If shipped offsite, no effect on land use would occur at the reactor. If held onsite, additional spent-fuel storage facilities might have to be built eventually, although many reactors can provide additional storage for up to 10 years by reracking the existing storage pool to allow higher-density storage. When additional onsite storage facilities are needed,

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they can generally be accommodated within the exclusion area boundary required for reactor accident considerations.

For the recycle alternatives, the spent fuel would be shipped offsite to a reprocessing plant, and the fuel cycle would have no effect on land use at the reactor. The recycle fuels with high gamma-ray activity (U-233/thorium fuels and high-burnup uranium-plutonium fuel) would require shielded storage, but this storage could be provided as for spent fuel with little impact on land use at the reactor.

Thermal and Chemical Discharges

The thermal discharge from the reactor plant depends on the thermal efficiency of the plant. Plant thermal discharges are about 2 MW of heat per MW of electricity generated for all the light-water reactor fuel cycle alternatives because the plant is unchanged.

The chemical discharges from the reactor plant are principally from the steam-generating, feedwater-treatment, and heat-rejection (i.e. cooling tower) equipment. These components are not affected by the fuel cycle, and thus the chemical discharges are unaffected.

Radiological Effluents

Regulatory requirements governing radioactive releases are established by the Nuclear Regulatory Commission and are included in the Code of Federal Regulations, Title 10, Parts 20 and 50 (Appendix I) and associated Regulatory Guides. Part 20 establishes limits for releases during normal operation, and Part 50 (Appendix I) provides numerical guidance for conformance with the policy of "as low as reasonably achievable." These Nuclear Regulatory Commission requirements conform to the Environmental Protection Agency guidance for total radioactivity releases to the environment for the whole fuel cycle, which are set forth in Title 40 of the Code of Federal Regulations, Part 190.

The radiological effluents from the alternative light-water reactor fuels will be required to meet these regulations for licensing. The "as low as reasonably achievable" policy will result in effluents from all the alternative fuels being substantially below the limits for release during normal operation. The high-burnup fuel case is expected to have a potentially

higher radioactive source term. However, the consensus is that the reactor coolant radioactivity operating limit can be maintained the same as that for the lower burnup fuel cycle. This limit is now substantially higher than operating experience with current fuel that has reached over 30 MWd/kg burnup, and the current high-burnup fuel development programs are expected to demonstrate fuel failure rates no higher than current fuel designs. For all the reasons above, the four alternative light-water reactor concepts are environmentally similar to the reference light-water reactor plant.

Occupational Exposures

The fuels with high gamma-ray activity will have to be handled with procedures and equipment similar to those used for spent fuel. Occupational exposures to radiation at the reactor for the uranium/plutonium self-generated recycle (which may be spiked for safeguarding), the denatured U-233/thorium recycle, and the plutonium/thorium spiked recycle fuels can probably be kept to levels equivalent to those experienced with low-enriched uranium fuel by adequate shielding, safety interlocks, and administrative procedures.

3.2 WATER-COOLED BREEDER REACTORS

Conceptual designs for both light-water- and heavy-water-cooled-breeder reactors have been prepared by the Department of Energy Division of Naval Reactors. The NASAP studies in this volume examine only the light-water version of the water-cooled breeder reactors.

The light-water breeder reactor can produce slightly more fissile material than it consumes. It would also make extensive use of current light-water reactor technology and would possess the safety characteristics of pressurized-water cooled and moderated reactors. However, the efficient fuel utilization of the light-water breeder reactor depends upon the use of U-233/thorium fuel recycle, and large amounts of U-233 are needed for the initial breeder reactor fuel loading. In subsequent years, potentially no additional U-233 is needed. The U-233 needed for the initial breeder reactor fuel loading must be made in prebreeder reactors that are light-water reactors loaded with thorium fuels. The prebreeder fuel can be either thorium and uranium enriched in U-235 or thorium mixed with plutonium.

A typical light-water breeder reactor commercial deployment strategy would consist of two stages: (1) a prebreeder phase to produce U-233 and (2) a light-water breeder reactor to use the U-233. These stages are sometimes

accomplished in two separate reactor designs, but in some cases, a single reactor design with two sequential fuel types is used in the commercial deployment strategy.

3.2.1 Safety Aspects

The 60-MW Shippingport light-water breeder reactor had operated successfully for more than 12,000 hours as of the end of September 1979, with a core loading of U-233/thorium fuel. Nuclear Regulatory Commission licensing was not required because the Department of Energy operates the reactor. However, the Nuclear Regulatory Commission and the Advisory Committee on Reactor Safeguards reviewed the reactor and considered the unique features of this modified light-water reactor, as discussed below, to be acceptable for the 60 MW experimental reactor with a planned 3-year operating period. The Nuclear Regulatory Commission has not yet reviewed a large, commercial-size light-water breeder reactor design, but some issues have been identified (e.g., emergency core cooling system; transient accident analysis; and, for the advanced breeder concept, advanced breeder control). These issues can probably be resolved during the engineering development of commercial plants. The design-basis criteria reviewed by the Nuclear Regulatory Commission for the Shippingport Light-Water Breeder Reactor are not expected to change for the commercial light-water breeder reactor design, except as these criteria may be generally changed for all reactors as a result of the ongoing reviews of the Three Mile Island incident.

Prebreeder and Breeder Reactors Based on Light-Water Breeder Reactor Type I Modules

The prebreeder and breeder core concepts would contain modules similar to the Type I modules now being used in the Shippingport Light-Water Breeder Reactor. The conceptual prebreeder module would be identical with the light-water breeder reactor Type I module, with the exception of the fuel in the fuel elements. From a safety standpoint, the conceptual breeder module would be very similar to the light-water breeder reactor Type I module in Shippingport.

In addition to reviews by the Nuclear Regulatory Commission and the Advisory Committee on Reactor Safeguards, the Shippingport reactor was also reviewed by an independent technical group appointed by the Commonwealth of Pennsylvania. The reviews uncovered no safety-related licensing issues that indicate concerns significantly different from those of the commercial pressurized-water reactor. Also, the unique features of the light-water breeder reactor cores

were found to be acceptable for the 60 MW experimental reactor with a planned 3-year operating period. These features are described below.

- o Tight lattice. The rod-to-rod spacing in the light-water breeder reactor is 60 mils, as opposed to the 120 mils used in commercial pressurized-water reactors. The close spacing requires a different fuel element and grid design and different fuel assembly procedures, compared with those used in commercial practice. Design assessments of fuel-rod bowing in the reactor core predict that rod-to-rod contact is unlikely to occur, and even if it did, based on experimental data, it is calculated to be acceptable. Bowing analysis of the blanket has also been completed on a worst-case basis. These data indicate that adequate margins can be incorporated and that a commercial-scale reactor of this design should be feasible. Additional thermal-hydraulic testing may be required to confirm that the successful design of the reactor with tight lattice spacing can be applied to commercial-scale light-water breeder reactors. Such testing would help to establish standards for a commercial emergency core-cooling system for the light-water breeder reactor.
- o Core thermal margins. The thermal analysis programs and correlations have been proved applicable for thermal-hydraulic analysis under widely ranging axial and radial heat flux distributions in a close-packed array including a coupled-region interface. Additional critical heat flux testing may be required to confirm the acceptability of the breeder seed lattice.
- o Provision for accident prevention. The probability of accident initiation is comparable to that of any pressurized-water reactor. Safety and protection features, including the emergency core-cooling system, have been designed in accordance with regulatory guides and requirements. No impediment to providing comparable protection features in a larger plant appears to remain.
- o Acceptability of movable fuel for reactivity control. All reactivity control functions required by an operating nuclear power plant can be satisfactorily performed by means of the movable fuel. For large-core applications, design will require additional consideration because the reactivity increases available from one control assembly during power operation is greater in the light-water breeder reactor than in current light-water reactors. In a conventional core, control rod design must meet the requirement that ejection of the rod is impossible; to design the movable fuel to meet that requirement is feasible.

- o Power and temperature coefficients. Calculations indicate that power and temperature coefficients for the prebreeder concept would be between those of current commercial pressurized-water reactor cores and the Shippingport Light-Water Breeder Reactor core. For the breeder concept, they would be comparably more negative than would those of the Shippingport core.
- o Control stability and adequacy. Experience gained from performance of the Shippingport plant indicates that control stability and adequacy can be maintained in a core with movable reactivity control.
- o Nuclear stability. Analytical studies (Reference 4) show that U-233/thorium systems are inherently more stable than uranium-fueled pressurized-water reactors, primarily because of lower total xenon yields, a larger fraction of xenon-chain yield direct to xenon, and more negative Doppler coefficients of reactivity. Data from initial testing at Shippingport provide confidence that the breeder concept would have acceptable stability properties. In the prebreeder concept, the stability should be at least as good as that of a commercial pressurized-water reactor because it would operate in the range from less than half to equal to the specific power of pressurized-water reactor cores.

The major unique feature of the prebreeder concept is the fuel pellet. If highly enriched uranium or plutonium is mixed with thorium for fuel, a homogeneous, binary solution fuel is used. However, if a medium-enriched uranium/thorium fuel is used, a duplex pellet is employed, in which a fissile-fuel-bearing uranium dioxide annulus is placed around a thorium dioxide core. The absence of fissile fuel in the thorium core, initially would cause most of the fission energy to be generated close to the cladding and therefore would result in lower average fuel-element temperatures than would those obtained with a solid uranium dioxide pellet. This arrangement would have the advantage of reducing stored heat, which is important during a loss-of-coolant accident. As irradiation proceeds, U-233 would be bred in the thorium core, and the power generation would shift from the outside annulus to the central core. In this duplex pellet concept, the annulus would experience a peak burnup of 23×10^{20} fissions per cubic centimeter (cm^3) of ternary fuel (this would correspond to 90 MWd/kg of uranium if the fuel were UO_2 rather than the ternary $\text{UO}_2\text{-ZrO}_2\text{-CaO}$). In contrast, the peak burnup in the light-water reactor, 15 percent modification is about 75 MWd/Kg of uranium. Currently, little information is available on the structural and thermal characteristics of fuel at high burnup, on fission-gas release, on fuel swelling and cladding strain rate, on fuel-cladding response to power transients, and on the effect of the high-temperature, inner annular surface on defective-rod performance. These matters would have to be investigated in the

research, development, and demonstration program before the capabilities of such a system could be established. This fuel system is being evaluated for possible further development under the Department of Energy's Water-Cooled Breeder program, including irradiation testing and analytical work to extend the scope of current fuel-element modeling and irradiation testing.

In common with all thorium-containing fuels, the light-water breeder reactor U-233 recycle fuel will also contain U-232 and must be shielded during shipping, transfer, and storage because of the fuel's high gamma-ray activity.

Light-Water Backfit Prebreeder Supplying Advanced Breeder

The prebreeder would utilize 15.4 percent enriched U-235, thorium dioxide duplex fuel, whereas the fuel of the advanced breeder would consist of a binary solid solution of highly enriched uranium (>82 percent) and thorium dioxides in the form of pellets.

The safety-related features typical of the breeder in this concept are similar to those listed for the prebreeder and breeder reactors based on light-water breeder reactor Type I modules. They were evaluated in the Shippingport Light-Water Breeder Reactor Safety Analysis, and were judged acceptable. The essential difference between this concept and the currently operating reactor at Shippingport would be that, instead of movable bundles of fuel rods (as in the Shippingport core), movable thorium dioxide control rods (arranged in a manner similar to the boron poison rods now being used in commercial pressurized-water reactors), would be interspersed throughout the module. Poison rods for shutdown only may also be required, either in the same modules as the thorium dioxide rods or in alternate modules. A program to determine the required additional development of these control methods is currently being evaluated. Present-day technology would need to be extended in the areas of fuel-assembly grids, thorium dioxide control rods, and the plenum.

Qualification of the fuel for the prebreeder will also be needed, as was discussed earlier for thorium-based light-water reactors and for the prebreeder based on Type I modules. This qualification will require an extensive program of fuel testing for behavior during normal operation and during transients.

Light-Water Backfit Prebreeder/Seed-Blanket Breeder System

The backfit prebreeder would use uranium fuel enriched to 93 percent U-235 as a binary solid solution of thorium and uranium dioxides in the form of pellets. The seed-blanket breeder fuel would consist of seed rods containing a binary solid solution of highly enriched U-233 and thorium dioxides, and the blankets would consist of thorium dioxide pellets.

The safety-related features typical of the backfit prebreeder/seed-blanket breeder system are shared by the current pressurized-water reactors, the Shippingport Light-Water Breeder Reactor, and the advanced breeder. Features of the breeder in this system not yet proved licensable are the large-scale use of a close-packed lattice seed-blanket core design and the use of thorium dioxide in control rods. Use of the close-packed lattice may have some effect on performance in a loss-of-coolant accident when the system is scaled to commercial size. Some changes in sizing and setpoints of the auxiliary and emergency core cooling systems of typical light-water reactor plants may be needed to accommodate this effect. The thermal-hydraulic behavior of the close-packed core during loss-of-coolant accidents and its response to emergency cooling will have to be verified in the development program.

Light-Water Backfit Low-Gain Converter Supplying Light-Water Backfit High-Gain Converter

The low-gain converter would utilize medium-enriched U-235 (<20 percent) and thorium in the form of thorium oxide pellets alternating with duplex pellets containing the uranium oxide in the annulus.

The high-gain converter would be fueled with binary uranium oxide-thorium oxide pellets containing about 1.3 percent U-233. The blanket would consist of thorium oxide. Two versions of the high-gain converter are being considered: a low burnup fuel (11 MWd/kg) and a high burnup fuel (22 to 30 MWd/kg).

The safety-related features typical of the breeder reactor in this concept are similar to those listed for the prebreeder and breeder reactors based on light-water breeder reactor Type I modules. They were evaluated in the Shippingport Light-Water Breeder Reactor Safety Analysis, reviewed, and judged acceptable. The essential difference between this concept and the currently operating Shippingport reactor would be that instead of movable bundles of fuel rods, movable rods containing thorium dioxide, arranged in a manner similar to the poison rods being used in commercial pressurized-water

reactors, would be interspersed throughout the module. A program to establish the expected performance of such a design and to provide supporting physics and engineering test data is being evaluated as part of the Department of Energy's Water-Cooled Breeder Reactor program. The high-gain converter would require thermal-hydraulic testing of the close-packed lattice, in common with other light-water reactor concepts. Also, if the high-burnup fuel were specified for the high-gain converter, additional fuel qualification tests beyond those currently planned in Shippingport would be required.

3.2.2. Environmental Aspects

The environmental impacts of the light-water breeder reactor or the prebreeder, excluding consideration of the fuel cycle facilities, are not expected to differ significantly from those of the pressurized-water reactor.

With regards to land use, the light-water breeder reactor and the prebreeder are basically the same as the pressurized-water reactor. Although some design versions of the prebreeder/breeder have a slightly larger pressure vessel, the land use will not be affected because the exclusion area for the reactor plant is established on the basis of the thermal power level of the reactor core, prescribed fission product release fractions, and site-specific meteorology. Because the thermal efficiency of the light-water breeder reactor is the same as that of the pressurized-water reactor, the land use will be the same for a particular power output and site.

The thermal discharge from the light-water breeder reactor will be the same as the pressurized-water reactor because the thermal efficiency of the two plants is the same. Likewise, the chemical discharges will be the same, because the heat transport and rejection systems are the same.

Radiological releases from the light-water breeder reactor or prebreeder are expected to be less than or equal to those from a pressurized-water reactor. In general, the concepts with lower fuel burnups are associated with lower fission product release. The concepts in which soluble boron is not used for reactivity control are associated with lower tritium releases.

Occupational exposures at the reactor can probably be kept to levels equivalent to those for the light-water reactor low-enriched uranium fuel, if adequate shielding, safety interlocks, and administrative controls are applied

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to prevent exposure caused by the high gamma-ray activity of the U-233 recycle fuel.

3.3 HEAVY-WATER REACTORS

3.3.1 Safety Aspects

The Canadian pressurized heavy-water reactor is licensed and operating successfully in Canada and in several other nations. To a large degree, the Canadian licensing requirements have developed independently of those for the light-water reactor in the United States; some Canadian requirements appear to be more conservative and some less conservative than those in the United States. In considering the licensability of heavy-water reactors in the United States from a safety viewpoint, it is useful to distinguish between design characteristics that are intrinsic to the nuclear steam supply system and those that are extrinsic (i.e., pertaining to peripheral subsystems that must be relied upon to operate to provide safe shutdown).

Current Nuclear Regulatory Commission regulatory requirements were developed for the light-water reactors with cognizance of their intrinsic safety-related characteristics. Had these regulations been written for another reactor type, with different characteristics, they would have been worded differently but presumably with the same ultimate intent.

Two separate studies concerning licensability of the Canadian pressurized heavy-water reactor in the United States used two different approaches. In one study, the extrinsic features of the Canadian reactor were reviewed and the required design changes for conformance with U.S. licensing criteria were identified (Reference 5). That study concluded that no fundamental problem would occur due to modifying the extrinsic features of the system to conform to appropriate U.S. criteria, although the recommended design changes would have significant impact on plant capital cost.

The second study focused on the intrinsic safety features of the Canadian reactor and emphasized the intent of U.S. criteria and design codes. The study concluded that relatively few modifications would be required to obtain conformance with the intent of U.S. requirements (Reference 6).

As part of the NASAP studies, modifications were made to the Canadian pressurized heavy-water reactor design to enhance its competitive position under U.S.

economic conditions, to improve fuel utilization, and to conform more closely to U.S. licensing requirements, construction practices, and sites (Reference 7). The major changes included:

- o Increased plant output of 1,260 MW as compared with the Canadian design of 600 MW.
- o Use of slightly enriched (1.2 percent U-235) uranium rather than natural uranium.
- o Primary coolant pressure of 2,250 psi rather than 1,600 psi; coolant-tube wall thickness was increased.
- o Reduction of the heavy-water reflector to one-half the thickness in the Canadian design.
- o Use of a two-loop primary system rather than the three loops used in the Canadian design.

The following intrinsic safety-related characteristics of the modified design remain substantially different from those of light-water reactors and would require substantial licensing review and, in some cases, research, development, and demonstration programs:

- o The presence of a large heat sink (the heavy-water moderator) that is dispersed in the core region and is provided with a continuously operating heat-removal system. This latter system will serve as a backup emergency core-cooling system in the highly unlikely event that a postulated, low-probability loss-of-coolant accident is followed by failure of the primary ECCS, or in the unlikely event of loss-of-primary-heat-sink accident.
- o The subdivision of the core into a large number of relatively independent power channels. This characteristic allows the primary heat-transfer system to be subdivided into two or more hydraulically independent loops; the characteristic also permits the rapid detection and location of fuel failures.

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- o The use of an on-line refueling system that allows excess reactivity to be kept low at any time during the fuel cycle and that permits easy on-line replacement of failed fuel. This system would have to be analyzed for behavior during accidents and natural disasters.
- o The use of a core lattice with power channels placed at a relatively large pitch, leaving ample space for the introduction of neutron poison devices for control and safety purposes in the low-pressure, low-temperature moderator region of the core.
- o The use of a coolant with a positive void coefficient of reactivity (negative coolant-density coefficient of reactivity). Loss of all coolant from the core region results in a relatively small reactivity increase.
- o An overall power coefficient of reactivity (at nominal power close to zero) that may be either slightly negative or slightly positive.
- o The presence of instability from xenon oscillations, if no external control is applied.
- o The use of a nonferrous material (zirconium-niobium alloy), exposed to the full neutron flux, as part of the pressure boundary of the primary heat-transport system.
- o The use of rolled joints to connect zirconium-niobium alloy pressure tubes to the stainless steel end fittings.

The overall conclusion is that the modified version of the Canadian pressurized heavy-water reactor is expected to be licensable in the United States, although some further modifications in design details might be required. The degree of modification would be determined from a Nuclear Regulatory Commission review of a Safety Analysis Report. These differences would have to be addressed in the research, development, and demonstration program. The use of a computer to control the reactor, which is done in Canada, may be an important licensing issue. Programs for testing components would be required to qualify pressure tubes and pressure-tube joints as pressure boundaries and to ensure that the on-line refueling equipment satisfies U.S. seismic criteria. Extensive testing would be required to define safety-related heat transfer criteria in the fuel bundle geometry for the modified Canadian design because of the higher coolant pressure. The adequacy of fuel hardware for burnups of 20 MWd/kg or over used in the modified design would

have to be proven in fuel irradiation tests with the slightly enriched fuels. From the safety viewpoint, these programs could be carried out in parallel with the design and licensing of the first heavy-water reactor.

3.3.2 Environmental Aspects

The thermal efficiency of the heavy-water reactor is about 29 to 30 percent, compared with about 33 percent for the light-water reactors. The effects of thermal effluents and chemical releases (which are related principally to the operation of the heat-dissipation system) from a heavy-water reactor are therefore about 15 to 20 percent higher than those of a light-water reactor. These releases and consequent environmental impacts are not considered significant.

Radioactive releases (gaseous and liquid) from a heavy-water reactor are smaller than those of a light-water reactor, except for tritium. Predicted tritium release in gaseous discharges to the atmosphere are 30 to 80 times larger than the corresponding releases from the reference light-water reactor. Tritium release in liquid discharges is also greater by a factor of 30 to 60. The dose effect, which depends on the site-specific dilution factor, may or may not present a problem, depending on the site.

Occupational exposures are expected to be similar to those in light-water reactor plants, as shown by experience with heavy-water reactors operating in Canada. Procedures for personnel protection may be more restrictive because of the increased levels of tritium concentration within the plant.

In general, no significant adverse radiological impact is expected from heavy-water reactor operation in the United States.

3.4 HIGH-TEMPERATURE GAS-COOLED REACTORS

The two types of high-temperature gas-cooled reactor plants studied by NASAP are:

- o A steam-cycle concept in which the helium reactor coolant transfers heat to steam generators to drive the steam turbine-generator to produce electricity; this concept is analogous to the "indirect" cycle used in

the pressurized-water reactors, wherein the pressurized water transfers heat to steam generators.

- o A "direct-cycle" concept in which the helium reactor coolant converts heat to work by expanding through helium turbines that drive electric generators and compressors; this concept is analogous to the direct cycle used in the boiling-water reactors, wherein the steam from the boiling reactor coolant water drives the steam turbine-generator to produce electricity.

3.4.1 Steam Cycle

Safety Aspects

Experience with licensing high-temperature gas-cooled, steam-cycle plants in the United States is second only to experience with light-water reactors. Two plants have been licensed, built, and operated in the United States: a 40-MW prototype at Peach Bottom, Pennsylvania, and a 330-MW demonstration plant at Fort St. Vrain, Colorado. The safety analysis reports for two commercial-size plants (Fulton Generating Station and Summit Power Station) had been reviewed by the Nuclear Regulatory Commission and the Advisory Committee on Reactor Safeguards, and limited work authorization permits had been issued before the plants were canceled in 1975. A design for a standard commercial plant was prepared by the General Atomic Company and was reviewed by the Nuclear Regulatory Commission during the period 1974 to 1977 (Reference 8). A later commercial design was being prepared by a utility group, Gas-Cooled Reactor Associates, in 1978 (Reference 9). This design has not yet been submitted to the Nuclear Regulatory Commission for review, although aspects of it have been discussed.

As noted previously, the current Nuclear Regulatory Commission regulatory requirements were developed with principal focus on the light-water reactor, and the great majority of the criteria and guides were developed for pressurized- and boiling-water reactor types. The establishment of high-temperature gas-cooled reactor licensing criteria has proceeded on the basis of direct adoption or suitable adaptation of light-water reactor criteria, which has been possible in most instances. Specialized high-temperature gas-cooled reactor criteria and guides have been developed as necessary. Considerable work is needed to further develop the specialized criteria and guides. As with the heavy-water reactor, it is useful to recognize some intrinsic safety features of the high-temperature gas-cooled reactor designs, as follows:

- o The negative power and temperature coefficient of the reactor due to the nuclear characteristics of the fuel tend to limit inadvertent power and temperature increases in the fuel.
- o The high heat capacity and relatively low power density of the reactor core caused by the large mass of graphite results in slower responses to transients and allows more time to take action to provide emergency cooling or other actions.
- o The lack of reactivity effects of the helium coolant eliminates nuclear effects caused by changes in coolant density.
- o The high-temperature stability of the ceramic-type fuel and the graphite moderator permit much higher core temperatures without changes in fuel or core geometry occurring.
- o The chemical inertness of the helium in the primary coolant circuit reduces chemical corrosion problems and the formation of radioactive debris or decomposition products.
- o The high integrity of the primary coolant circuit caused by the integration of the circuit into one prestressed concrete reactor vessel reduces concerns for major, sudden pipe or vessel ruptures.

The refractory nature of the core and the properties of the helium coolant result in a substantial delay in fission product release from the fuel in comparison to light-water reactors in the event of a loss of coolant accident. The safety-related issues for the high-temperature gas-cooled reactor, steam-cycle concept (Reference 8) are reflected in a list of 29 questions and comments covering 8 topics, submitted to the Department of Energy by the Nuclear Regulatory Commission in 1978. These questions and comments concern the current safety-related issues pertinent to licensing review of a commercial high-temperature gas-cooled reactor; they are not complete or definitive statements of anticipated licensing needs. The topics are as follows:

- o Graphite as structural material. Identification of mechanical design requirements, justification of design criteria for graphite structures, and identification of parameters significant to graphite corrosion.

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- o Core seismic response. Provision and justification of seismic design criteria for the core and other nonmetallic structures relating to integrity of the core; description of seismic analysis methods for the core and related structures; and description of the function of any nongraphite materials in the reactor, in terms of core seismic response.
- o Fuel transient response. Description of conditions to which the fuel is exposed; description of response to the same conditions of materials, other than fuel, affecting fuel integrity; description and design aspects of reference fuel; provisions of fuel irradiation data; and description of the basis for predicting fuel response to postulated accidents and transients.
- o Inservice inspection and testing. Provision of criteria for determining the need for inservice inspection, and descriptions of plans for development of techniques and instrumentation to meet the intent of the Code of Federal Regulations Chapter 10, Part 50.55, a(g).
- o Low-probability accidents. Description of the best estimate and uncertainty determinations of consequences of selected low-probability accidents; identification of research programs, in progress or planned, that relate to critical assumptions made in the accident study; relation of experience factored in the study of low-probability accidents; identification of nonprobabilistic criteria used to distinguish between design-basis accidents and accidents sufficiently remote that they can be excluded from the design basis; and decisions concerning whether or not the maximum hypothetical fission product release which implies loss of integrity of both prestressed concrete reactor vessel liner cooling and secondary containment, is an appropriate factor in siting considerations.
- o Containment requirements. Determination of criteria for the selection of design bases for containment and/or the description of the impact on containment from combustion of carbon monoxide originating from the oxide or oxycarbide fuels during an unrestrained core heatup accident.
- o Primary system integrity. Provision of thermal and mechanical design criteria for all components, structures, and systems of primary coolant system; demonstration of compliance with General Design Criteria 14 and 15; demonstration of how the design accommodates hot streaks in both upper and lower plenums; demonstration of how experience has been used in design of the asymmetric reactor vessel used for the lead plant

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design; description of protection for the reactor vessel penetration and closures against sudden depressurization; inspectability of the primary system pressure boundary; and determination of the bases for acoustic excitation in the primary system.

- o Emergency core-cooling system provisions. Provision of performance criteria for the system and discussion of the relationship of containment backpressure and loop-isolation valves to expected performance.

In addition to the eight topics discussed above, which illustrate licensing concerns about the high-temperature gas-cooled reactor, steam-cycle concept, some core design problems were encountered at the Fort St. Vrain Nuclear Power Station. Power/temperature oscillations were observed during power ascension. The fluctuations, attributed to core movements and coolant flow redistribution, have caused the Nuclear Regulatory Commission to limit power operations at the plant to less than 70 percent of rated thermal power. A test of fuel block restraining mechanisms intended to eliminate the oscillations, is scheduled for 1979 or 1980 in the Fort St. Vrain reactor.

The reference fuel for the high-temperature gas-cooled reactor is oxycarbide-coated particles, which are bonded into carbonized fuel pellets and loaded into graphite fuel blocks. The current fuel for the Fort St. Vrain reactor is highly enriched uranium mixed with thorium. Test elements of medium-enriched uranium (about 20 percent U-235) are being fabricated for irradiation in this reactor. The other fuel variants in Table 1 do not present new safety issues, except for the problem of high gamma-ray activity in the fresh fuels, which contain U-233 or which are spiked with radioactivity. These fuels will have to be shielded during shipping, transfer, and storage operations in a manner similar to spent fuel.

Environmental Aspects

The environmental effects of the high-temperature gas-cooled reactor steam-cycle, exclusive of the fuel-cycle facilities, are not significantly different from those of the light-water reactor, except for water consumption and thermal discharge from the plant.

Water consumption and thermal effluent are about 29 percent less than those from the reference light-water reactor because the high-temperature gas-cooled reactor plant has a higher thermal efficiency (39 percent) than the light-water reactor (about 33 percent). This factor may be important in some areas

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where water use is critical. The chemical effluents are similar in kind and quantity to those from the light-water reactor. Normal-operation radiological releases are similar to those from the light-water reactor, but specific differences occur in the relative amount of various isotopes released. The Fort St. Vrain reactor operating experience over the last four years has shown significantly lower occupational exposures than light-water reactors.

Environmental impacts of the high-temperature gas-cooled reactor steam-cycle are thus either similar to or less than those from the reference light-water reactor, and there should be no impediment to licensing because of routine releases.

3.4.2 Direct Cycle

Safety Aspects

The high-temperature gas-cooled, direct-cycle reactor is designed to accommodate the same basic core as the steam-cycle plant. The primary differences in core design and performance characteristics are related to the higher temperatures of the helium coolant entering and leaving the core. The direct cycle concept shares several important features with the steam-cycle, such as the prestressed concrete reactor vessel, the graphite fuel blocks containing encapsulated fuel particles, and the use of three independent auxiliary cooling loops. Safety features such as the control rod reserve shutdown system and liner cooling system for the prestressed concrete reactor vessel are mostly identical. The major differences in the reactor components are the absence of steam generators and associated equipment and the presence of helium power turbines, compressors, helium heat exchangers, and helium-to-water coolers. The additional components are incorporated inside the prestressed concrete reactor vessel. The shafts from helium power turbines extend through the prestressed concrete reactor vessel to drive the electric generators, which are located inside the containment vessel.

The direct-cycle concept is at an earlier stage in terms of design and licensing, but it appears to be licensable. Many anticipated safety-related issues that will arise will be the same as those for the steam cycle concept; however, some will be unique and will require extensive analysis, testing, and demonstration.

A Preliminary Safety Information Document for the direct-cycle reactor was submitted by the General Atomic Company in July 1975 (Reference 10). A

question-response interaction between the Nuclear Regulatory Commission and General Atomic followed but was terminated because of funding and manpower limitations.

The safety-related concerns listed for the steam cycle apply for the direct cycle concept, but some of the topics are not as well investigated as those for the steam-cycle. The direct-cycle concept has a number of features that lead to some new safety and licensing questions. The most significant of these are discussed below.

- o Shaft seal failure. The turbomachine/generator shaft penetrates the primary coolant system boundary. Failure of the seal as a result of machine or shaft malfunction can cause rapid depressurization of the containment. Design features must be incorporated to ensure that such accidents have an acceptably low probability of occurrence.
- o Internal pressure equilibrium accidents. Failures of internal components such as the turbomachines or recuperators can cause rapid pressure equilibration within the reactor vessel. These pressure pulses/transients could be more severe than those associated with the most rapid postulated reactor vessel depressurization for the high-temperature gas-cooled, steam-cycle reactor. The gas turbine pressure equilibration accidents place stringent design requirements on reactor-vessel internals and may dictate component designs different from those of the high-temperature gas-cooled, steam-cycle reactor. Pressure equilibration accidents require that the failure phenomena be defined, modeled, and verified. This action, in turn, depends on experimental data related to failure, as well as experimental or other data that verify the model. The model would include a computer code that describes the transient behavior of the compressible fluid flow following the accident.
- o Turbomachine failures. In addition to causing rapid pressure transients, these failures can create missiles; protection must be provided to safety systems components, including the reactor vessel and the secondary containment. The potential for internal missiles generated by gas turbine or compressor failures is larger for the direct-cycle concept than for the steam-cycle concept.
- o Turbine intercept and bypass valves. In the event of loss of generator electrical load or loss of compressor load, the helium flowing to the

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power turbine must be very rapidly interrupted and diverted to prevent turbine overspeeding.

- o Acoustical vibrations from turbomachinery. The very high power levels in the power turbine and compressor combined with the properties of the helium coolant cause a potential for large acoustical vibrations to occur in the fuel elements and the components in the cooling circuit.

The direct cycle control system, protection systems, and engineered safety factors differ significantly from those of the steam cycle, but it is expected that the increased simplicity of the direct cycle systems will enhance overall safety. The research, development, and demonstration program will nevertheless have to explore the response of the turbomachines and compressor system to mechanical failures of seals, valves, and controls. The adequacy of the containment envelope and penetrations will have to be demonstrated. Maintainability of the turbine within acceptable occupational exposure limits will have to be demonstrated. Fuels for the direct cycle will have to be qualified at Fort St. Vrain or in a comparable reactor facility.

The introduction of major new components within the primary system of the direct-cycle reactor will require safety classification of these components. Classification will require analysis of consequences from postulated failures of these components.

Environmental Aspects

The environmental effects of the high-temperature gas-cooled, direct-cycle reactor have not yet been evaluated in detail. However, general assessments indicate that the radiological and chemical effects from normal power operation will be similar to or less than those of the high-temperature gas-cooled steam-cycle reactor and the light-water reactor. A potential advantage of the direct cycle, compared with light-water reactor, is that the use of dry cooling towers may be more economical and encourage their use, almost eliminating water consumption at the reactor. The use of dry cooling is more attractive for the direct cycle than for plants using the steam cycle because the higher temperature of the heat rejected from the direct cycle facilitates efficient transfer of heat to the atmosphere. If evaporative cooling towers are used with the direct cycle, water consumption is reduced to about one-half to two-thirds of that needed for a light-water reactor with the same electrical output, depending on whether or not a bottoming cycle to recover power from the rejected heat is used.

3.5 LIQUID-METAL FAST-BREEDER REACTORS

The two liquid-metal fast-breeder reactor concepts studied by NASAP are:

- o The loop-type design in which the sodium primary coolant system and the fuel-handling mechanisms are external to the reactor vessel and are connected by large-diameter piping.
- o The pool-type design in which all the sodium primary coolant system and fuel-handling system are contained within the reactor vessel together with the core. These components include the reactor core, the pumps, the intermediate heat exchangers, the connecting piping system, and the fuel handling equipment.

The fuel types considered included oxide fuels and carbide fuels. Variations of fissile and fertile materials which were considered included:

- o Uranium-plutonium cores with uranium blankets.
- o Uranium-plutonium cores with thorium blankets.
- o Plutonium-thorium cores with thorium blankets.

Two spatial arrangements of the core and blanket were studied: (1) a homogeneous core arrangement, in which the blanket assemblies surround the core; and (2) a heterogeneous core arrangement, in which some blanket assemblies are included within the core.

At the present time the Department of Energy is undertaking conceptual design studies aimed at selection of a reference design for further development and demonstration. These studies include variations in plant arrangement, core arrangement, and fuel similar to those described above.

3.5.1 Safety Aspects

Loop-Type Designs

The loop-type 200-MW Enrico Fermi Power Plant, Unit I, near Detroit, Michigan, was licensed and operated between 1963 and 1966. It was shut down in 1966 after a coolant flow blockage accident. After cleanup, inspection, and addition of improved instrumentation to the reactor, the plant was relicensed, but because of the high projected cost of impending safety changes and for programmatic reasons, it was permanently shut down in May 1972.

The loop-type Fast Flux Test Facility, rated at 400 MW (thermal), is under construction and is scheduled to begin operation in 1980. Because the Department of Energy operates this reactor, Nuclear Regulatory Commission licensing was not required, but both the Nuclear Regulatory Commission and the Advisory Committee for Reactor Safeguards reviewed the reactor. The Advisory Committee on Reactor Safeguards concluded that if due regard is given to the Nuclear Regulatory Commission recommendations concerning weld inspections, natural circulation, mitigation of possible consequences of certain low probability accidents, and other matters, the startup and operation of the Fast Flux Test Facility is acceptable. The Department of Energy is presently evaluating the Nuclear Regulatory Commission staff recommendations regarding containment adequacy for low probability accidents.

The Clinch River Breeder Reactor Plant, a loop-type demonstration plant rated at 350 MW, was undergoing licensing review by the Nuclear Regulatory Commission and Advisory Committee for Reactor Safeguards for siting at Oak Ridge, Tennessee, but the review was suspended in 1977 as a result of changes in U.S. Government policies. Although several licensing issues remain unsolved, licensing was expected with possible modifications in the design (Reference 11).

The two safety aspects of the liquid-metal fast breeder reactor that have drawn substantial attention are the potential for the core to be driven into a more critical geometrical arrangement and the presence of large quantities of sodium. Thus, considerable emphasis has been placed on core-disruptive accidents (CDA's) and other accidents that could lead to melting of the core. A core-disruptive accident is an accident that destroys the core geometry, thereby possibly affecting ability to cool or change the nuclear reactivity. To date, the licensing evaluation of the Clinch River Breeder Reactor indicates that the state of technology and experience would result in a safe design, but the particular design proposed was not satisfactory for some

low-probability accidents. The major concerns are under study in the United States and other countries. The outlook is favorable for satisfactory resolution of the concerns by completion of current research, development, and demonstration programs and certain design modifications to the plant design.

On the other hand, a favorable safety-related characteristic of the liquid-metal fast breeder reactor is the low pressure in the primary coolant system which reduces the concerns about the whole class of depressurization accidents that must be handled with high-pressure systems such as the light-water reactor or the high-temperature, gas-cooled reactor. Licensing reviews to date have been based on oxide-fueled systems. Carbide-fueled systems are now considered as alternatives in Department of Energy designs, and the systems introduce significant variations in safety considerations. The following discussion applies to all the fuel alternatives in an oxide or carbide form, except as discussed for the containment system design for core melting and core disruptive accidents. The items listed are also relevant to the pool-type design, although the evaluation of the item differs in some instances.

To assure low probability of core-disruptive accidents and core melting in liquid-metal fast-breeder reactor systems, the Nuclear Regulatory Commission staff identified, in its review of the Clinch River Breeder Reactor Project, the following design-related issues that reflect general liquid-metal fast-breeder reactor safety concerns:

- o Scram systems. The scram (rapid shutdown) systems must be shown to have sufficient redundancy and diversity that the probability of failure is acceptably small. To provide the necessary reliability, two redundant and diverse scram systems are required. Each system has to be capable of shutting the reactor down under extreme conditions. No electrical or other external power can be required for the scram of any control rod. The Nuclear Regulatory Commission concluded that it is feasible to design such a system and that the Clinch River Breeder Reactor Project design has the potential for satisfying the scram-system reliability criterion.
- o Residual heat removal system. The heat-transport system design must have sufficient redundancy and diversity to make the probability of its not being able to remove heat under a wide variety of shutdown conditions acceptably small. The Clinch River Breeder Reactor Project residual-heat-removal design as submitted for review had not been shown to provide the necessary redundancy and diversity, but Nuclear Regulatory Commission concluded that it is technically feasible to

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provide an adequate residual-heat-removal system for liquid-metal fast-breeder reactors.

- o Fuel rod and subassembly failure detection. The Nuclear Regulatory Commission concluded that the potential for fuel failure propagation beyond a single subassembly can be limited so that it will not initiate a whole-core accident and that reliable means to detect and cope with fuel-rod failure and subassembly faults must be provided. Substantial analytical and experimental evidence indicates that local faults affecting a single rod or a few rods within a subassembly will not rapidly propagate to adjacent rods. The current fuel subassembly inlet designs, having multiple inlet ports at different planes with interposed strainers, should prevent the occurrence of blockages that could significantly reduce flow to a subassembly. This prevention has been demonstrated experimentally. Increased operating experience with failed fuel rods in the research and development program will enable designers to quantify the tolerance limits for failed fuel and to set acceptable limits on failure propagation potential.

- o Primary coolant system integrity. A continuing high integrity of the heat-transport system must be provided. Because operating temperature is well below that at which sodium will boil at near atmospheric pressure, the sodium in the primary system is at a low operating pressure. Typical peak pressures do not exceed about 200 psi. Thus, there is no stored energy for flashing to vapor in the event of a pipe break. Proper layout of the piping, including guard pipes around coolant pipes in some areas and the inclusion of check valves in the cold legs, can prevent the core from being uncovered in the unlikely event that a leak in the system occurs. Provided that the design is suitable, that proper leak-detection equipment is installed, and that an in-service inspection program is carried out, the heat-transport system can be designed for a high level of integrity and for continued assurance of this integrity throughout the operating history of the plant. To this end, the Nuclear Regulatory Commission concluded that, for the Clinch River Breeder Reactor Project, a double-ended rupture of the primary-system cold-leg piping need not be considered as a design-basis accident. Because of the higher operating temperatures, it was determined that hot-leg ruptures should be considered as design-basis events for the Clinch River Breeder Reactor Project but that containment design features should be included to cope with the consequences of the resulting large sodium releases.

The above issues, when favorably resolved, ensure that the probability of core melt and core-disruptive accidents is minimized. The Nuclear Regulatory

Commission identified three more issues for minimizing the probability of containment failure in the event of a more severe accident.

- o Containment system design for large sodium spillages. The containment must be able to accommodate the consequences of spillage of large quantities of sodium from the primary or intermediate coolant system. At the operating temperatures of these reactors, sodium will ignite and burn readily if sprayed into the air, even in reduced-oxygen atmospheres. It will also burn as a pool. The heat released from a sodium fire can damage concrete, and water released from the heated concrete reacts exothermally with sodium. The net effect of these reactions is to increase containment-cell temperatures and pressures, with structural degradation of the concrete and the production of potentially explosive hydrogen. However, because of the substantial experience in handling sodium, the Nuclear Regulatory Commission stated that it is possible to design features in the containment system to alleviate the sodium hazards. Sodium fires are investigated as part of the component research and development program.

- o Containment/confinement system design for radioactivity releases. The containment/confinement system must be capable of adequately mitigating the radioactivity releases that could result from all events within the containment design basis.

The general safety design criteria issued by the Nuclear Regulatory Commission state that the reactor containment structure, including access openings and penetrations (if necessary, in conjunction with additional post-accident heat removal systems) shall be so designed that it and its internal compartments can accommodate, without exceeding the design leakage rate, the calculated pressure and temperature conditions resulting from normal operation, anticipated operational occurrences, and any of the postulated accidents. In a liquid-metal fast-breeder reactor, the accidents that represent the greatest potential for damaging the containment are sodium fires, coupled with potential sodium-concrete reactions resulting from failure and subsequent release of sodium from the primary heat-transport-system equipment. The design concept best suited to prevent these accidents is a containment-confinement system, in which the steel containment building is surrounded by a thick concrete confinement shell, with the annulus between the two maintained at a reduced pressure. An annulus filter system could be added to reduce radioactivity release from the annulus to the environment while maintaining a reduced pressure in the annulus. After examining the Clinch River Breeder Reactor Project containment/confinement system design and the range of conditions to

which it might be subjected, the Nuclear Regulatory Commission concluded that it is technically feasible to implement design features to meet their stated criteria.

- o Containment system design for core melting and core-disruptive accidents. The containment system should be so designed that it could maintain its integrity under a broad range of conditions involving the energetic disassembly of the core, production of vaporized fuel, and other possible consequences resulting from core melting accidents. The initial position of the Nuclear Regulatory Commission was that the containment integrity must be maintained for 24 hours. They have agreed to reconsider the 24-hour criterion when the licensing review is reinitiated. The Nuclear Regulatory Commission concluded that the Clinch River Breeder Reactor Project must be designed to preclude early containment-system failure from core melting and core-disruptive accidents. The Clinch River Breeder Reactor Project should achieve a level of safety at least comparable to that of present-generation light-water reactor plants. Studies of a spectrum of events beyond the design-basis accident reveal that some such accidents (e.g., loss-of-flow accident with scram-system failure) have a high probability of leading to large-scale fuel melting in the core and, possibly, to the generation of significant quantities of fuel vapor. Study of these events beyond the design basis and the challenges to containment integrity that might result from their occurrence must include two generic types of consequences: excessive fission-energy release during the transient (energetics) and failure to adequately cool the core and to accommodate the molten-core debris resulting from the transient.

For those events beyond the design basis, in which the control system is assumed to operate and the core melts because of lack of adequate residual-heat-removal capability, post-accident heat removal is the primary concern. This aspect was discussed under the preceding discussion of the residual heat removal system. For the class of so-called unprotected transients where scram-system failure is assumed to occur, both energetics and post-accident heat removal issues must be examined. Three issues are associated with unprotected accidents in the liquid-metal fast-breeder:

1. The positive sodium void worth associated with medium- to large-size liquid-metal fast-breeder reactors.
2. Vapor explosions occurring when molten fuel and coolant interact.

3. Recriticality events in the disrupting cores.

For reactors the size of the Clinch River Breeder Reactor Project, the positive sodium void worth puts the system in super-critical state and at 5 to 20 times nominal power, when the fuel pins begin to disrupt in the unprotected loss-of-flow accident. The subsequent scenario is quite sensitive to the initial motion of this disrupting fuel. A strong potential exists for this early motion to be dispersive, which may dispel the recriticality concern. However, the Nuclear Regulatory Commission is not yet convinced that sufficient fuel would be dispersed to shut down the core during this phase of the accident. A substantial research and development program is currently aimed at demonstrating the existence of this early fuel-dispersal mechanism. For liquid-metal fast-breeder reactors with total void worths on the order of \$3,* successful demonstration of early fuel dispersal is highly likely. However, more stringent requirements for rapid early fuel dispersal would exist in large liquid-metal fast-breeder reactors that have homogeneous cores. These larger liquid-metal fast-breeder reactor cores have been designed with larger length-to-diameter ratios for economic reasons, and the total void worth is predicted to be in the range of \$5 to \$6. This fact has caused attention to be focused on core-design alternatives in which the void worth would be reduced to the range of \$2.5 to \$3.5. The reduction can be achieved by the so-called heterogeneous core designs, in which blanket subassemblies are placed in the core regions.

For oxide-fueled systems, the earlier concern about vapor explosions from fuel-coolant interactions that might offer an energetics threat appears to be unwarranted. Research carried out over the past 6 to 7 years showed these energetics events to be unlikely, on the basis that the interface contact temperature between the fuel and the coolant is well below the spontaneous nucleation limit for sodium. Thus, the

*The dollar is a unit of reactivity. For a given reactor design, \$1 in reactivity is equal to the delayed neutron fraction. The significance of this unit is that with an excess reactivity of \$1, the reactor will be critical on prompt neutrons, and the time scale of events for positive reactivity insertions greater than \$1 will be determined by the prompt neutron lifetime, implying a need for unachievably rapid control system actions. Thus, core designs with \$1 or more in reactivity must have inherent shutdown characteristics to accommodate the range of transients. For reactivity insertions of less than \$1, the time scale of events is controlled by the delayed neutrons, which have much longer lifetimes.

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Nuclear Regulatory Commission, in its evaluation of Clinch River Breeder Reactor Project safety issues, gave them small concern. Again, for the reference oxide-fuel system, the energetics potential, once associated with recriticalities after the initial core disruption, has been shown to be small. The arguments used to reduce concern about these events are based on the dispersive effect provided by vaporization of the steel cladding, which precludes energetic recriticalities. Additional confirmatory work is being done on both the fuel and coolant interactions and recriticality issues to provide further support for these arguments.

Whether or not significant energetics result from these severe accidents, large amounts of molten core material will be produced. The post-accident heat removal considerations associated with demonstrating that this debris can be contained, either within the reactor vessel or outside it, and without causing early containment failure, have received considerable attention. For the smaller Fast Flux Test Facility Reactor core, containment of the post-accident debris within the vessel is highly probable. For larger plants, such as the Clinch River Breeder Reactor Project, vessel containment does not appear to be possible, and efforts have been made to demonstrate that the debris can be accommodated in the reactor cavity below the vessel without threatening containment integrity. Although the Nuclear Regulatory Commission staff was not convinced that the proposed Clinch River Breeder Reactor Project design could accommodate the debris in the reactor cavity for 24 hours with containment intact, they did believe that the technology exists to meet a 24-hour-no-failure criterion. However, the 24-hour criterion will be re-examined when the licensing review is reinitiated. A substantial research and development program is in progress to further develop and refine post-accident heat removal technology for large liquid-metal fast-breeder reactors.

Carbide-Fueled Systems--Resolution of most of the design-related issues discussed above is insensitive to whether the system is carbide-fueled or oxide-fueled. However, the containment system design for core melting and core-disruptive accidents is sensitive to fuel type. The likelihood of strong public concerns about liquid-metal fast-breeder reactor accidents will mean that, for licensing carbide fuels, the Nuclear Regulatory Commission will probably require extensive experimental evidence that these fuels are as safe as oxide fuels. To date, most experience from foreign programs and from U.S. reactor operation and safety programs has focused on the oxide alternative.

With respect to the positive sodium void worth associated with medium-to-large-size liquid-metal fast-breeder reactors, the loss-of-flow accident

scenario is sensitive to the initial motion of the disrupting fuel. Carbide fuels also exhibit strong potential for this early motion to be dispersive. As in an oxide-fueled system, carbide fuel has a lower melting point than oxide fuel but operates at lower temperatures so that in an accident the two fuels would become molten and mobile at about the same time. In addition, carbide fuel retains more fission gas, thus increasing its fission-gas-induced dispersive potential relative to oxide fuel.

Regarding the issue of the energetics of molten fuel and coolant interactions, the higher thermal conductivity of carbide fuel, compared to oxide fuel, makes it difficult to rule out energetic fuel-coolant interactions that could result in sodium vapor explosions. Experiments have shown the increased potential for energetic fuel coolant interactions for carbide fuels in connection with the problem of sodium voiding.

Carbide fuels also warrant additional concern relative to oxide fuels in the energetics of recriticality. A molten mass of carbide fuel and stainless steel is inherently less dispersive, because the carbide melting point is some 400°C below the steel boiling point; hence the fuel would have to be heated more before steel vapor production could become a dispersive force. Meanwhile, a compaction of this molten mass would be possible. A more significant recriticality concern in the carbide system is that of a pressure-driven recompaction.

If a carbide core with high thermal conductivity undergoes a mild disassembly, then a pressure-driven recompaction induced by a fuel coolant interaction may occur, on a scale much less than the whole core. Such a condition must be considered, along with the potential barriers to its occurrence.

In summary, the accident-energetics-potential question is not as easily resolved for the carbide system as it is for the oxide system. Two reasons are that, in at least two areas, fundamental physical principles do not strongly support the carbide case; and that much less experimental work has been done on carbide fuels. These problems would have to be further investigated in the safety research, development, and demonstration program, to prove that carbide-fueled designs are as safe as oxide-fueled designs. Answers to the post-accident heat removal question are not substantially different for the two fuel types. To confirm this, much research on carbide fuel is necessary, but the Nuclear Regulatory Commission believes that the technology exists to meet a criterion of 24 hours with no containment failure, if this criterion were required.

Pool-Type Designs

The foregoing discussion applies specifically to loop-type designs such as Clinch River Breeder Reactor Project and the Fast Flux Test Facility although all the items must also be considered for pool-type designs.

The pool-type 62.5-MW (thermal) Experimental Breeder Reactor-2, at Arco, Idaho, although not licensed by the Nuclear Regulatory Commission, has operated successfully for the last 12 years. Pool-type liquid-metal fast-breeder reactors are being developed in several foreign countries. Demonstration reactors of about 250 MW have operated successfully in the United Kingdom and France for several years. The 1200 MW Superphenix is under construction in France and is scheduled for operation in 1983. A pool-type design has not been reviewed by the Nuclear Regulatory Commission as part of a construction application, but the general criteria and issues which were defined for loop-type designs are also believed to be applicable to pool-type designs.

In December 1974, a joint study between Bechtel and Societe Technique pour l'Energie Atomique (Technicatome) was initiated to evaluate the commercial application of liquid-metal fast-breeder reactor power plants, based on the Superphenix nuclear steam supply system concept and U.S. standards and licensing criteria. This study indicated that, although current U.S. licensing criteria are not completely defined in some important areas, the Superphenix design appears to be consistent with safety and licensing criteria that have been applied to liquid-metal fast-breeder reactors in the United States. No items were identified that were expected to impose significant additional licensing requirements upon total plant design. However, it appears that detailed analytical verification as well as research and development support will be required to satisfy regulatory authorities about the adequacy of the safety-related features of the Superphenix design.

Several generic characteristics of pool reactors may help eliminate, mitigate, or contain both likely and very unlikely accident events (Reference 12). Among these potentially inherent safety characteristics are these:

- o A large sodium inventory to act as a heat sink and protect system components from thermal shocks.
- o A large main tank at relatively low operating temperatures, with no piping penetrations beneath the sodium surface.

- o "Cold-leg" pumps with short piping runs to the reactor inlet plenum submersed in the cold inlet sodium pool.
- o A large cross-sectional area of the main tank as well as a large sodium inventory to help dissipate and absorb the energy release and after-effects of a core-disruptive accident and to provide the potential for debris retention and post-accident heat removal capability.

Preliminary containment calculations using somewhat conservative assumptions indicate that the pool concept, because of the above generic characteristics, can accommodate significant energy releases from a core-disassembly accident. The study gives reasonable indication that the main-tank, deck, and plug structures can be designed to survive the given core-disruptive accident loads without failure of the main tank or significant breach of the shield deck.

Both the cold-pool and hot-center-pool systems can accommodate loss-of-piping-integrity accidents without losing primary-system coolant. The probability exists, based on steady-state scoping calculations (which should be the subject of further study), that both systems can provide sufficient core flow to avoid boiling. Further, with simultaneous reactor and pump trip at pipe rupture, longer pump coast-down times could be incorporated to aid initial decay-heat removal and institute natural-convection cooling.

Designing the system to provide adequate natural-convection cooling upon loss of all pumping power coincident with reactor scram appears feasible for intact geometries. Preliminary calculations have shown that both the cold-pool and hot-center-pool designs can provide adequate cooling margins.

Preliminary assessment of the current pool designs indicates that in-tank cooling should occur for a significant quantity of core debris following a core-disruptive accident. It is not clear at this time that the entire core inventory is coolable without the presence of an engineered core-retention device. To a significant degree, the fraction of the core debris that is coolable by inherent processes is likely to be a function of the radius of curvature of the vessel bottom.

3.5.2 Environmental Aspects

The liquid-metal fast-breeder reactor has less thermal impact on the environment than the reference light-water reactor, because the liquid-metal fast-breeder reactor has a higher thermal efficiency (37 percent) and rejects 19 percent less heat for a given amount of electricity generated. Chemicals releases are similar in kind and quantity to those from the reference light-water reactor; therefore, the effects are similar.

The estimated releases of radioactivity from the liquid-metal fast-breeder reactor are potentially much smaller than those for the reference reactor because of several factors. The radioactive gas treatment system is designed to retain the noble gases rather than using the controlled-release method employed by the light-water reactors. Most of the tritium generated in the liquid-metal fast-breeder reactor is removed in the primary and secondary cold traps of the sodium purification system, rather than being retained in the primary coolant water and being released periodically, as in the light-water reactors. However, the leakage of the radioactive sodium coolant in the fast-breeder reactor must be held to acceptably low levels; current designs so provide. Occupational exposures at the reactor can be comparable to those of the light-water reactor, if adequate shielding, safety interlocks, and administrative controls are applied to prevent exposures due to the high gamma-ray and neutron radiation in the recycle fuel.

The effects of fuel variations on the radioactive source term are not expected to alter the basic conclusion that radiation releases are potentially smaller than with the light-water reactor, at least for uranium/plutonium oxides. There is virtually no experience on thorium/plutonium oxide fuels for drawing any meaningful conclusions. Similarly, experience with carbide-type fuel is limited; preliminary observations show that carbide fuel does not release gaseous fission products as readily as oxide fuels.

The choice of fuel for the liquid-metal fast-breeder reactor is expected to be based primarily on safety and operational characteristics rather than environmental considerations.

3.6 GAS-COOLED FAST REACTORS

3.6.1 Safety Analysis

No gas-cooled fast reactor has been built. Research and development work is underway in the United States and, until recently, in Europe. Conceptual designs for the gas-cooled fast reactor prepared in the United States have many components which are similar to those of the high-temperature, gas-cooled reactor and the fuel design is similar to that of the oxide fuel being developed for the liquid-metal, fast-breeder reactor. Thus, although the concept has not been built, the industrial developers of the gas-cooled fast reactor concept plan to proceed initially to a large demonstration plant generating electric power.

A gas-cooled fast-reactor concept, based on a 300-MW demonstration plant conceptual design, was reviewed by the U.S. Atomic Energy Commission and the Advisory Committee on Reactor Safeguards. A Preliminary Safety Information Document for the 300 MW demonstration plant was submitted to the Atomic Energy Commission by the Gulf General Atomic Company in 1971 (Reference 13); and the Commission issued a Preapplication Safety Evaluation Report in 1974 (Reference 14), which concluded that the plant was fundamentally licensable but that a number of key problems remain to be resolved. Subsequent to that review, the Nuclear Regulatory Commission has provided informal reviews of the design. The conceptual design is currently undergoing some extensive changes in order to promote resolution of the problems identified.

The major (unresolved) safety concerns identified in the Atomic Energy Commission safety evaluation of the gas-cooled fast reactor are described below.

- o Acceptable power density levels and thermal margins. Fuel heat transfer rates are substantially higher for the gas-cooled fast reactor than for the high-temperature gas-cooled reactor and approach those for the liquid-metal fast-breeder reactor, as is evident from comparison of the core power densities and the fuel heat fluxes. These higher heat transfer rates are the result of higher system pressure, higher mass flow and velocities, and improved thermal conductance, compared to the high-temperature gas-cooled reactor. Thermal conductance is enhanced by roughening of the fuel metal cladding. In view of these factors and the potential for more rapid temperature changes due to the high heat transfer rates, the proposed design limit of $1,370^{\circ}\text{C}$ to prevent cladding damage (this limit is $1,200^{\circ}\text{C}$ for similar stainless steel cladding used in light-water reactors) was considered unacceptable. Also, the

analysis of the peak cladding temperature during the design-basis depressurization accident did not consider air ingress; the Commission required that this effect be considered, because the ingress of air and its mixing with helium coolant would reduce the core-cooling capability of the circulators. The response of the gas-cooled fast reactor core to transients will require extensive out-of-pile testing to ensure that all anticipated accidents can be mitigated and that fuel design limits are not exceeded.

- o Definition of depressurization accidents. The proposed design-basis accident is a depressurization accident that occurs with the failure of the seal for the central cavity closure, which is the largest penetration closure for the prestressed concrete reactor vessel. A flow restrictor device limits the effective depressurization area to 25 square inches. Both the definition of the depressurization accident by mechanistic means and the size of the depressurization area used to compute the depressurization rate and forces were judged to be unacceptable. The Commission decided that the limited experience with high-pressure prestressed-concrete reactor vessels does not allow identification of a controlling mechanism without first addressing a spectrum of other conceptually possible failure modes, and also decided that the design should provide a suitable margin against undefined failure modes.

- o Definition of core-disruptive accidents. Postulated accidents whose results are severe enough to lead to large-scale melting of the core and explosive disassembly were not addressed in the Preliminary Safety Information Document for the gas-cooled fast reactor. It was recommended that three such accidents be considered: depressurization with failure to scram, loss of coolant circulation with subsequent scram, and transient overpower without scram. The Commission believed that it would be difficult to demonstrate systems reliability of sufficiently high degree to be able to dismiss large-scale fuel melting. The research, development, and demonstration program will have to determine the impacts of such events on the integrity of the containment.

- o Diversity in reactor shut-down system. A principal concern of both the Atomic Energy Commission and the Advisory Committee for Reactor Safeguards was that the concept of shutdown rods may not offer an emergency reactivity control system that is sufficiently diverse from the normal control system. In their initial evaluation of anticipated transients without scram, the Commission concluded that acceptable diversity and separation have been provided through two control rod drive systems of different design, and this has been acceptable for

light-water reactors. However, they did not consider this approach acceptable for the gas-cooled fast reactor at this stage of its development. The development program will have to test alternatives to show independence from accidents that might fail the normal shutdown rods and high reliability of functioning under all other accident conditions.

- o Adequacy of core cooling. Several of the design bases and features of the emergency core-cooling system were found to be unacceptable. In particular, the depressurization accident used as a design basis for the system was not acceptable, and the design did not recognize the occurrence of a core-disruptive accident. The Commission suggested that the feasibility of "hardening" certain portions of the system, so that they remain functional following a core-disruptive accident, should be explored in conjunction with the design of a post-accident heat removal system. The system's diversity was also questioned in the event that a post-accident heat removal system was not included as an engineered safety feature. The use of gas circulators within the containment as the only means to provide reactor cooling was not considered to offer sufficient diversity to assure protection against a simultaneous failure mode common to the main and auxiliary systems.

- o Containment-system design. Because of possible plutonium aerosol retention following core-melting or core-disruptive accidents, the containment-system design was found unacceptable and will probably have to be redesigned to handle the plutonium released from a core-melting or core-disruptive accident. The containment was considered to be adequate for the design-basis depressurization accident involving a leak area of about 50 square inches.

- o Prestressed-concrete reactor vessel response to core-disruptive accidents. Although the reactor vessel is an extremely massive and strong structure with inherent capability to withstand large dynamic loadings such as might be imposed by an explosive core-disruptive accident, analytic work will be needed to quantify the dynamic loadings on the reactor vessel and determine criteria as to when transient loads can be represented by static analyses. Knowledge of the response of the prestressed-concrete reactor vessel internals to core-disruptive accidents is also important in assessing the amount of available core cooling capability following such events. Safety-related testing of model reactor vessels will probably be required as part of the research, development, and demonstration program.

- o Nuclear design--neutron steaming. One of the most significant differences between the gas-cooled fast reactor and other reactors is the level of fast neutron streaming and its potential effects on reactor safety. Fast neutrons that enter the helium coolant channels in directions closely parallel to the axis of the fuel element have a relatively large probability of streaming from the core region, in comparison to the sodium-cooled liquid-metal fast breeder reactor. This streaming effect leads to two safety concerns: the level of fast neutron exposure on the grid structure supporting the fuel, and the amount of positive reactivity change that could be caused by a streaming decrease in the event of a steam ingress accident. Monte Carlo* methods are being considered to compute the amount of neutron streaming, and critical experiments are planned to include measurements related to streaming effects. The extent of radiation damage to structural materials will have to be experimentally demonstrated as acceptable. A recent design change uses a bottom-supported grid structure, and this change would have to be analyzed.

- o Inservice inspection. The Preliminary Safety Information Document did not adequately address inservice inspection, and this issue remains unresolved. The document stated that provisions for inservice inspection would follow the high-temperature gas-cooled reactor precedent. However, the Atomic Energy Commission never approved the proposed inservice inspection program for high-temperature gas-cooled reactors. In particular, the problem of inspectability of the prestressed-concrete reactor vessel liner has not been resolved for the high-temperature gas-cooled reactor, and consequently, it remains an unresolved item also for the gas-cooled fast reactor. In addition, differences between these two designs affect accessibilities to the reactor vessel cavities, along with differences in shielding requirements and design.

The gas-cooled fast reactor design studies and research programs are making progress in resolving some of the above-mentioned safety problems, particularly in core-cooling reliability, thermal margins, and core-disruptive accidents. The outlook is favorable that these safety issues can be resolved during the course of engineering development, but resolution of the issues will entail significant changes to the previous reference design.

*Monte Carlo methods are probabilistic calculation techniques based on the use of random numbers.

Core-cooling reliability for decay-heat removal is recognized as the primary gas-cooled fast reactor safety concern. Since the earlier review, detailed reliability analyses have shown that the core cooling system, as designed, can have high reliability; they have also shown that shutdown cooling reliability is better than had been previously estimated for that design.

Recently a major design revision has been made to employ upflow cooling which allows natural convection for emergency core cooling. The gas-cooled fast reactor metal cladding thermal limit (700°C) for normal operation is in the range encountered in liquid-metal fast breeder reactor technology, and the same reference cladding material has been chosen. Thus, many of the liquid-metal fast breeder reactor materials irradiation tests are applicable. A gas-cooled fast reactor fuel design program, including establishing thermal limits and emphasizing areas of difference between these two reactors, was formulated in 1975 and is continuing. In an amendment to the Preliminary Safety Information Document submitted in 1976 for the gas-cooled fast reactor, the design-basis depressurization accident was reexamined to consider a threefold increase in the depressurization accident flow area over the design value. The analysis showed that cladding temperature would remain below $1,260^{\circ}\text{C}$, which would provide a margin of several hundred degrees below clad melting. This is comparable with accepted light-water reactor cladding-temperature limits for loss-of-coolant accidents.

A comprehensive program has been initiated to investigate all classes of core-disruptive accidents applicable to the gas-cooled fast reactor. These accidents include complete flow blockage in a single assembly, total loss of forced circulation with reactor shutdown, loss of flow without shutdown, and continued reactivity insertion without shutdown. To date, analytical results for these accident classes are most favorable to the design, indicating only very small fuel vapor fractions and little coupling of core energy releases to the helium coolant; the mechanical work by the coolant on the vessel is well within the structural integrity limits of the prestressed concrete reactor vessel.

3.6.2 Environmental Aspects

Because the gas-cooled fast reactor has a higher thermal efficiency (37 percent) than the reference light-water reactor, it rejects 19 percent less heat for a given amount of electricity generated. Chemical releases are similar in kind and quantity to those from the reference light-water reactor.

The estimated releases of radiation (and corresponding impacts) are potentially much smaller than those for the reference light-water reactor. However, the gas-cooled fast reactor design employs a gaseous radwaste system based on retention of noble gases rather than the controlled-release method used for the light-water reactor. Occupational exposures at the reactor can be comparable to those for the light-water reactor, if adequate shielding, safety interlocks, and administrative controls are applied to prevent exposures due to high gamma-ray and neutron radiation from the recycle fuel.

3.7 LICENSING-TIME ASSESSMENT

The issues of safety and environmental impact that are likely to affect the time for deployment of commercial units for each of the alternative reactor concepts were discussed previously and were independent of other considerations such as commercialization factors. This section estimates the time required for each alternative reactor concept to reach a stage at which a definitive commercial design could be completed and at which a Preliminary Safety Analysis Report and an Environmental Report could be submitted with a commercial plant construction license application to the Nuclear Regulatory Commission for review. Estimates are included for the heavy-water reactor, the light-water breeder reactor, the steam-cycle and direct-cycle high-temperature gas-cooled reactors, the liquid-metal fast-breeder reactor, and the gas-cooled fast reactor. In addition to these reactor concepts, two alternatives are examined that involve primarily the qualification of the fuel rather than the reactor plant.

These alternatives are the light-water reactor, using a core capable of extended burnup and other modifications to reduce uranium requirements by 15 percent and the light-water prebreeder and breeder reactors, using cores that would be backfitted into an existing or new pressurized-water reactor. Time estimates for the qualification of other fuel concepts are not provided because the time to achieve commercial fuel cycle facilities is longer than the expected licensing time.

The criteria used to estimate when a reactor plant and/or core is ready for a definitive commercial design and preparation of a Preliminary Safety Analysis Report are:

- o Successful operation of full-size heat-transfer loops, critical plant components, and fuel-element assemblies in an integrated test.

- o Determination of failure mechanisms and safety margins for both critical components and fuel by completion of transient tests.
- o Demonstration of critical component reliability and materials capability.
- o Accumulation of basic physics and materials design data covering ranges of uses up to failure.

Satisfying the above criteria through basic safety research, development, and testing can be realistically performed only in a demonstration plant (full-size heat-transfer loop and fuel assemblies), in a commercially operating plant (fuel assemblies), in safety test facilities (transient tests), or in component-reliability test facilities.

Based on these criteria and the current state of development of reactor and fuel-cycle concepts, the heavy-water reactor and the high-temperature gas-cooled reactor steam-cycle plant and fuel concepts were determined to have an adequately demonstrated safety basis to proceed in preparing a definitive commercial design and a Preliminary Safety Analysis Report. (The Nuclear Regulatory Commission indicates the Canadian heavy-water reactor plant experience may not qualify to demonstrate the U.S. design for this reactor.) The direct-cycle high-temperature gas-cooled reactor, the liquid-metal fast-breeder reactor, and the gas-cooled fast reactor plant concepts require further demonstration of safety bases; the fuels for these concepts also need to be demonstrated, together with the plants. The light-water reactor extended burnup fuel and the light-water breeder reactor fuels require demonstration.

Table 3 (in Chapter 2) shows minimum time estimates in years from the present time for submittal of a Preliminary Safety Analysis Report and an Environmental Report for both demonstration and commercial stages for each reactor concept and fuel cycle examined.

Because the introduction of the fast breeders implies the introduction of a new fuel cycle with major social implications, the licensing process may be lengthened to consider the generic impacts of fuel recycle, the proliferation aspects of plutonium fuels, and the social desirability of the fission breeder as an "inexhaustible" energy source in relation to other alternatives (e.g., solar and fusion). The degree of consensus and political support for the breeder during the licensing process could strongly affect the licensing

schedule. The time assessments in Table 4 do not include these fuel-cycle licensing considerations. Explanatory discussions of the estimates follow.

3.7.1 Light-Water Reactor, 15-Percent Improved

Licensing of this fuel modification for commercial operation will require completion of successful irradiations of fuel to the peak burnups expected and completion of safety analyses and tests. These peak burnups are in the range of 60 MWd/kg for solid fuel pellets and 75 MWd/kg for annular pellets. The initial phase of the fuel-irradiation program, now under way, consists of irradiation of a few assemblies to higher fuel burnups by using current-design light-water reactor fuel. The next phase will consist of the irradiation of demonstration assemblies designed specifically for high fuel burnup. These demonstration assemblies will be followed by an entire batch loading of improved fuel. The batch load irradiation is included before broad-scale commercial use for commercial risk considerations rather than for safety reasons. Ramp transient tests will be performed in parallel with the irradiation of demonstration assemblies.

Minimum time for design for demonstration fuel assemblies specifically for high burnup and preparation of a Preliminary Safety Analysis Report and an Environmental Report for the demonstration assemblies is estimated at 2 years. The design of fuel element assemblies specifically for extended burnup will be done in parallel with the irradiation of current-design light-water reactor fuel to extend burnups. From this time to submittal of a Preliminary Safety Analysis Report and an Environmental Report for a commercial core loading will be 6 years at a minimum, which includes approximately 1 year for Nuclear Regulatory Commission review and loading of the demonstration assemblies, and 5 years of irradiation for average fuel burnups of 50 MWd/kg. The Preliminary Safety Analysis Report for the commercial core is assumed to be prepared during completion of the irradiation period.

Improvements in fuel utilization of less than 15 percent could be licensed earlier as test data become available and design analyses are completed.

3.7.2 Light-Water Breeder Reactor, Prebreeder/Breeder

The light-water prebreeder and breeder reactor cores would be backfitted into existing pressurized-water reactor plants or into new pressurized-water reactors. The reactor plant would not require demonstration from a safety viewpoint because the heat transfer loop and safety system components are

Table 4. Minimum Time Estimates for Commercial Use (years)

	Minimum Time to Submit PSAR and ER for Demonstration Stage ^a		Minimum Time to Submit PSAR and ER for Commercial Use ^a
	Plant	Fuel	
Light-Water Reactor, 15% improved	N/A ^{b/}	2	8
Light-Water Breeder Reactor Prebreeder	N/A	5	12
Light-Water Breeder Reactor	N/A	c	7
Heavy-Water Reactor	d	d	3
High-Temperature Gas-Cooled Reactor, Steam Cycle	N/A	e	2
High-Temperature Gas-Cooled Reactor, Direct Cycle	4	f	16
Liquid-Metal Fast- Breeder Reactor, loop-type	4	f	16
Liquid-Metal Fast- Breeder Reactor, pool-type	g	g	6
Gas-Cooled Fast Reactor	6	f	18

^a Measured from the present time; PSAR, Preliminary Safety Analysis Report; ER, Environment Report.

^b Not Applicable.

^c Demonstration is under way at the Shippingport reactor.

^d Not needed, based on operation of Canadian deuterium-uranium reactors and availability of Canadian safety data.

^e Demonstration of the fuel is under way at the Ft. St. Vrain reactor.

^f Demonstration of the fuel to be completed along with the demonstration of the plant.

^g Not needed, based on demonstration of plant components and fuel in Superphenix and availability of French safety data.

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essentially unchanged from current pressurized-water reactors. The fuels, however, will need qualification.

The prebreeder fuel would be either a high-enriched uranium/thorium binary-solution type, a 15- to 20-percent U-235 enriched uranium/thorium duplex type, or plutonium/thorium binary-solution type. All of these would require irradiation testing of assemblies, followed by irradiation testing of one batch loading before a commercial core could be licensed. The minimum time to a submittal of a Preliminary Safety Analysis Report and an Environmental Report for a demonstration batch would be 5 years for lead assembly design, fabrication, and irradiation testing. The Preliminary Safety Analysis Report and the Environmental for the demonstration batch would be prepared during the irradiation of the fuel assemblies.

After submittal of the Preliminary Safety Analysis Report and the Environmental Report for the demonstration batch, 7 years would be needed before Preliminary Safety Analysis Report submittal for a commercial core. This time includes the Nuclear Regulatory Commission review for the demonstration batch (approximately 1 year), fabrication and loading of the demonstration batch (approximately 2 years), and irradiation of the batch (approximately 4 years). The commercial core design and Preliminary Safety Analysis Report and Environmental Report preparation is assumed to take place in parallel with the demonstration batch irradiation.

The breeder fuel is currently being demonstrated at Shippingport Atomic Power Station, and full data are expected within 5 years. The thermal-hydraulic behavior of the close-packed lattice core design will be demonstrated in a nonnuclear test facility during this period. With 2 additional years for core design and document preparation, a Preliminary Safety Analysis Report and an Environmental Report for a commercial core could be submitted in 7 years. The light-water breeder reactor concepts that are not backfitted into existing pressurized-water reactor designs but rather into new plants with some design changes could proceed directly to commercial plant licensing on the basis that the changes to the plant components will not involve new safety issues. However, the use of this alternative is limited by the availability of U-233 from the prebreeder and the availability of fuel-cycle facilities. As discussed under the licensing time assessment for the various fuel cycles, the time needed to bring the U-233/thorium fuel to a commercial operating license could be at least 20 years.

3.7.3 Heavy-Water Reactor

A demonstration stage for this plant and fuel concept may not be needed because of its successful operation in Canada. Using the Canadian data base on the reactor and the slightly enriched uranium fuel, which is routinely made available as part of the standard Canadian Deuterium-Uranium licensing agreement, a Preliminary Safety Analysis Report and an Environmental Report for a commercial plant in the United States could be submitted in 3 years. This time includes some preliminary resolution of differences between U.S. and Canadian design criteria and standards, completion of commercial design criteria and document preparation.

3.7.4 High-Temperature Gas-Cooled Reactor

Although the criteria for full-size heat-transfer loop operation are not satisfied by the Fort St. Vrain Reactor plant, a commercial unit was almost at the point of being licensed by the Nuclear Regulatory Commission for construction in 1975. This fact, combined with existing operating experience of commercial-size gas-cooled reactors in England, supports the assumption that a commercial plant can be licensed in the United States without a preceding full-size demonstration plant. A definitive commercial design could be completed and a Preliminary Safety Analysis Report and an Environmental Report prepared in about 2 years, considering that such a design was in progress by Gas-Cooled Reactor Associates in 1978, before suspension of design work.

Direct Cycle

The direct-cycle high-temperature gas-cooled reactor shares a number of safety features with the steam-cycle version. However, the introduction of turbomachinery (turbine, compressor, shaft, and seals) into the primary system leads the safety and licensing questions not addressed in the steam-cycle development. Successful demonstration-plant operation is not only prudent in terms of economic risks, but is also a requirement before licensing for general commercial use.

Based on plans formulated by the Gas-Cooled Reactor Associates, the time to complete this demonstration plant design and to prepare for the submittal of the Preliminary Safety Analysis Report and the Environmental Report would be approximately 4 years. After submittal, 12 years will be needed for submittal for a commercial plant. This time includes 3 years for review of the demonstration plant Preliminary Safety Analysis Report, 7 years for construction,

and 2 years for operation of the demonstration unit. Completion of the commercial design and document preparation are assumed to take place during the last stages of construction and operation of the demonstration plant.

3.7.5 Liquid-Metal Fast-Breeder Reactor

Commercial-scale liquid-metal fast-breeder reactors have not been demonstrated with full-size heat-transfer components or fuel-element assemblies, although smaller-size plants have operated in the United States and in other countries. The most advanced project at this time is the 1200-MW Superphenix, scheduled for operation in France, beginning in 1984. However, the U.S. commercial design may not be the same as the pool-type Superphenix. Loop-type designs such as the Clinch River Breeder Reactor or the Fast Flux Test Facility are also being considered.

The current Department of Energy plan is that decisions about the timing and the type of fast breeder reactor demonstration plant in the United States will be made by the end of 1981.

Loop-Type Design

If the U.S. commercial design were of the loop type, a demonstration plant with full-size heat transfer loops would be required to ensure that safety issues were resolved and that components were realistically tested for reliability and maintainability before commercial deployment. It would probably require about 2 years to complete a demonstration plant design and prepare a Preliminary Safety Analysis Report and an Environmental Report, that is, 4 years from the present time, assuming that the conceptual design study now under way for the large demonstration plant is continued at an adequate level of effort during this 4-year period.

After submittal of the Preliminary Safety Analysis Report and the Environmental Report for the demonstration plant, 12 years would be needed for a submittal for a commercial plant. This time includes 3 years for review of the documents, 7 years for construction, and 2 years for operation of the demonstration unit. Completion of the commercial plant design and document preparation are assumed to take place during the last stages of construction and operation of the demonstration plant.

Pool-Type Design

Conceivably, a decision may be made to proceed with a breeder based on the Superphenix design if its operation is successful. In that case, assuming the French safety data and analyses are available, a Preliminary Safety Analysis Report and an Environmental Report for a commercial plant modified appropriately for U.S. siting considerations and Nuclear Regulatory Commission requirements, could be submitted after 2 years of Superphenix operation, that is, 6 years from the present time. If the U.S. design were substantially different from that of the Superphenix, a demonstration plant would probably be required, and the schedule would resemble that of the loop-type design.

3.7.6 Gas-Cooled Fast Reactor

Because a gas-cooled fast reactor has not been built yet, the issue has been raised as to whether or not a small prototype would be needed before operating a large-size demonstration plant. This assessment concludes that a small prototype would not be needed from a safety viewpoint, although there might be some element of risk to the research and development program. The prototype would not be needed before the demonstration plant from the safety viewpoint because many of the components and systems are generically similar to those of the steam-cycle high-temperature gas-cooled reactor, which has been demonstrated in the Fort St. Vrain plant. The fuel and physics are generally similar to those of the liquid-metal fast-breeder reactor. In addition, a partial regulatory framework of general design criteria has been developed for fast-breeder reactors through the licensing process for the Clinch River Breeder Reactor Project. The core heat transfer and fluid flow that are unique to the gas-cooled fast reactor could be addressed in nonnuclear experiments and test loops before construction of the demonstration plant. Physics experiments can be performed in the existing critical facilities for the liquid-metal fast-breeder reactor. Transient safety tests can be performed in the Experimental Breeder Reactor-2 and in the Fast Flux Test Facility.

Assuming that a small prototype is not necessary, the first reactor could be a demonstration plant containing at least two heat-transfer loops of a size equal to the first commercial plant. Based on plans formed by the Helium Breeder Associates, the time required for the submittal of a Preliminary Safety Analysis Report and an Environmental Report for the demonstration plant is estimated at approximately 4 years and would allow for completion of the design and the experimental work. Because the decision on the future of the breeder is not expected until at least 1981, this timeframe means 6 years from

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the present time, assuming that the necessary level of research, development, and demonstration plant design effort is conducted during this 6-year period.

After the submittal of the Preliminary Safety Analysis Report and the Environmental Report for the demonstration plant, 12 years would be needed before a submittal could be made for a commercial plant. This time includes 3 years for review of the demonstration plant documents, 7 years for construction, and 2 years for operation of the demonstration unit. Completion of the commercial plant design and document preparation is assumed to take place during the last stages of construction and operation of the demonstration plant.

4. FUEL CYCLE LICENSING CONSIDERATIONS

Each of the fuel concepts listed in Table 1 may require some or all of the following fuel-cycle facilities:

- o Uranium and/or thorium mines and mills
- o Uranium hexafluoride conversion plants
- o Enrichment plants
- o Fuel fabrication plants
- o Chemical reprocessing plants
- o Waste handling and treatment facilities
- o Transportation equipment (vehicles, containers)

Safety issues for licensing each of these types of facilities are discussed in Section 4.1. Section 4.2 addresses the environmental effects of alternative fuel cycles as compared to the once-through light-water reactor fuel cycle. Finally, the time to achieve licensing of commercial fuel-cycle facilities is discussed in Section 4.3.

4.1 SAFETY ASPECTS

4.1.1 Mining and Milling

No major unresolved safety issues related to uranium mining and milling remain. No Federal licensing authority currently exists for mines; however, Federal regulations do govern the occupational health and safety of mine and mill workers. A large industry exists in the United States for the mining and milling of uranium ores, and safety requirements for the industry are well defined. An environmental development plan (Reference 15) has been prepared to describe the planning and management requirements and the schedules needed to evaluate and assess the safety, health, and environmental aspects of the uranium mining, milling, and conversion technologies.

At present, no industry in the United States currently produces thorium as a primary product, and no industrial experience is available for use in producing thorium from thorite ore. No licensing or safety evaluations of thorium mines and mills have been performed; however, the processes for producing uranium and thorium are similar. Therefore, safety considerations should be similar for uranium and thorium mines and mills. No research, development,

and demonstration requirements for licensing have been identified for thorium mining and milling.

4.1.2 Uranium Hexafluoride Conversion

Two conversion facilities in the United States are licensed to operate with U_3O_8 concentrates as feed. No major problems are foreseen for future licensing of new or expanded plants of this type, and no unresolved development requirements related to nonenvironmental safety remain.

No reason is known for believing that a uranium hexafluoride conversion facility cannot be designed to operate safely to produce uranium hexafluoride from U_3O_8 concentrates or uranyl nitrate produced by a chemical reprocessing plant. In the latter instance more stringent requirements may be necessary for containing the slightly radioactive process stream and for controlling potential criticality.

Because no commercial reprocessing exists in the United States, and because prior reprocessing experience has excluded the uranyl nitrate to uranium hexafluoride conversion step, plants using uranyl nitrate feed have not been licensed. However, there has been considerable Government experience, and it is reasonable to conclude, that licensing, safety, and development requirements for this type of plant would be similar to those for a U_3O_8 concentrate conversion facility. A primary difference would be the effect that feed from reprocessing--feed with characteristics differing from the reference feed (e.g., contaminated with radioactivity slightly above background)--would have on plant safety. Fuel cycles in which recovered U-235 might be contaminated with very small amounts of U-232 and then reenriched are of principal concern, such as the light-water reactor self-generated recycle. Another consideration is potential criticality if the recycle uranium is above natural U-235 enrichment. The presence of residual organics could create an additional hazard.

4.1.3 Enrichment

No major unresolved safety problems are identifiable for gaseous-diffusion or gas-centrifuge technology. Work is under way to identify and resolve the specific safety and environmental problems of the advanced isotope separation processes.

Safety issues for gaseous-diffusion and gas-centrifuge plants enrichment are similar and include criticality control; contaminant control if recycle uranium is returned for reenrichment; and protection against accidental releases of uranium hexafluoride. No remaining safety development requirements have been identified for gaseous-diffusion enrichment because the technology is highly developed and has been operated on a large scale for more than 30 years.

Because gas-centrifuge plant inventories of uranium hexafluoride, oil, and chemicals are generally lower than those for a diffusion plant, accident consequences may be reduced. The engineering design of the centrifuge facility will have to consider the effects of failure of high-speed rotating machinery, a requirement that is generally absent in gaseous-diffusion facilities.

An environmental development plan (Reference 16) has been prepared by the Department of Energy for the gaseous-diffusion and the centrifuge programs that identifies and examines the safety and environmental concerns and requirements. An environmental development plan (Reference 17) has also been prepared by the Department of Energy to describe the planning and management requirements and schedules needed to evaluate and assess the safety, health, and environmental issues of advanced isotope separation processes. Some key issues are similar to those for the gaseous diffusion and centrifuge processes: the reduction of radioactive effluents to a level as low as reasonably achievable and the handling of radioactive contaminants in the advanced isotope separation processes. Another issue is the biological effects of process chemicals. Unique issues for advanced isotope separation processes are the potential occupational health effects from magnetic/electromagnetic fields and from lasers.

For the near term, all gaseous-diffusion and gas-centrifuge enrichment facilities will be owned by the Federal Government and, therefore, not subject to licensing by the Nuclear Regulatory Commission. They are, however, subject to air- and water-quality standards and to occupational safety standards of both State and Federal Governments. Nuclear Regulatory Commission licensing requirements for commercial enrichment facilities are not yet defined although a Safety Analysis Report guide has been issued. There are no commercial license applications active at the present time.

4.1.4 Fuel Fabrication

The fuel materials employed in the principal fuel cycles considered in the NASAP assessment, as presented earlier in Table 1, are oxides of uranium, uranium/plutonium, uranium/thorium, and plutonium/thorium, except for the oxycarbide compounds that are used in the high-temperature gas-cooled reactor. Carbides are an alternate fuel for the liquid-metal fast-breeder reactor. Three fuel forms are being considered: pellet-type fuel for low-radioactivity materials; pellet or sol-gel, spherepac fuels for high-radioactivity materials; and coated-particle fuel for the high-temperature gas-cooled reactor.

Fuel fabrication includes the manufacturing steps, starting with converted fissile material from a reprocessing plant (see Section 4.1.5) and for various pieces of hardware, and ending with an assembled fuel element ready to ship to a power reactor.

The principal difficulty in licensing fuel fabrication plants will be the radioactivity of the fuels. Low radioactivity permits contact operation and maintenance. High radioactivity requires remote operation and, possibly, remote maintenance. Low-radioactivity fuels include those used in once-through fuel cycles and plutonium recovered from low-burnup fuel, for recycle either in light-water reactors or in breeder reactors. High-radioactivity fuels include plutonium recovered from high-burnup fuel, U-233 produced from thorium in power reactors, fuels with a spikant added at the reprocessing plant, and fuels derived from low-decontamination reprocessing (see Section 4.1.5). If preirradiation of low-radioactivity fuel assemblies is used for safeguards purposes or if the fuel is spiked during the fabrication step for the same reason, it becomes high-radioactivity fuel and is discussed under that heading.

Low-Radioactivity Fuels

No major unresolved safety issues remain in the fabrication of low-radioactivity fuels. Safety concerns center around prevention of accidental criticality, fires, and explosions.

Plants for fabricating low-radioactivity, low-enriched uranium oxide light-water reactor fuel have been licensed and are now operating. Pilot facilities have been licensed to fabricate low-radioactivity, uranium/plutonium fuel for light-water reactor and liquid-metal fast-breeder reactor demonstration fuel

tests and for the core loadings for the Fast Flux Test Facility, a liquid-metal fast-breeder reactor. Pilot facilities to fabricate low-radioactivity, uranium/thorium fuel for the high-temperature gas-cooled reactor prototype and demonstration reactors are licensed and operating. A pilot facility to fabricate low-radioactivity, uranium/thorium fuel was owned by Nuclear Fuel Services and was licensed to produce prototype fuel for critical facility experiments. Fuel elements for the light-water breeder reactor at Shippingport were fabricated from low-radioactivity U-233/thorium fuel at the Bettis Atomic Power Laboratory. All of these facilities produced pellet fuels, except the high-temperature gas-cooled reactor facilities, which produced coated-particle fuel compacts for the Peach Bottom I prototype reactor and coated-particle fuel rods for the Fort St. Vrain demonstration reactor.

High-Radioactivity Fuels

No commercial facilities to manufacture high-radioactivity fuels have been reviewed or licensed by the Nuclear Regulatory Commission; thus, the safety issues have not been clearly defined. When compared with fabrication of low-radioactivity fuels, the fabrication of high-radioactivity fuels is expected to introduce the additional safety issues of meeting acceptable limits on occupational exposure and preventing releases of radioactivity during accidents or natural disasters. It appears feasible to meet the requirements for occupational exposure through shielding, remote operations, and containment. Protection against releases of radioactivity appears to be feasible through adequate design of safety systems and mitigating systems, such as the containment building. Because releases of high-radioactive materials from these facilities would create a more severe radiological hazard to workers and to the public, the structure of the interior and exterior and the systems used to contain the radioactive materials will have to meet more stringent design requirements than those used for low-radioactivity fuel.

To reduce worker exposure, remote operations will be required for some or for all of the following operations: production of powder and pellets, encapsulation of fuel, and performance of final assembly operations. Stringent quality control will be required in all of these operations and will have to be accomplished with minimum acceptable worker exposures. These activities and the maintenance procedures involved (which may also have to be performed remotely) will require extensive development before licensing of a commercial facility could proceed.

The accidents that will be of concern are similar to those for low-radioactivity fuel, but the consequences of a release would be greater. Such releases could occur in case of nuclear criticality, fire, or explosions.

Flammable liquids, which may be used in scrap-recovery operations, diluted hydrogen gas used in sintering furnaces, or combustible gases used in high-temperature gas-cooled reactor sphere-coating operations are potential sources of fires or explosions. The impact of these accidents occurring in a remotely operated facility must be addressed in the design of the facility.

Experience to date in fabricating high-radioactivity fuels is limited. Questions of safety requirements must be resolved before the licensing of a commercial facility. Laboratory-scale process development with nonradioactive materials is underway for fuels for the high-temperature gas-cooled reactor and for the liquid-metal fast-breeder reactor. Small amounts of radioactive fuels have also been fabricated for test purposes. A pilot-plant facility has been constructed, and equipment is being developed for fabrication of high-temperature gas-cooled reactor uranium/thorium fuel. A similar facility is being constructed and equipped for the fabrication of liquid-metal fast-breeder reactor fuel. Substantial development time and effort will be required to bring fabrication processes for these high gamma-ray and neutron-radiation fuels from the laboratory stage through pilot-plant demonstrations and into commercial production. However, on the basis of experience in fabricating small quantities of these fuels, no insurmountable safety obstacles are foreseen.

If preirradiation of the fuel assemblies were performed for safeguards reasons, a reactor facility would be associated with the fuel-fabrication plant. A description of a conceptual design of this type of facility is provided in Reference 18. The operation of this facility would require many of the same safety features as would a power reactor. Some novel features might be needed because of frequent loading and unloading of fuel.

4.1.5 Chemical Reprocessing

The chemical reprocessing plant includes (1) spent-fuel storage; (2) a head-end treatment for dissolving the fuel materials; (3) a selective extraction to separate the uranium, plutonium and thorium from the fission products; (4) interim storage of plutonium or U-233; (5) a conversion step to convert the recovered product solutions to solid or other forms for further use; and (6) a waste pretreatment to convert the fission product solution to a solid form for shipment to a waste repository. Items (1), (2), and (3) are discussed under the heading of Product Separation; items (4), (5), and (6) are discussed under subsequent headings.

Product Separation

The basic chemical processes utilized for dissolution and selective extraction of the fuel alternatives being considered in NASAP are based on the PUREX process, which is used to recover the uranium and the plutonium, and on the THOREX process, which is used when thorium is present.

PUREX Process for Light-Water Reactor Fuel--Although commercial reprocessing plants were reviewed and licensed in the 1960's and early 1970's, no commercial facility is now licensed to operate. The key safety issues affecting licensing of new PUREX reprocessing facilities include minimizing in-plant occupational exposures; designing building and critical equipment to withstand natural disasters (earthquakes, tornadoes); preventing accidental criticality, explosions, or fires; and protecting fuel and equipment from accidentally dropped fuel elements. Improvements in occupational exposures are particularly important if the current limits are lowered, as the Federal Interagency Review Group is contemplating recommending. Through appropriate design and operating procedures, these issues are expected to be satisfactorily resolved to allow safe operation of a PUREX reprocessing plant.

Even though important safety issues still require resolution in PUREX reprocessing, more data are available for the PUREX process than for any reprocessing technique. The process has been used in nonlicensed government facilities to process low-burnup uranium fuels for many years. Three commercial fuel reprocessing facilities have been constructed but only one of these (the Nuclear Fuel Services plant at West Valley, New York) was operated. The Nuclear Fuel Services plant was shut down in 1971 for modifications to increase capacity but was not modified because of excessive estimated costs, to meet upgraded licensing requirements. The General Electric plant at Morris, Illinois, was not commissioned because of technical problems and because of the volatile nature of licensing and environmental issues. However, these two plants are currently licensed to store irradiated light-water reactor fuel.

The licensing review for start of operation of the Allied General Nuclear Services plant at Barnwell and the review for the construction license for an Exxon Nuclear Corporation plant were suspended in 1977. The Barnwell plant construction is essentially completed for the process steps through the PUREX separation of nitrate solutions of uranium and plutonium; construction had not begun for the Exxon plant.

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In addition to the safety issues previously identified, research, development, and demonstration must be completed in areas of reducing radioactive emissions, especially Kr-85, iodine, tritium, and C-14 during normal operations; immobilizing and encapsulating high-level, intermediate-level, and gaseous wastes; and converting plutonium nitrate to the solid form (see later discussions).

Environmental effects are treated separately; the environmental impacts of reprocessing are compared with those of the once-through cycle in Section 4.2.

THOREX Process for Light-Water Reactor Fuel--The key safety issues for THOREX reprocessing of light-water reactor fuels are similar to those for the PUREX process but with the additional concerns of controlling corrosive attacks in the dissolution step, possibly leading to accident, and protecting workers against the high radioactivity of the recovered U-233 product.

No commercial facilities are licensed to reprocess thorium-containing light-water reactor fuel, although the Nuclear Fuel Services plant in West Valley, New York, did reprocess the Indian Point 1 uranium/thorium dioxide fuel in the late 1960's. Most of the experience with the THOREX process has been in Government-owned facilities. The THOREX pilot plant demonstration program at the Oak Ridge National Laboratory recovered U-233 from irradiated thorium metal in the late 1950's. During the 1960's, large quantities of thorium were irradiated and reprocessed at the Hanford Reservation, Richland, Washington, and at the Savannah River Plant at Aiken, South Carolina, to recover purified U-233 for use in demonstration of the thorium-fueled concepts, such as the high-temperature gas-cooled reactor and the water-cooled breeder reactor.

Safety developmental programs unique to the THOREX process are expected to be directed toward selection of construction materials for extended use in a THOREX facility. These materials are needed to impede corrosion by the fluoride used to dissolve the thorium dioxide. The programs will also be directed toward designs and procedures for reducing occupational exposure from U-233. Measures similar to those that would be used for low-decontaminating or spiking alternatives (see Other Process Variants, below) would probably be required for reducing exposure from U-233.

THOREX Process for Water-Cooled Breeder Reactor Fuel--This reprocessing option will be similar to the THOREX process for light-water reactor fuel, except for the use of a selective dissolution process or a different head-on operation

for duplex fuel. Safety issues are expected to be the same as those discussed above for light-water reactor fuel containing thorium.

THOREX Process for High-Temperature Gas-Cooled Reactor Fuel--Additional safety issues for high-temperature gas-cooled reactor fuel reprocessing as compared to reprocessing light-water reactor fuel containing thorium involve principally the head-end combustion process for the graphite fuel. Resolving these issues will require research, development, and demonstration.

Several novel considerations would be involved in licensing of a high-temperature gas-cooled reactor fuel reprocessing plant. This fuel reprocessing requires a head-end operation of crushing, burning, and leaching the ash to separate the fuel materials from the graphite fuel element and to put them into solution. Reprocessing the graphite fuel blocks requires the use of off-gas treatment systems for C-14 and Rn-220 released from burning the fuel blocks and fuel elements. Such systems will be subject to accidental failure that could cause gas releases, which could endanger the operators and possibly the environment (e.g., graphite fluid-bed combustion units may have burnout and explosion protection problems).

Development programs have supported high-temperature gas-cooled reactor fuel reprocessing with cold pilot plant work and laboratory-scale reprocessing of small quantities of irradiated fuel. As with all THOREX processes, product handling is complicated because the U-232 radiation persists into the final product.

Processes for Liquid-Metal Fast-Breeder Reactor or Gas-Cooled Fast Reactor Fuel--Major safety issues in reprocessing breeder reactor blanket fuels (by using the PUREX process for the uranium/plutonium cycle or the THOREX process for the uranium/thorium cycle) include those previously discussed for the basic processes for light-water reactor fuel. Reprocessing of breeder cores adds other safety concerns. Plutonium-fueled breeder cores exhibit higher fissile content, requiring more stringent control over accidental criticality, and show higher heat generation rates than light-water reactor fuels (especially for breeder fuels that are cooled for short times). These factors could create greater potential for fuel failure if cooling in the fuel storage facility were lost. In addition, the liquid-metal fast-breeder reactor reprocessing plant may require fuel storage in sodium and may involve a sodium removal step in the head-end, either of which introduces a safety design problem not present in light-water reactor fuel reprocessing.

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No breeder fuel reprocessing plants have been licensed or reviewed by the Nuclear Regulatory Commission. Nonetheless, appropriate developmental programs on light-water reactor reprocessing and on the unique aspects of breeder fuels (higher heat-generation rates, high fission product concentrations, higher plutonium concentrations in core fuel, and different cladding materials) are expected to resolve the above safety issues.

Other Process Variants

Coprocessing--By making some adjustments in the solvent extraction step of the PUREX process, production of a mixed uranium/ plutonium product is possible that may be more difficult to divert or use for nuclear-weapons production. This variant of the PUREX process introduces no new safety issues.

Low Decontamination or Spiking--A second variant of the PUREX process, low decontamination, would modify the process to leave some of the radioactive fission products in the product stream. The product would be radioactive and, therefore, be more difficult to divert for nuclear weapons use. Similar results could be achieved by adding radioactive materials (spiking) like Co-60 to the product stream.

These process variants would introduce new safety issues. Reprocessing options involving low decontamination or spiking will have to be evaluated for additional occupational exposures and the possibility of additional releases from accidents involving the product storage tanks. These process variants have not been addressed in licensing applications.

Interim Storage of Plutonium or U-233

Storage of plutonium or U-233 creates safety concerns for accidental criticality, for occupational exposure, and for radioactive releases caused by industrial accidents to the storage facility. These issues are not expected to prevent licensing of such plants.

No facilities for storage of plutonium or U-233 are currently licensed, and none are planned. These materials could be stored either as a solution or as a solid. Solutions of both U-233 and plutonium have been stored previously: plutonium for short times at the Nuclear Fuel Services facility at West Valley, New York, and U-233 at Oak Ridge National Laboratory for more than 10 years.

Research, development, and demonstration requirements for safety of the interim storage facilities for plutonium or U-233 are expected to be met during design and licensing of the commercial facilities.

Conversion Processes

Safety issues in converting to solid form the liquid products recovered from the solvent extraction step include preventing nuclear criticality, maintaining acceptable levels of occupational exposure, and preventing radioactive releases from industrial-type accidents or natural catastrophes.

No licensed commercial facilities for product conversion exist. Once a reference process is selected, a decision that may be affected by safeguards and nonproliferation properties of the solid product form, the design and licensing of a facility can be expected to proceed on the basis of safety issues already resolved.

Conversion of plutonium nitrate solution into plutonium dioxide powder utilizes technology that has been demonstrated in licensed pilot plants producing mixed oxide fuels for testing in light-water reactors and for the Fast Flux Test Facility. Radioactive spills and contamination through dust transport require greater attention in the future to prevent possible occupational exposures.

Coprecipitation of the product from coprocessing results in a blended uranium/plutonium product. This process introduces no new safety concerns as compared with conversion of plutonium nitrate to PuO_2 powder.

The sol-gel conversion process, now under pilot-plant development, produces sintered gel microspheres rather than powder. This process has the potential advantages of simpler processing steps, assurance of fuel homogeneity, and elimination of some "dusty" process operations. The process steps appear to be particularly suited to the remote operations required in fabricating such radioactive fuel such as those encountered in thorium-based fuel cycles, low-decontaminated fuel, or spiked fuel. No new safety concerns are introduced by this process that cannot be resolved by development of the process itself.

Waste Pretreatment

Safety concerns in waste pretreatment are the safe storage of high-level wastes in solution before solidification, worker exposure, and hazards to the public in the event of accidents and/or failures of the operating equipment for solidifying the wastes.

The highly radioactive fission product solutions from reprocessing plants must be converted to solid form for transfer to a Federal repository. One process being considered is spray-drying the solution. The resulting calcine granules are mixed with glass frit and melted as a glass into a container for shipment to the repository. The development and demonstration of waste vitrification processes, equipment, and procedures are a major effort.

If plutonium were to be stored with the wastes, an assessment of the complications such as the potential for criticality in waste disposal of large amounts of such materials in aqueous and solid wastes would be required. Facilities to handle and package the transuranic wastes would have to be developed.

4.1.6 Waste Handling and Treatment

The waste-handling needs for the alternative fuel cycles under review are summarized in Table 5.

Table 5. Waste-Handling Needs for Alternative Fuel Cycles

	Once-Through	Recycle
Retrievable interim storage of spent fuel	X	
Geologic disposal of spent fuel	X	
Shallow land disposal of low-level waste	X	X
Transuranic waste disposal		X
Geologic disposal of solidified, high-level waste		X

Each type of waste-handling facility is discussed in the paragraphs below. All of the technical issues and problems for each type of facility are expected to be resolved in development of the alternative fuel cycle. Nontechnical issues may be more difficult to resolve and are indeterminate at present.

Retrievable Interim Storage of Spent Fuel

Key safety issues of retrievable spent-fuel storage are primarily those of criticality, loss of cooling for wet storage, occupational exposures, fires with high-temperature gas-cooled reactor graphite fuels, ruptured fuel caused by dropped fuel or spent fuel-casks, and radioactive release caused by industrial-type accidents. All of these issues can be resolved in the facility design. In fact, light-water reactor spent fuel has been stored safely for more than 20 years.

Retrievable storage of light-water reactor fuel assemblies is currently licensed at reactor storage pools. No licensed away-from-reactor storage facilities exist except for those associated with reprocessing plants at West Valley, New York, and Morris, Illinois. Because the space in these two facilities is limited, utilities are expanding reactor storage pool capacity by installing new storage racks with closer center-to-center spacing. However, the storage achievable by this method is limited. New at-reactor storage facilities or away-from-reactor storage will be required for some reactors, beginning in the mid-1980's.

The only licensed commercial fuel-storage facility in the United States is for light-water reactor fuel. The high-temperature gas-cooled reactor demonstration plant is licensed to store one-third of a core but must then ship spent fuel to a U.S. Government facility in Idaho. Canadian deuterium-uranium reactors routinely store spent fuel at reactor storage facilities. These systems have not been reviewed by the Nuclear Regulatory Commission against U.S. criteria; however, no reason has emerged for suspecting any unique safety problem that would prevent licensing such a storage facility at a heavy water reactor in the United States.

Geologic Disposal of Spent Fuel

Key safety issues in the geologic disposal of spent fuel include the effects of (1) accidents in the encapsulation facility at the surface of the repository, (2) mining-type industrial accidents when constructing, operating,

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and back-filling the repository, (3) accidental criticality in the storage media, and (4) accidental intrusion into the underground storage area many years after its decommissioning.

No licensed spent-fuel geologic repositories currently exist, nor are any sites yet identified for wide-scale commercial use. Geologic disposal of spent fuel is under intense review by Government agencies and was the subject of a recent major report by the Federal Interagency Review Group (Reference 19). The Department of Energy has work under way leading to tests of spent fuel disposal at the Nevada Test Site and at Hanford, Washington. This work is part of the research and development program to develop the technology and evaluate geologic media leading to the siting, licensing, and construction of one or more large-scale disposal facilities. Draft criteria for radioactive waste disposal have been issued for comment and guidance by the Environmental Protection Agency (Federal Register, Vol. 43, No. 221, pg. 53262, Nov. 15, 1978). The Nuclear Regulatory Commission is developing implementing standards and guides, and is expected to review the geologic repository facility design, and the current outlook is that such a facility would be Government operated.

Accidents in the encapsulation facility would be similar to those in a surface facility for retrievable storage of spent fuel but would also include accidents and occupational exposures in handling the spent fuel during the encapsulation operation. The safety of underground operations will require extensive development to address the long-term safe storage of spent fuel.

Shallow Land Disposal of Low-Level Waste

No key safety issues are known to exist in the shallow land disposal of low-level waste. The major safety issues at the burial location are transportation and industrial-type accidents. Transportation requirements over the whole fuel cycle are considered in Section 4.1.7 and are not discussed further in this section.

Shallow land disposal sites exist for low-level wastes generated by reactors and fuel cycle facilities for light-water reactors on the once-through cycle. Problems have been experienced with migration of low-level radioactivity from the burial location and with some "poor-housekeeping" operating practices which led to a poor public image. The issues involve environmental and institutional control rather than safety.

Transuranic Waste Disposal

There are no safety issues involved in transuranic waste disposal that will not be resolved in developing acceptable methods for the geologic disposal of either spent fuel or solidified high-level waste. Development requirements are expected to involve principally detailed procedures for preparation, packaging, and placement of these wastes rather than any generic safety issue.

Geologic Disposal of Solidified High-Level Waste

No licensed facilities for disposal of these wastes exist. Until issues are resolved concerning acceptability of reprocessing, preferred waste forms, and geologic media, no licensing of such facilities will be required. The safety issues for disposal of solidified high-level waste include those already identified for spent-fuel disposal, except that nuclear criticality is not a problem provided plutonium is not classified as "waste." If the plutonium from spent-fuel reprocessing is included in the high-level waste, then the potential for criticality must be considered.

Government research, development, and demonstration programs related to spent fuel disposal also emphasize disposal of solidified high-level waste. The major issues of this step in the fuel cycle are institutional and environmental concerns rather than safety.

4.1.7 Transportation

The primary safety issue in transportation is accidents that cause death or injury to transportation workers or to the public. Differences in transportation requirements between all alternative fuel cycles are judged to be well within a factor of 2, and small enough not to be a basis for preferring one cycle over another. These differences, correspond to reductions of 5 injuries and 0.3 fatalities per year for 180 GW of installed liquid-metal fast-breeder reactor generating capacity in 2025 as compared to the light-water reactor once-through cycle. This amount of breeders is the maximum deployment considered in this assessment and is the maximum difference in transportation impact expected between all the alternative fuel cycles.

Shipment of one or more of the following materials is required for the alternative fuel cycles: uranium and/or thorium ore; uranium orthorium

concentrates; natural UF_6 ; enriched UF_6 ; fresh fuel assemblies; spent-fuel assemblies; and radioactive wastes from all the above operations. The following discussion divides the fuel-cycle materials into low-specific-activity and high-specific-activity materials. These designations are defined in detail in Title 10 of the Code of Federal Regulations, Part 71. For the fuel cycles examined, low-specific-activity materials include uranium and thorium ores, unirradiated uranium (depleted or natural), uranium and thorium concentrates, natural UF_6 , enriched UF_6 , and waste streams containing these materials. All other radioactive materials in the fuel cycles would normally be classified as high specific activity unless diluted to meet the criteria for low specific activity specified in Title 10, Part 71.

Some radioactive wastes, primarily from 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. However, the radioactivity levels of these medium-level wastes are not high enough 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 report classifies as low-specific-activity materials all radioactive waste that could be transported in either lightly shielded or unshielded drums.

Low-Specific-Activity Materials

Because of the high energy potential per unit of weight in nuclear fuels, transportation requirements in the nuclear fuel cycle are far smaller than for other energy sources, such as coal, oil, etc. Table 6 shows typical annual transportation requirements for low-specific-activity materials including low- and medium-level wastes for the light-water reactor on a once-through cycle and for the liquid-metal fast-breeder reactor. In terms of number of shipments and total distance traveled (number of shipments times the average distance per shipment), the transportation steps at the beginning of the fuel cycle involving low specific activity materials are important.

The data in Table 6 for the light-water reactor once-through cycle were taken from the 1974 Atomic Energy Commission report, (WASH-1248), "Environmental Survey of the Uranium Fuel Cycle," (Reference 20) except for low- and medium-level wastes which were taken from NUREG-0116 (Supp. 1 to WASH-1248). For the liquid-metal fast-breeder reactor cycle, the average annual quantities of

Table 6. Lifetime Average Annual Transportation Requirements for Low-Specific-Activity Materials (Per GW)

Light-Water Reactor, Once-Through Cycle ^a			
<u>Material</u>	<u>Quantity (t)</u>	<u>Shipments (number)</u>	<u>Distance (miles)</u>
Ore	91,000	3,350	16,800 truck
U ₃ O ₈	182	12	12,000 truck
Natural UF ₆	270	22	11,000 truck
Enriched UF ₆	52	5	3,750 truck
Low- and Medium-Level Wastes	Not available	13	6,500 truck

Liquid-Metal Fast-Breeder Reactor Cycle ^b			
<u>Material</u>	<u>Quantity (t)</u>	<u>Shipments (number)</u>	<u>Distance (miles)</u>
Heavy Metal (Depleted U or Thorium)	32-35	2	1,000 truck
Low- and Medium-Level Wastes	Not available	56 5.8	27,900 truck 5,800 rail

^aData from WASH-1238, "Environmental Survey of the Uranium Fuel Cycle" (1974). Low and medium level wastes from NUREG-0116 (Supp. 1 to WASH-1248) (October 1976).

^bQuantity of Heavy Metal from NASAP Final Report, Vol. III, heavy metal/shipment and miles per shipment assumed equal to light-water reactor. Low and medium level wastes from ERDA-1535, Final Environmental Statement, liquid-metal fast-breeder reactor (December 1975).

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natural U_{38} , depleted UF_6 , and thorium were derived from the system analyses performed for resource studies reported in Volume III of the NASAP Report. Material quantities per shipment and miles per shipment were assumed to be the same as assumed for the once-through light-water reactor cycle. Low- and medium-level waste transportation requirements for the liquid-metal fast-breeder reactor were taken from the Energy Research and Development Agency report "Final Environmental Statement, Liquid Metal Fast Breeder Reactor Program," (ERDA-1535) dated December 1975.

Statistics on truck accidents have been analyzed (Atomic Energy Commission WASH-1238, 1972) 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 (WASH-1238) to give 1.4×10^{-7} accidents 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 transportation requirements given in Table 6, the annual accident rate per 1 GW reactor would be 0.085, resulting in 0.043 injuries and 0.0026 fatalities.

Although the liquid-metal fast-breeder reactor has increased requirements for transportation of low and medium level wastes, the overall liquid-metal fast-breeder reactor transportation requirements are less than the once-through light-water reactor fuel cycle.

As Table 6 shows, truck transportation requirements for low specific activity for the liquid-metal fast-breeder reactor cycle are only 59 percent of the requirements for the light-water reactor once-through cycle. However, the liquid-metal, fast-breeder reactor cycle has an additional requirement for 5,800 rail miles for shipments of low-specific-activity materials. The annual accident rate per 1 GW reactor would be 0.050, resulting in 0.027 injuries and 0.0016 fatalities.

High-Specific-Activity Materials

High-specific-activity materials in the fuel cycle include fresh fuel, spent fuel, and certain radioactive wastes from reactor operations and from reprocessing facilities which must be placed in specially designed shielded shipping casks.

The annual transportation requirements for high-specific-activity materials in the light-water reactor once-through fuel cycle are summarized in Table 7. Total vehicle miles, assuming truck shipments, for these materials are 160

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Table 7. Lifetime Average Annual Transportation Requirement for High-Specific-Activity Materials (per GW)

Once-Through Light-Water Reactor Cycle ^a		
<u>Material</u>	<u>Shipments</u>	<u>Distance* (Miles)</u>
Fresh Fuel	5	5,000 truck
Spent Fuel	60	60,000 truck
High-Level Wastes	--	-----
Liquid-Metal Fast-Breeder Reactor ^b		
	<u>Shipments</u>	<u>Distance* (Miles)</u>
Fresh Fuel	20	15,000 truck
Spent Fuel	27	20,000 truck
High-Level Wastes	0.16	250 truck
	29	29,000 rail

*Plus an equivalent additional distance for return of empty shipping containers.

^a NUREG-75/038, Supplement 1 to WASH-1238, April 1975, for the LWR.

^b ERDA-1535, "Final Environmental Statement, Liquid Metal Fast Breeder Reactor Program," December 1975.

percent greater than the transportation requirement for low-specific-activity material given in Table 6. The resulting accident, injury and fatality rates for 1 GW of installed generating capacity would be about 0.11 accidents, 0.056 injuries, and 0.0033 fatalities per year.

As shown in Table 7, the total vehicle miles traveled in transporting high specific activity materials for the liquid-metal fast-breeder reactor is about the same as for the once-through light-water reactor cycle. Reduced spent fuel transportation requirements for the liquid-metal fast-breeder reactor are offset by the requirement for transport of high-level wastes from reprocessing. However, almost one-half of the shipping distance would be for rail shipments, which have one-tenth the accident rate, and about one-half the injury and fatality rates as compared to truck shipments. The resulting accident, injury, and fatality rates for 1 GW of installed liquid-metal fast-breeder reactor generating capacity would be about 0.064 accidents, 0.0416 injuries, and 0.0026 fatalities per year.

Total Fuel-Cycle Transportation Impacts

A summary of the annual accidents, injuries, and fatalities from all fuel-cycle activities, including the low-specific activity and the high-specific activity materials, is presented in Table 8 for the light-water reactor once-through cycle and the liquid-metal fast-breeder reactor. The impacts of other fuel cycles would fall between these two cycles. Thus, the liquid-metal fast-breeder reactor fuel-cycle activities would be expected to result in 15 fewer accidents, 5 fewer injuries, and about 0.3 less fatalities per year, if 180 GW of breeders were deployed by 2025. This amount of breeders corresponds to the maximum deployment considered in NASAP for the 1978 high growth (see Volume III) of 910 GW of nuclear power output by 2025 in the United States.

Recent developments in the regulatory requirements for transporting of irradiated nuclear fuel could affect the assessment of transportation impacts. In June 1979, the Nuclear Regulatory Commission published for comment NUREG-0561 to provide interim guidance on "Physical Protection of Shipments of Irradiated Reactor Fuel." In accordance with this guide the Nuclear Regulatory Commission will require advance notification and approval of routes to be used in any shipment of irradiated fuel. It also requires that arrangements be made with law enforcement agencies along the proposed route to assure their response to an emergency or a call for assistance. Routes must be planned to avoid heavily populated areas.

Table 8. Annual Accidents, Injuries, and Fatalities
from Fuel-Cycle Transportation Activities
(per 1 GW of generating capacity)

	<u>Accidents</u>	<u>Injuries</u>	<u>Fatalities</u>
<u>Light-Water Reactor, Once-Through Cycle</u>			
Low-specific-activity materials	0.085	0.043	0.0026
High-specific-activity materials	<u>0.111</u>	<u>0.056</u>	<u>0.0033</u>
Total	0.196	0.099	0.0059
 <u>Liquid-Metal, Fast- Breeder Reactor</u>			
Low-specific-activity materials	0.050	0.027	0.0016
High-specific-activity materials	<u>0.064</u>	<u>0.042</u>	<u>0.0026</u>
Total	0.114	0.069	0.0042
 Difference between the light-water reactor once-through and the liquid-metal fast breeder reactor	0.082	0.030	0.0017

Resolution of the issues raised in NUREG-0561 will be required for all fuel cycles requiring transportation of irradiated fuel. Thus the impact is judged to be about the same for all fuel cycles under consideration, recognizing the fact that cycles involving recycle or spikants will have to meet the physical protection transportation requirements in shipping fuel both to and from the reactors.

4.2 ENVIRONMENTAL IMPACTS

The environmental impacts of a reactor and fuel-cycle system can be measured in broad terms according to natural resource use, potential public and occupational radiological exposures from fuel-cycle operations, radioactive wastes, and the amounts and type of major effluents from the various facilities required in the cycle. Natural resource use can be subdivided into land use, water use, and fuel use or energy input to the cycle. Major effluents can contain radioactivity, chemicals, and waste heat.

4.2.1 Natural Resource Use

Land Use

Land commitments may be classified as temporary or permanent. A temporary land commitment is for the life of the facility or succeeding facilities at the same location. After the abandonment or decommissioning of the facility, such committed land can be used for any purpose. A permanent commitment of land means that the land may not be released for use after facility shutdown and decommissioning.

Uranium mining is the largest contributor to land use among the fuel-cycle facilities and accounts for approximately 85 percent of the temporarily committed land. Together, uranium mining and milling account for approximately 95 percent of the permanently committed land. It follows, then, that those fuel cycles that minimize U_3O_8 and, therefore, uranium ore requirements will also result in minimum land use requirements for fuel-cycle facilities.

Land use requirements for an individual reactor and associated fuel cycle are in approximately the same ratio per unit of power as the relative lifetime requirements for U_3O_8 , if the average ore grade (that is, the U_3O_8 content per unit of ore) is the same over the lifetime of the alternative reactor. The annual land-use requirements of a nuclear-power system containing several types of reactors and associated fuel cycles will be determined by the fractions of the total electrical output supplied by each type of reactor and fuel cycle. The cumulative land use at any time will be directly related to the cumulative U_3O_8 use, which, in turn, depends on the deployment pattern for the alternative reactors and fuel cycles, and on their characteristics.

The lifetime requirements for U_3O_8 relative to the U_3O_8 requirement for the standard light-water reactor using an enrichment tails assay of 0.20 percent are presented in Table 9 for alternative fuel cycles.

The cumulative U_3O_8 usages were calculated for many scenarios of reactor fuel-cycle deployment in three nuclear growth projections which are described in Volume III. The cumulative U_3O_8 usage relative to the standard light-water reactor on the once-through cycle for the same nuclear growth is shown in Table 10 for some selected scenarios that cover the range of land use impacts. The maximum impact through 2025 is a reduction in relative use of about 30 to 35 percent when the liquid-metal fast-breeder reactor is introduced at an early date (2006).

The U.S. Atomic Energy Commission (Reference 21) estimated that the total temporary land commitment required by the fuel cycle to support a 1 GW standard design light-water reactor on the once-through cycle was about 750 acres and that the total permanent land commitment for this cycle, assuming a 30-year reactor life, was estimated as 140 acres. The annual ore requirement corresponding to these land-commitment estimates was 100,000 short tons/yr for an ore grade of 0.2 percent. A grade of 0.1 percent is more likely to be the average during the period through 2025 (see Volume III, section 2.2.1), and is assumed for this analysis, resulting in a temporary land commitment of 1,500 acres, and a permanent land commitment of 280 acres during a 30-year lifetime for a 1 GW standard light-water reactor on the once-through fuel cycle which uses about 5,717 short tons of U_3O_8 . The land commitments for a system of these reactors in operation in the years 2000 and 2025 are shown in Table 10 for the 1978 high-, median-, and low-nuclear growth projections discussed in Volume III.

The magnitude of the land use impacts can be estimated for alternative reactors and fuel cycles by using the data in Tables 10 and 11. The maximum impact through 2025, which occurs from early deployment of the liquid-metal fast-breeder reactor and light-water reactor fuel utilization improvements or recycle, is equivalent to a reduction of 35 percent in the high growth case, equal to about 400,000 acres of temporary commitment and 73,000 acres of permanent commitment. For comparison purposes, the Department of Energy Hanford Reservation in Washington State occupies about 400,000 acres.

Table 9. Relative Lifetime U_3O_8 Requirements for Alternative Fuel Cycles

<u>Reactor/Fuel Cycle</u>	<u>Relative Lifetime</u> <u>U_3O_8 Requirement</u>		<u>U_3O_8 Requirement</u>	
	<u>0.20% Tails Assay</u>	<u>0.10% Tails Assay</u>	<u>0.05% Tails Assay</u>	
Light-Water Reactor Standard, Once-Through	1.0	--	--	
Light-Water Reactor 15% Modification, Once-Through	0.85	0.74	0.69	
Light-Water Reactor 30% Modification, Once-Through	0.70	0.61	0.57	
Light-Water Reactor Standard Uranium-Plutonium, Recycle	0.60	0.52	0.49	
High-Temperature Gas-Cooled Reactor, Uranium-Thorium, Once-Through	0.67	0.57	0.55	
High-Temperature Gas-Cooled Reactor, Uranium-Thorium, Recycle	0.50	0.43	0.41	
Light-Water Breeder Reactor Prebreeder/Light-Water Breeder Reactor Using Enriched Uranium- Thorium for the Prebreeder and Uranium-Thorium Recycle	0.33	--	--	
Light-Water Breeder Reactor Prebreeder/Light-Water Breeder Reactor Using Plutonium-Thorium for the Prebreeder and Uranium- Thorium Recycle	0 ^a	--	--	
Fast-Breeder Reactor, Uranium- Plutonium, Recycle	0 ^b	--	--	

^aUntil depletion of the light-water reactor spent-fuel stockpile.

^bAssuming availability of plutonium from spent fuel for initial core inventory and depleted uranium from enrichment tails stockpile.

Table 10. Relative Cumulative U_3O_8 Usage for Various Scenarios for the Years 2000 and 2025

Scenario	1978 Low Growth		1978 Median Growth		1978 High Growth	
	2000	2025	2000	2025	2000	2025
<u>Once-Through</u>						
(Standard Light-Water Reactor, 0.20% Tails: millions of short tons)	(0.73)	(2.16)	(0.86)	(3.21)	(0.99)	(4.26)
15%-Improved Light-Water Reactor, 0.10% Tails deployed in 1990 to 2000 period in all reactors	0.92	0.79	0.91	0.78	0.90	0.77
30%-Improved Light-Water Reactor, 0.05% Tails deployed beginning ^a in 2000 in all new reactors	0.92	0.73	0.91	0.71	0.90	0.69
<u>Recycle</u>						
Light-Water Reactor-15% Improved, Self-Generated Recycle beginning in 1990	--	--	0.87	0.74	0.88	0.74
Liquid-Metal Fast-Breeder Reactor, 2006 introduction ^b	--	--	0.91	0.67	0.90	0.65
Liquid-Metal Fast-Breeder Reactor, 2014 introduction ^a	--	--	0.91	0.76	--	--

^aOther reactors in system are 15%-improved light-water reactors, 0.05% tails assay.

^bOther reactors in system are 15%-improved light water reactors, 0.10% tails assay.

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Table 11. Land Commitments for Standard Design Light-Water Reactor on the Once-Through Fuel Cycle (acres)

Nuclear Growth Scenario	Installed Capacity GW	Land Commitment Year 2000		Installed Capacity GW	Land Commitment Year 2025	
		Temporary	Permanent		Temporary	Permanent
High	395	260,000	48,000	910	1,118,000	209,000
Median	325	226,000	42,000	615	842,000	157,000
Low	255	192,000	36,000	320	567,000	106,000

Reactor and fuel-cycle systems using thorium have less demand for U_3O_8 than do nonbreeding systems having high U-235 demand. The use of ThO_2 , however, necessitates mining and milling of thorium ore. Because of the higher ore grades in thorium mining, the land used for extraction of a unit of U_3O_8 is about three to four times more than that required for a unit of ThO_2 . Because uranium for use in a light-water reactor would normally be enriched to 3 percent--requiring about six times as much feed as product--and ThO_2 is used directly in the uranium/thorium fuels, the mining of ThO_2 theoretically would require about 18 to 24 times less land commitment. Furthermore, the delayed recycling of ThO_2 would further reduce thorium ore mining requirements by several orders of magnitude. Thus, land use for ThO_2 mining would be relatively insignificant for all the fuel cycle alternatives as compared to that used for uranium ore mining.

Water Use

Although alternative fuel cycles have different water requirements, the magnitude of the differences is not significant in evaluating alternative reactor-fuel cycle combinations.

For the light-water reactor once-through cycle, 98 percent of the water use in fuel cycle facilities is required for the gaseous diffusion enrichment process. If the more energy-efficient gas centrifuge process were used, fuel cycle water use would be reduced by about 85 percent, but enrichment would still account for 85 percent of the water use in the fuel cycle, exclusive of

reactor needs. Even with gaseous-diffusion enrichment, water requirements for fuel-cycle facilities supporting the once-through light-water reactor fuel cycle is a small fraction (less than 2 percent) of the water requirement for the reactor itself. Water is needed principally for rejecting waste heat from the reactors.

Energy Use

Energy use requirements among fuel-cycle alternatives are in approximately the same ratio as relative lifetime separative work requirements because the enrichment process is by far the most energy consuming activity in the fuel cycle. However, assuming that this energy for the nuclear fuel-cycle activities comes from coal-fired power plants, energy use represented by coal consumption for any nuclear fuel-cycle alternative would be less than about 3 percent of national coal consumption expected in the year 2000. Thus, energy use is not a discriminator in selecting among alternative fuel cycles, although the least energy consuming of the alternative fuel cycles would be the fast-breeder reactor fuel cycles.

The electrical energy required for producing one unit of separative work with the gaseous-diffusion process requires burning about one metric ton (t) of coal or its equivalent. If 325 GW of light-water reactors are operating on the once-through cycle in the year 2000, the energy for fuel enrichment (by gaseous diffusion) would require about 38×10^6 t of coal. Coal use would be 5 percent of this quantity if all enrichment were done by centrifuge. These figures can be compared with the estimated coal consumption for central-station power generation of 6×10^6 t/yr in 1978, which is expected to double by the year 2000. On this basis, the energy consumed in the enrichment process would be about 3 percent of the national coal consumption in the year 2000 with gaseous diffusion. If the gaseous centrifuge were used for future plant additions the fraction would be reduced further.

Fast-breeder options require no enrichment if the plutonium for the initial core is obtained from spent fuel and would save 1.1×10^5 t of coal (or equivalent) per GWyr compared with the once-through light-water reactor cycles.

4.2.2 Radiological Exposures

The major contributors to both occupational and public radiation exposure in nuclear fuel cycles are mining and milling and chemical reprocessing--the

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latter is applicable to recycle alternatives only. If future emissions of reprocessing off-gases are controlled to an appreciable degree and if radon emissions from mining and milling tailings are appreciably reduced, the total radiological impacts for all fuel-cycle options will be quite similar. Current control technology is capable of being upgraded in both areas so that it is feasible for all alternative fuel cycles examined in this assessment to meet equivalent radiation exposure criteria.

Under normal operating conditions, most of the radiation dose to the population would come from radon effluents from mining and milling of uranium and from the atmospheric releases of radionuclides from fuel reprocessing plants. Table 11 gives total-body person-rem estimates from Reference 22 for one GWyr for all stages of both once through and a uranium/plutonium recycle light-water reactor fuel cycle. For the once-through cycle, mining and milling account for 61 percent of the total radiological impact to the plant workers and the general population, and reactors, 35 percent. For the recycle case, a 20 percent reduction occurs in dose from mining and milling, but is more than offset by the additional dose from reprocessing. As shown in Table 12, the net effect of recycle in the light-water reactor is an increase in total radiological impact of 20 percent. Analyses of fuel cycles of light-water breeder reactors (Reference 23), liquid-metal fast-breeder reactors (Reference 24), and high-temperature gas-cooled reactors (Reference 11) also indicates that fuel reprocessing contributes a large percentage of the total radiological impact for these cycles. However, in breeder cycles the radiological impacts from mining and milling will be small compared with those in Table 12, but the exposures from reprocessing will increase.

For a scenario involving 400-GW-installed nuclear capacity, the total radiological impact for light-water reactor recycle would be 0.9×10^6 person-rem, or about 2 percent of the total population dose from natural background. The incremental dose from recycle is 20 percent of this amount and thus represents less than 0.5 percent of the total population dose. This difference can be reduced or eliminated by use of improved emission control technology.

Estimated radiological impacts for the various reactor concepts are similar. Heavy-water reactors with relatively large tritium releases may have slightly greater radiological impact than other reactors. Radiation exposure to individuals is strongly regulated and radiation dose standards exist for fuel-cycle facilities (except for tritium and C-14 from reprocessing). However, gaseous emissions of radon from mines and mills are not Federally regulated in the sense of population dose limits or standards, although state regulations are in use. The Department of Energy has initiated a remedial action program to reduce population exposures from old mill tailings that have been abandoned or distributed. Krypton-85, iodine, and actinide emissions are

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Table 12. Estimates of 50-Year Cumulative Dose from
Light-Water Reactor Fuel-Cycle Components
(total-body person-rem per GWyr)^a

Fuel Cycle Component	Light-Water Reactor Once-Through		Light-Water Reactor Uranium and Plutonium Recycle	
	General		General	
	Population	Occupational	Population	Occupational
Mining	628	256	507	206
Milling	122	119	99	96
Conversion	9.5	1	7.8	9.8
Enrichment	0.02	0.7	0.02	0.7
Fabrication, UO ₂	0.6	11.6	0.53	10.3
Fabrication, Mixed Oxide (uranium- plutonium) Fuels	0	0	0.06	4.8
Reactor	75.8	562	76.8	562
Fuel Storage	0.009	3.6	0.003	1
Reprocessing	0	0	354	25.5
Transportation	0.1	0.36	0.16	0.84
Waste Management	0.002	0.35	0.002	0.32
Foreign Exposure from U.S. Releases	52	0	272	0
	888	954	1317	917
Total Person-Rem	1842		2234	

^a Source: U.S. Nuclear Regulatory Commission. Final Generic Environmental Statement in the Use of Recycle Plutonium in Mixed Oxide Fuel in Light-Water-Cooled Reactors, NUREG-0002, Volume 3, August 1976.

now regulated for the light-water reactor-uranium fuel cycle; release limits become effective in 1983.

Breeder options with no U_{308} requirements may have the least radiological impact. Assuming successful containment of tritium and C-14 in reprocessing, breeder and light-water reactor recycle options should have less total radiological impact than once-through systems because the lower U_{308} requirements would reduce radon emissions from mining and milling (unless these are also reduced by future containment and control). However, a study of reprocessing and refabrication of liquid-metal fast-breeder reactor fuels showed significant potential increases in population dose from advanced carbide and nitride fuels, as compared with mixed-oxide fuels, as a result of greater emissions of C-14 associated with carbide and nitride fuel reprocessing (Reference 25).

4.2.3 Radioactive Waste Disposal

The safe handling and long-term isolation of radioactive wastes is of less concern for the plutonium recycle systems than for the once-through systems because the latter systems require disposal of large amounts of plutonium which is much longer-lived than fission products or other radioactive isotopes. The requirement for very long isolation (thousands of years) for waste containing large amounts of plutonium is a significant difference among fuel cycles.

The major impacts of waste disposal lie in (1) land use, (2) radiological exposures, (3) transportation requirements, and (4) the time period over which surveillance may be required (The time is related to the "lifetime" of a perceived hazard.). The first three have been discussed in Sections 4.2.1, 4.2.2 and 4.1.7, respectively. These discussions show that (1) the impact of waste disposal in the first two subject areas is a negligible part of the total impact; and (2) in the third area, the impact of the whole fuel cycle is a negligible part of the whole U.S. transportation industry. Considering only these three aspects, the different impacts among the alternative fuel cycles caused by differences in waste disposal are not a basis for preference of an option.

Regarding the fourth impact of long-term effects, a major difference arises in the postirradiation operations of the fuel cycle in comparing once-through systems with recycle and breeder systems because of the relatively long half-life of plutonium compared with the other (fission product) wastes. The quantity of fission products produced per GWyr is less for higher thermal efficiency systems such as breeders (the generation of fission products is

roughly proportional to the thermal energy produced) resulting in a maximum difference of about 45 percent between the least efficient (the heavy-water reactor) and the most efficient (the high-temperature gas cooled direct cycle reactor). The differences in impacts caused by the differences in amount of fission products generated, however, are not the controlling issue because of the very long half-life of plutonium (e.g., Pu-239 half-life is about 24390 years) compared with the half-life of such fission products as Sr-90 (29 years) or Cs-137 (30 years). That is, a difference in quantity of fission products results in different impacts in the first three areas identified above, and this report has concluded that those differences are not a basis for discriminating among fuel cycles.

It has been concluded that the environmental analysis of total waste management systems, including land-use requirements, involved in conventional geologic disposal showed no clear environmentally based choice favoring one or the other fuel-cycle option or repository medium (Reference 20). The health effects after closure of a repository are assumed to be zero in the case of design basis performance. Recycle and breeder fuel cycles both use much of the plutonium in reactors and generate less of these as wastes than do once-through systems; thus, the result may be of less concern for safe long-term, geologic storage. The subject of the disposal of plutonium-containing waste and other actinide waste (e.g. neptunium, americium, and curium) is under study. One alternative for reducing the long-term hazard is separation and burnup of the long-lived isotopes. An Oak Ridge National Laboratory report on the incentives for pursuing this alternative is scheduled for completion in March 1980.

4.2.4 Nonradioactive Effluents

Chemicals

Except for coal combustion products from the generation of electricity for the fuel cycle, the effect of differences between alternate fuel cycles on other chemical effluents does not appear big enough to be a discriminator in choosing a fuel cycle. Because the nuclear fuel cycle is projected to consume less than 3 percent of the total coal used in the year 2000 (see Section 4.2.1) these differences are insignificant.

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The major chemical effluents associated with operating fuel-cycle facilities are:

1. Combustion products from coal used to produce electricity--primarily for the enrichment process.
2. Fluorine gases in the conversion, enrichment, and fabrication processes.
3. Nitrogen oxides (NO_x) from combustion of natural gas for process heat in milling and conversion.
4. Chemicals, such as fluorides and nitrates in liquid effluents from conversion, enrichment, fabrication, and reprocessing plants.

Coal combustion products would be minimized in fuel cycles not requiring enrichment or by using nuclear power to generate the electricity for enrichment. Fluorine releases associated with about 400 GW of nuclear capacity were estimated at less than 0.3 percent of the total annual fluorine releases from U.S. industries in 1974. Keeping releases of fluorine, NO_x , and other chemical effluents from fuel cycle facilities below environmentally acceptable levels does not appear to be a major problem.

Waste Heat

Waste heat from fuel-cycle facilities is not a meaningful criterion for discriminating among alternate fuel cycles because the reactors are responsible for more than 90 percent of the waste heat to the environment. For the light-water reactor once-through cycle, waste heat associated with enrichment is less than 7 percent of the waste heat from the nuclear power plants. Other fuel-cycle operations contribute less than 0.4 percent of the total waste heat. Differences in waste heat associated with differences in enrichment requirements for alternate fuel cycles make relatively small differences in the overall waste heat loads including the reactors.

4.2.5 Nontechnical Issues in the Licensing Process

In addition to the technical issues discussed in the preceding sections, other issues are important to the schedule for licensing the fuel cycle.

A few examples of such issues are:

- o Proliferation risk. The risks of the new process must be compared with existing risks, and standards for dealing with these risks must be put into regulations. For example, shipments of plutonium fuel might have to be diluted with uranium, and spiking might be required. The possibility of such requirements introduces a high risk to the license applicant. Preferably, all these regulations would apply to the radioactive pilot plants (the research, development, and demonstration program) so that the feasibility of the regulatory approach could be demonstrated.
- o Societal needs for power. Future perceptions that the short supply of coal, uranium, and other fuel resources is limiting societal growth or welfare could induce favorable consideration of the licensing of fuel recycle facilities. Societal needs for power would be balanced against other concerns such as proliferation risk.
- o Local and regional issues. Local and regional participation in site selection and regulation of nuclear fuel cycle sites, especially waste sites, is an open issue that creates institutional barriers to the authorization of sites for full-scale operations (and even for research, development, and demonstration activities).
- o Societal preferences for long-term energy options. A societal preference for fission breeders would imply increased societal commitments to long-term management of nuclear sites and wastes. A very broad review of the issues over a period of decades is underway. Society may wish to defer further commitment to fission reactors until further experience with the existing commitment is obtained.

4.3 LICENSING-TIME ASSESSMENT

The following estimates for licensing time are based primarily on the estimated time required for resolution of technical issues. Other societal issues that enter into the licensing process could alter the overall implementation schedule.

The only fuel cycle for which there has been substantial experience in establishing a licensing and regulatory basis is the once-through light-water

reactor cycle. The licensing time assessments for alternative fuel cycles are judgments of the time required to achieve a comparable basis.

4.3.1 Once-Through Light-Water Reactor

The once-through light-water reactor cycle with interim storage of spent fuel at reactor pools is currently licensed. Several applications from utilities are being reviewed by the Nuclear Regulatory Commission for expanding storage at reactor pools by reracking (rearranging and adding to existing racks) to increase the density of the storage. However, additional storage will be needed for some reactors by 1983. The Department of Energy is considering several options for providing additional interim storage until decisions are made on, and facilities are available for, permanent disposal or reprocessing of spent fuel. One option for interim storage is the use and expansion of one or more of the existing facilities at Morris, Illinois; Barnwell, South Carolina; or West Valley, New York. An Environmental Impact Statement for a specific site could be completed in about 1 year, and the Safety Analysis Report could be submitted in about 2 years for use and expansion of the existing facilities. These facilities could begin operation in about 3 to 4 years. Another Department of Energy option is to build a new interim storage facility. A Generic Environmental Impact Statement for such a facility will be completed by early 1980. A site-specific Environmental Report and Safety Analysis Report for a new storage facility could be submitted within 2 years, and operation of the facility could begin 4 to 5 years later.

If spent-fuel disposal in a geologic repository becomes a requirement of the once-through light-water reactor cycle, the technology must be developed and followed by the siting, licensing, and construction of one or more large-scale facilities. A Federal Interagency Review Group submitted a report on Nuclear Waste Management to the President in March 1979 (Reference 19), which made recommendations for the conduct of the high-level waste-disposal program, including spent fuel disposal. One issue concerning the establishment of high-level waste repositories, which was referred to the President for determination of policy, is the question of how many sites (and geologic media types) should be evaluated before an attempt is made to seek licensing approval to construct a repository. One option is to proceed with a repository based on investigating only a few geologic media (e.g., salt domes and basalt) at a few sites. This choice would result in submittal of a site-specific Safety Analysis Report and Environmental Report to the Nuclear Regulatory Commission by 1982; initial operation of the repository is considered to be feasible in 1990. Another option is to acquire data and investigate additional geologic media (e.g., granites and shales). This choice would lead to submittal of a site-specific licensing application in 1985 and start of operation in 1994.

4.3.2 Light-Water Reactor Recycle

Licensing of recycle facilities for the light-water reactor system may require from 10 to 20 years, depending on requirements regarding recycle fuel characteristics and resolution of high-level waste-disposal problems.

The technology for recycle of plutonium in light-water reactors has been developed and demonstrated in pilot-scale licensed facilities. Licensing of a commercial-scale reprocessing plant could be accomplished most rapidly in the existing Allied General Nuclear Services reprocessing facility at Barnwell. Startup at Barnwell would require design and operation of the following:

- o Facilities to convert plutonium nitrate to a solid plutonium product.
- o Facilities to handle and package various radioactive wastes containing transuranium elements for transfer to a Federal repository.
- o Facilities to convert the high-level and intermediate-level radioactive liquid wastes to solid form for transfer to a Federal repository.

Design of the high-level waste solidification and plutonium-conversion facilities at Barnwell, which had progressed to a preliminary design and planning stage, was suspended in 1977 pending (1) the resolution of issues about the wide-scale use of mixed-oxide fuel in light-water reactors and (2) the definition of criteria and establishment of charges for the acceptance of various radioactive wastes at Federal repositories.

Estimates have been made (Reference 27) that (1) when the issues surrounding recycle and waste disposal are resolved, and (2) should the decisions allow completion of the as-designed Barnwell facility, final designs for high-level waste solidification and plutonium conversion facilities acceptable to the Nuclear Regulatory Commission would require 1.5 to 5.5 years (1.5 years for as-designed PUREX reprocessing and up to 5.5 years for modified reprocessing potentially required by U.S. policy). In addition, an estimated 2 years for licensing review and hearings may be required for the operating license. Allowing an additional 2.5 years for construction of facilities over and above the upper limit of this estimate (some construction is assumed to proceed during the late phases of the final design period of the 5.5-year schedule) implies that the Barnwell facility could begin operating to produce a purified

plutonium-oxide product and UF_6 for recycle 10 years after a decision to proceed.

Assuming that the issues concerning disposal of radioactive wastes generated from reprocessing must be resolved for a recycle facility to be licensed to operate, a 10-year time to begin Barnwell operations would be compatible with the time to start up a Federal repository for high-level waste for the early repository option (see above discussion of the light-water reactor once-through). The later repository option would add 4 years to the time when the Barnwell facility could begin operation. Even with this later repository option, Barnwell could operate in 10 years if high-level waste could be stored on-site as a liquid or a solid pending availability of the repository 4 years later.

If nonproliferation concerns dictate modifications or additions to the basic PUREX process (such as coprocessing or spiking), design, licensing, and construction of reprocessing and mixed-oxide fabrication facilities might be delayed beyond 10 years. One estimate (Reference 27) is that the Barnwell Purex process redesign would take 1.5 to 4 years, the range depending primarily on whether only slight or major changes in the current design and processes are required. Only if this effort were coincident with design of the new facilities would the 10-year schedule be possible.

In the event that either spiking or use of low-decontaminated recycle fuel is required to meet nonproliferation concerns, new remote fabrication technology would be required. Facilities using this technology would require possibly 5 to 10 years of research and development followed by perhaps 10 years of licensing, hearings, construction, and operational check-out of the first fabrication plant.

4.3.3 Uranium/Plutonium Cycle in Breeders

An estimated 20 years or more will be required from the decision to proceed to the operation of licensed commercial, large-size plants for this fuel cycle.

Substantial portions of the light-water reactor recycle technology are expected to be applicable for uranium/plutonium recycle in breeders. Differences in fission product concentrations, heat-generation rates, fissile concentrations, and cladding material will require additional research and development. The removal of sodium from the spent fuel requires some reprocessing head-end work. The use of carbide fuel in the liquid-metal fast-breeder reactor rather than oxide fuel as is used in the light-water

reactor would require substantial additional research, development, and demonstration. Demonstration of liquid-metal fast-breeder reactor recycle facilities are expected to be required prior to commercial plant licensing whether or not commercial light-water reactor recycle occurs because of the unique aspects of breeder fuels. The research and development for this technology is now underway. It is estimated that the technology could be demonstrated by the mid-1990's so that a commercial plant design could be defined and a construction license application submitted to the Nuclear Regulatory Commission at that time. Operation of the commercial fuel-cycle plants could begin soon after the year 2000. However, the commercial fuel-cycle facilities would probably be delayed 5 to 10 years after this time until a sufficient number of breeder reactors were in operation to provide the amounts of spent fuel needed for a commercial-size reprocessing plant (see Volume IV).

4.3.4 Uranium/Thorium Cycle

An estimated 25 years or more will be required from a decision to proceed to the operation of licensed commercial, large-size plants for this fuel cycle.

The uranium/thorium recycle alternatives are less developed than are those for the uranium/plutonium fuel cycle and have difficult technology development requirements. THOREX reprocessing and remote fabrication of fuel containing high gamma-ray activity must be demonstrated; difficult safeguards in process accounting problems must be overcome. Although some work has been started for the light-water breeder reactor "proof-of-breeding" tests, an adequate recycle technology development program for light-water prebreeder and breeder reactor fuels is not underway. Small pilot-scale facilities must be built, followed by demonstration facilities for reprocessing and remote fuel fabrication. This sequence of facilities would require at least 15 years to progress to the stage where the technology could be demonstrated and a design for commercial facilities could be submitted to the Nuclear Regulatory Commission. Assuming three years for review prior to the construction license, commercial operation could start about 9 to 10 years after the license application.

The high-temperature gas-cooled reactor fuel would present unique research, development, and demonstration needs. However, some technology development has been done, a pilot facility has been initiated, and the program is ongoing. Thus, commercial facilities for recycle could probably be in operation about the same time as those for recycle of light-water reactor fuels.

5. COLOCATED FACILITIES

Fuel service centers were considered in the NASAP assessment for colocating facilities for the handling and use of sensitive fuel-cycle materials, such as plutonium, in order to improve safeguards and control. Two types of fuel service centers are considered in this assessment: power-generating centers, in which reactors are colocated with fuel-recycling facilities; and centers without power generation, in which only fuel-recycling facilities are colocated (References 28 and 29). The power-generating centers contain onsite reactors and fuel-recycling facilities (i.e., reprocessing and fabrication facilities) to service the onsite reactors and associated offsite reactors. The centers without power generation contain fuel recycle or storage facilities to service offsite reactors.

In general, the safety considerations and environmental impact of fuel service centers without power generation are not significantly different from those of dispersed fuel-cycle facilities having the same capacity. But the impacts of power-generating service centers can differ substantially from those of the dispersed facilities.

Certain categories of environmental impacts, such as water consumption, are additive in nature; that is, the effects of colocated versus dispersed systems involves only the sum of water requirements for equivalent independent reactors, reprocessing plants, and other onsite facilities.

Another category of impact involves synergistic effects, which add more than the sum of the individual reactor effects. Heat and water-vapor loads on a locale containing 40 reactors dissipating perhaps 100,000 MW of heat within a relatively small area would result in environmental modifications during stable meteorological conditions, whereas one individual reactor dissipating its heat load to the same locale would rarely produce significant effects.

A third category also exists, for which the effects of the colocated facility may be equal to, or less than, the sum of the effects of individual facilities having the same capacity. Radiological dose commitments from the radio-nuclides released during normal operation of nuclear facilities are quite sensitive to variations in source-receptor distance, and adequate sizing of a colocated facility can result in doses to a maximally-exposed individual that are similar in magnitude to those from a single reprocessing plant or reactor. Another example of such an impact might be the effect of cooling-tower drift,

in which the large size of a colocation site would result in retention onsite of a larger fraction of drift than would be the case for dispersed reactors.

5.1 RADIOLOGICAL RELEASES FROM NORMAL OPERATION OR FROM ACCIDENTS

5.1.1 Power-Generating Fuel Service Centers

Normal Operation

For fuel service centers with onsite power generation (energy centers), the limiting dose to be considered in normal operation, for both reactors and reprocessing plants is from emission of radioactive iodine. For the power-generating service centers, the offsite radiological impact depends on specific site and climate characteristics. A taller exhaust stack or more stringent off-gas cleanup can be employed at the reprocessing plant to mitigate the potential radiation-dose problem, if necessary. The reactor land areas and exclusion distances are made large so that offsite radiological impact is acceptable compared with dispersed facilities (Reference 30).

Although the incremental radiation exposure to individual construction workers and to operating personnel working near other operating facilities in fuel service centers may be somewhat higher than in plants at dispersed sites, the total impact would be too small to become a negating factor to the use of service centers. This is because normal operations contribute only a small fraction of the background radiation outside of the reactor plants.

Accidents

The accidents that might occur in power plants and reprocessing plants range from incidents of very low consequence and moderate frequency to accidents with potentially great consequences (including design-basis accidents and more severe accidents involving a number of successive failures) but with very little probability of occurrence (low-frequency accidents) either at dispersed sites or at fuel service centers. For very low-frequency accidents with serious consequences, the potential radiological impact to offsite areas is not worse with fuel service centers because the center can be designed so that the probability of two such accidents happening simultaneously would be negligibly small and because the land areas and exclusion distances are much larger than those at dispersed sites. The long-term biological effects to the distant offsite population from the low-consequence reactor accidents would be

similar to, or possibly much smaller than those from the same reactors at dispersed sites, because these effects result primarily from low-level radioactivity spread over a wide area.

The risk to the local population of a single design-basis accident involving a large release of radioactivity from service centers that have power-generating reactors is somewhat higher than from dispersed reactors because the probability of such an accident, although low, is somewhat higher at power-generating centers, where there are many reactors at one site.

A significant reduction in the frequency and severity of transportation accidents is possible for fuel service centers compared with the equivalent number of dispersed reactors and fuel-cycle facilities. The onsite spent fuel and recycle fuel shipments would be over much shorter distances, compared with dispersed siting, and would use a special, single-purpose transport system that would not be used for transport of other hazardous materials (e.g., gasoline and chemicals).

Severe natural disasters at a fuel service center site (e.g., an earthquake of greater magnitude than a safe-shutdown earthquake) could result in worse consequences than the same event at one of a group of dispersed sites because of the greater aggregation of reactors. The likelihood that such an earthquake would affect at least one dispersed reactor site in the region is nevertheless much higher for dispersed sites than for service centers because the number of sites is larger. A comparison of these two types of probabilities must be site specific, and no conclusion can be drawn here.

Damage to fuel service center reactors or plants could also arise from terrorist or sabotage activities. However, because the primary purpose for service centers is to colocate sensitive fuel-cycle and reactor operations for better safeguards, the probability of such an event occurring at a fuel service center reactor or plant is expected to be less than at dispersed facilities.

5.1.2 Fuel Service Centers Without Power Generation

In fuel service centers without power generation, the major radiological impacts are from routine operations and from potential accidental radioactivity releases from reprocessing plants. In both cases, the long-term risk to the distant offsite population would not be greater, and would possibly be less, than for dispersed siting of fuel-cycle facilities, because the

size of the reprocessing plant would be the same as the plant at the dispersed site, and because the site would be at least as large. The onsite effect of radioactivity release from a severe accident or from a natural disaster would be somewhat worse than that at dispersed sites because personnel in other onsite facilities (such as the fabrication plant) could be exposed along with those at the reprocessing plant.

As mentioned previously, fuel service centers show the likelihood of potentially fewer transportation accidents, compared with dispersed facilities. This difference is not significant for centers without power generation, except that the numbers and distances of shipments between the reprocessing plant and the fuel fabrication plant would be less than those for dispersed siting of these facilities.

5.2 LAND-USE

5.2.1 Power-Generating Fuel Service Centers

Power-generating fuel service centers generally require more land than dispersed power plants, because all of the land needed for radiological or heat dispersal is at one location. Land requirements would be very large for power-generating service centers, e.g., as much as 69 square miles for a site with 25 GW. The large area suggests that fuel service centers would be located in areas of low-value land, at some distance from the metropolitan areas that would use the power. A remotely-located fuel service center would require longer and possibly larger power-transmission corridors, but quantitative comparison cannot be made without analysis of particular sites. Use of transmission-corridor land is usually limited to animal grazing and some farming.

5.2.2 Fuel Service Centers Without Power Generation

The land required for a fuel service center without power generation would be about the same as the land required for a reprocessing plant of similar capacity, because the exclusion area would be governed by the potential radiological releases during an accident at the reprocessing facility.

5.3 WATER USE

5.3.1 Power-Generating Fuel Service Centers

The water use at a fuel service center is equivalent to that at dispersed sites, except that the former is highly localized. For equal generating capacity, about the same water use will result. For instance, a service center with 25 GW of onsite power would consume about 220,000 gal/min, whereas a dispersed site with 2.5 GW of power would consume 1/10th as much. However, the impact of water use at the service center may be spread over a limited stretch of a single river or in a single watershed, whereas the impact of the dispersed power plants may involve many water bodies and several watersheds.

5.3.2 Fuel Service Centers Without Power Generation

Water use for colocated fuel recycle facilities is not significantly different from that of dispersed facilities having the same capacity.

5.4 ECOLOGICAL IMPACTS

5.4.1 Power-Generating Fuel Service Centers

The effect of a power-generating fuel service center on aquatic and terrestrial ecology is potentially much greater than that of dispersed reactor sites. The greater heat-dissipation and land-use requirements of grouping many facilities on one site and the larger scale and much longer duration of the construction period are the major causes of these potentially increased impacts. The magnitude of these impacts will be dependent on the particular site, the design features of the fuel service center, and the steps taken to lessen the impacts. For instance, if the site has few species, if water use is minimized in the design (e.g., use of wastewater or dry cooling towers), and if species are protected or transferred, the impacts can be minimized.

At a power-generating fuel service center, the disruption of terrestrial habitats covers a larger area. Because the disruption takes place over a construction period of 20 to 25 years, 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 power-generating service center will provide a large exclusion area for wildlife use.

Electric-field effects (from power lines) on bird habitats and migration patterns are greater for energy centers than are dispersed facilities because of the concentrated impact of transmission lines in the areas around the power-generating fuel service center.

Prolonged disruption or permanent loss of habitat and destruction of aquatic life can result from (1) changes in physical land features, (2) loss of vegetation; and (3) soil erosion, sedimentation, and water contamination by dredging operations at the site, at water intakes and discharges, and at power-transmission corridors during the several decades required for constructing a power-generating fuel service center. The ecological impact is more concentrated than that at dispersed sites and is also recurring; the exposed ecosystems therefore have less opportunity to recover between disruptions.

The destruction of aquatic life or habitat alteration and loss caused by water withdrawal, discharge, and warming can be controlled so that they are not much worse at a power-generating service center than they are 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 those at dispersed sites, depending on both site characteristics and the nature of the weather changes. The effects could include reduction in crop yields and disruption of wildlife balances.

At each energy center construction stage, new toxic and chemical materials are released or resuspended and result in degraded water quality. The concentrated adverse impact on water quality from the releases of chemicals (e.g., chlorine) during the operation of large energy centers could be mitigated by treatment or control of discharges.

5.4.2 Fuel Service Centers Without Power Generation

Ecological impacts from this type of center would not be significantly different from those of dispersed fuel-recycling facilities because the magnitude and duration of the construction activities are not significantly different, and because water use is comparable between the two.

5.5 SOCIOECONOMIC IMPACTS

5.5.1 Power-Generating Fuel Service Centers

In comparing the socioeconomic impact of power-generating service centers to an equivalent number of dispersed facilities, several factors must be considered, including: total center size, relative location to urban or rural areas, and time schedule for construction of each of the several facilities. A conclusion that the impact of power-generating service centers is less, or greater, than that from dispersed sites will depend on an analysis of the specific factors identified above, rather than on the basis of intrinsic features of either option.

A critical determinant of the effect a large power-generating fuel service center will have upon a community is the size of the labor force required for an extended period for construction of the facilities. The degree of impact also depends on the size of the existing population and on the infrastructure of the community before construction is begun: small communities are subject to the greatest change.

An influx of labor force into the service center area causes a great demand for public services, changes in cultural and political institutions, and growth in the local economy. Institutional factors also exist that are related to the ability of the existing tax and legal structure to handle such a high concentration of economic activity. Each of these issues is discussed below.

The construction of a twin-unit nuclear power reactor requires a peak labor force of about 4000 workers. A power-generating fuel service center with 20 reactors may have a peak labor force of between 10,000 and 20,000, depending on the startup schedule of the units, among other factors. At a manpower level of 10,000 workers, the local population increase would be from 33,000 to 63,000, the lower number corresponding to a more developed location (e.g., near a major metropolitan area) and the higher number corresponding to an underdeveloped area (remote location).

Metropolitan areas would be less affected, since a construction force would be more readily available without major disruptions to the community. Such locations are typically objectionable, however, because of adverse public reaction, aesthetics, and climatic and radiological impacts. On the other hand, locating a large energy center in isolated, rural areas would have

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a much greater initial impact. The rapid influx of a substantially increased force to an area would establish a new or much larger community, which would likely be disruptive to the local population, government and environment.

Rural areas are generally unable to provide highly developed private and public services (e.g., housing, schools, recreation, transportation, and police and fire protection). Therefore, the new demands for services on the host community or region would be substantial and would require front-end capital outlays for public facilities and local private investments. For a large center, the incentives for both public and private long-term investment would be greater than for options with shorter lifetimes, because 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 energy centers. Hence, in this regard a large center would be better than dispersed sites, where work forces and their impacts are transient.

New social values and modernizing forces undoubtedly contribute to cultural, organizational, and political disruptions in the host community as a result of the massive population inflow needed at rural and isolated areas.

However, only one locale would be affected by an energy center, and any adverse social impacts might not be much greater than they would be on a community associated with a single dispersed site. Socially, therefore, the service center might 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 could conceivably be mitigated by the expectation of greater employment opportunities and by the provision for more and better services to the residents in the vicinity of an energy center, on the premise that the center would promote steady, long-term community growth. The fact that a large center could trigger and then sustain industrial and economic growth in an underdeveloped rural area should be a consideration in siting. Such growth would, however, call for regional cooperation and planning in a concerted effort with state and Federal governments.

Local property taxation and its potential political problems are major economic issues in fuel service-center siting. The centers would generate vast tax revenues that, under conventional property-tax laws in the United States, would accrue to the local jurisdiction. The impact would be more severe than it would be with dispersed sites. Lack of tax-revenue redistribution would create gross fiscal inequities, especially for surrounding jurisdictions permanently affected by the work-force influx, and it could become a statewide

political issue and require overhauling state tax codes and new development policies.

The tax issue, the capital-outlay programs for public services, and the application of laws and ordinances would probably be broader and more complex for large power-generating service centers with multinational participation and strong government involvement or ownership. In view of the near-monopoly control over most phases of foreign fuel cycles by their governments, such strong governmental involvement could be reasonably expected if the energy center were internationally owned.

In addition to tax law inequities, certain legal constraints may impede the development of a large power-generating service center. Among the issues that will undoubtedly be raised are antitrust concerns over a large concentration of power-producing facilities and increased concern over waste-disposal legislation and plant siting, local real estate zoning, and environmental laws.

Finally, a facility located in a remote area may alter the aesthetic appearance of the region. Because of its size, it may also occupy land potentially valuable from an archaeological viewpoint. These issues need to be dealt with for particular sites, and the solutions must be acceptable to the public.

5.5.2 Fuel Service Centers Without Power Generation

The same issues discussed under power-generating fuel service centers arise in considering a center without power reactors. However, if no power reactors are present, the relative impacts are much reduced because (1) the number of construction workers is substantially lower; (2) the operating labor force is roughly equal to the construction labor force, which has a stabilizing influence on the community; (3) the concentration of economic power is less; and (4) the land, water, and other environmental impacts are reduced.

The socioeconomic impact of a center without power generation is not much different from the impact of locating the fuel-cycle facilities at dispersed sites. In fact, assuming that the impacts of centers without power generation are manageable, it has been proposed that the centers be located in more remote or rural areas to benefit the locale and provide economic growth that would otherwise be unavailable to the region. Such growth would have to be carefully planned and controlled to achieve this benefit without introducing major offsetting disadvantages.

5.6 SAFEGUARDS CONSIDERATIONS

5.6.1 Power-Generating Fuel Service Centers

The design and operation of the power-generating fuel service centers should enhance safeguardability because the fuel cycle plants that produce or handle strategic special nuclear materials (plutonium, U-233, or uranium enriched to 20 percent more in U-235) and the power reactors that use them as fuel are located at a single, protected site, eliminating offsite transportation. This is in contrast to the extensive shipment requirements of dispersed power reactors and fuel-cycle-support plants. The centers clearly alleviate important regulatory concerns over the shipping of large quantities of strategic special nuclear materials.

On the other hand, a power-generating fuel service center requires extensive multiple physical-protection barriers around individual facilities, as well as the entire center. These barriers are also required around multiple and diverse material-accountability installations inside the site. The latter must be integrated into a centralized control system to ensure that the safeguards are effective against potential threats from both outside and inside. These requirements imply two primary concerns: (1) assignment of safeguards responsibility and (2) division of physical-protection responsibility between Government and the center personnel. Application of full-scope International Atomic Energy Agency safeguards to all fuel service center activities has been proposed as the most sensible institutional solution, and specific institutional arrangements or international agreements among the participating countries are recognized as being required to ensure effective safeguards in fuel service centers.

5.6.2 Fuel Service Centers Without Power Generation

Fuel service centers that do not contain power reactors show certain safeguards benefits compared with dispersed facilities. In general, the safeguards advantages inherent in power-generating service centers apply to centers without reactors. Such centers would contain storage ponds for irradiated (spent) fuel, reprocessing plants, waste management plants, and fuel refabrication facilities. The primary difference between centers with and without power-generation is size, the former being significantly larger. Certain circumstances may call for spiking fresh recycle fuel by preirradiation before shipping it offsite, in which case an irradiation facility would also be part of the center. However, this facility would be

smaller than a single power reactor and would not present safeguards concerns that would differ from those related to dispersed facilities.

5.7 DECOMMISSIONING AND DECONTAMINATION

Power-generating fuel service centers could have advantages in decommissioning, but they could also present more extensive problems after radioactive contamination.

5.7.1 Power-Generating Fuel Service Centers

The decommissioning of power plants would be less disruptive in a large center because it would involve a small fraction of the power generated and because the required technical support (manpower and machinery) could be continuously maintained at the site. The center with resident technical personnel would hold a similar advantage in renovating or modifying fuel-cycle plants and power plants.

An accidental release of radioactivity in a service center could potentially contaminate many reactor areas. The contamination would generally be limited to several reactors because of the large exclusion distance between them, but depending on the severity of the accident and the prevailing weather conditions many reactors could also be affected. In contrast, only a few reactors would be contaminated by a similar accident at a dispersed site.

In energy centers, an accident in the fuel-cycle area (the reprocessing plant primarily) could also contaminate power plant areas, in addition to the fuel-cycle plants themselves, although this potential contamination effect would be much smaller than for power reactors. Therefore, clustering many power reactors at a single site has potential disadvantages such as large-scale loss of power generation or the need for extensive decontamination. However, the probability of significant contamination of the power plant areas due a fuel-cycle area accident would be very small because of containment provisions and separation distances.

5.7.2 Fuel Service Centers Without Power Generation

The decommissioning of colocated fuel-cycle facilities without power generation would be similar to that for dispersed facilities but less disruptive

socioeconomically if it were done gradually or if it involved the renovation or modification of selected facilities.

As with power-generating centers, facility colocation 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. Such contamination would cause fuel-supply disruptions, but operation of the associated reactors might be affected minimally, provided that fuel were available from other sources.

5.8 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

5.8.1 Power-Generating Fuel Service Centers

Because of their large impacts, power-generating fuel service centers require significant research, development, and demonstration. The primary need is to understand the climatic effects of concentrated heat rejection. Another need is to find technological ways to reduce the cost and ecological penalties from transmission of large amounts of power to distant regions. Research, development, and demonstration in both areas are expected to be satisfied by that required for the current sites containing three or four large power reactors.

In connection with the study of heat-rejection problems, initial emphasis should be on collecting weather data and observing wildlife habitat and migration patterns. Specific technical areas for research, development, and demonstration are:

- o Vorticity and turbulence.
- o Precipitation enhancement.
- o Interaction of vapor plumes with pollutants.
- o Optimization of heat-rejection systems.
- o Criteria for power plant layout.

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For power transmission, the power-generating center would benefit more than would dispersed sites from major advances in higher-capacity (higher-voltage) power-transmission technology or improved system reliability. Higher-capacity lines and advanced switching and control would reduce the large right-of-way land requirements and thus make the large power-generating center more practical for both nearby regions and remote sites. Improved system reliability could make remote siting more attractive, which would favor energy center siting.

5.8.2 Fuel Service Centers Without Power Generation

No research, development, and demonstration needs have been identified that are different from those needed for separate facilities.

5.9 LICENSING-TIME ASSESSMENT

Safety considerations and environmental impacts of fuel service centers without power generation are not significantly different from those impacts for dispersed fuel-cycle facilities having the same capacity. Licensing of fuel service centers without power generation would take about the same time as licensing of separate facilities. Safeguards considerations may show some advantage to colocating fuel-cycle facilities that would make such facilities more easily licensed.

Power-generating service centers can have significant environmental and socioeconomic impacts that are substantially different from those associated with the same capacity of dispersed facilities. Major impacts are land and water use, size of supporting work force, and regional economic effects. Time requirements for accommodating these impacts and overcoming the associated institutional barriers make it unlikely that large power-producing service centers could be licensed until after the year 2000. Expansion of Federally-owned sites which have on-going nuclear programs, have large land areas, and have water available would probably be the earliest possibility for establishing these centers. These sites are at Oak Ridge, Tennessee; Richland, Washington; and Aiken, South Carolina.

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