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VOLUME I

LIGHT-WATER REACTORS

January 1980

NONPROLIFERATION ALTERNATIVE SYSTEMS
ASSESSMENT PROGRAM

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

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

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

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

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

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

The preparation of this document takes into consideration NRC responses to the DOE preliminary safety and environmental submittal of August 1978. Responses to these initial comments have been, to the extent possible, incorporated into the text. Comments by the NRC on this PSEID were received in mid-August 1979 and, as a result of these comments, some changes were made in the document. Additional

comments were incorporated as Appendix B. Comments that are beyond the scope and resources of the NASAP may be addressed in research, development, and demonstration programs on systems selected for additional study. The intent of this document (and the referenced material) is to provide sufficient information on each system so that the NRC can independently ascertain whether the concept is fundamentally licensable.

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

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Chapter 1

GENERAL DESCRIPTION

Light-water reactors (LWRs) have been licensed at a variety of sites in the United States, although the plant features necessary to achieve acceptability vary widely from site to site. Variations in the design of the balance of plant (BOP) are not expected to significantly affect conclusions about the Nonproliferation Alternative Systems Assessment Program (NASAP) alternatives; for that reason, a reference reactor and plant has been defined for this comparative evaluation. The reference reactor and plant is a System 80 (Ref. 1) reactor by Combustion Engineering, Inc. (C-E), with the BOP as defined for the Perkins nuclear power station (Ref. 2). The other NASAP alternatives will be evaluated in terms of differences from this reference reactor. The issues and/or design considerations relative to other LWR designs (the Babcock & Wilcox pressurized-water reactor (PWR), the Westinghouse pressurized-water reactor (PWR), and the General Electric boiling-water reactor (BWR)) are expected to be similar. Analysis would be needed in each case to establish specific similarities or specific differences.

The reference system is a PWR with a two-loop reactor coolant system and the auxiliary systems directly related to the nuclear steam supply system (NSSS), as illustrated in Figure 1-1. The NSSS is housed in a containment building designed to meet all compatibility requirements. The C-E System 80 NSSS is a design for a single unit in that it has its own set of components that are important to safety. They are not shared with the other two NSSS at the Perkins nuclear power station.

A summary of major plant characteristics is given in Table 1-1.

1.1 REACTOR SYSTEM

The reactor system includes the reactor vessel, integral supports of a standard design, reactor-vessel head cover, reactor core, and all internal structures required to support the reactor core.

1.1.1 FUEL AND CONTROL RODS

The fuel rods consist of uranium dioxide pellets enclosed in Zircaloy-4 tubes with welded end plugs. The fuel tubes are grouped and supported in assemblies of 16 x 16 fuel rods with five guide tubes (see Figures 1-2 through 1-5). The four outer tubes are for control-element fingers; the center tube is for in-core instruments. It is possible to include in-core instruments in fuel assemblies with CEAs because the in-core instruments enter from the bottom and because the control-element assembly (CEA) fingers do not enter the center guide tube of fuel assemblies. The peak linear heat rate at full power is 12.5 kW/ft.

The CEAs have either 4, 8, or 12 fingers. The fingers are individually guided and protected from hydraulic forces by shroud tubes in the upper guide structure. The standard magnetic-jack control-element-drive mechanism can drive any of the three CEA types. Part-length CEAs are provided to shape power distribution in the core if necessary.

1.1.2 REACTOR INTERNALS

The internal structures include the core-support barrel, the core-support plate, the core shroud, and the upper guide-structure assembly. The core-support barrel is a right-circular cylinder supported from a ring flange from a ledge on the reactor vessel. The flange carries the entire weight of the core. Lateral-motion limiters (snubbers) are provided at the lower end of the core-support-barrel assembly. The core-support plate transmits the weight of the core to the core-support barrel by means of vertical columns and a beam structure. The core shroud surrounds the core and minimizes the amount of coolant bypass flow. The upper guide structure uses control-element shroud tubes to protect the individual fingers of CEAs from the effects of crossflow.

The C-E System 80 includes 89 installed control-element drive mechanisms (CEDMs) and 97 CEDM nozzles. The eight additional CEDMs can be installed during construction or at a later refueling. This provides significant flexibility for managing fuel-cycle economics.

1.1.3 VESSEL AND SUPPORTS

The C-E System 80 reactor pressure vessel is somewhat larger than previous Combustion Engineering vessels. Increased distance from the core edge to the vessel results in a 40-year fluence of 3.15×10^{19} neutrons/cm^2 (≥ 1 MeV). The maximum nil ductility transition temperature (NDTT) is 150° after 40 years of operation at an 80% plant factor.

The C-E System 80 vessel is supported by four flexible beams, one under each of the cold-leg nozzles. These beams positively restrain the vessel against cavity pressures generated during certain loss-of-coolant accidents (LOCAs). Keys at the bottom of the vessel restrict seismic "rocking" of the vessel.

Table 1-1. Summary of Perkins Plant characteristics

Principal design parameters of the reactor vessel	
Material	SA-533, Grade B, Class I, SA-508, Class II, clad with Type 304 austenitic stainless steel
Design pressure, psig	2,485
Design temperature, °F	650
Operating pressure, psig	2,235
Inside diameter of shell, in.	182-1/4
Outside diameter across nozzles, in.	267-1/8
Overall height of vessel and enclosure head, to top of CEDM nozzle, in.	601-5/8 (including bottom instrumentation nozzles)
Minimum cladding thickness, in.	1/8
Principal design parameters of the reactor-coolant piping	
Material	SA-516, Grade 70 with stainless steel Rollbond cladding
Hot leg, inside diameter, in.	42
Cold leg, inside diameter, in.	30
Between pump and steam generator, inside diameter, in.	30
Design pressure, psig	2485
Principal design parameters of the reactor-coolant system	
Operating pressure, psig	2,235
Reactor inlet temperature, °F	564.5
Reactor outlet temperature, °F	621
Number of loops	2
Design pressure, psig	2,485
Design temperature, °F	650
Hydrostatic test pressure (cold), psig	3,110
Total coolant volume, ft^3	11,643
Total reactor flow, gpm	445,600

Table 1-1. Summary of Perkins Plant characteristics (continued)

Principal design parameters of the reactor-coolant pumps	
Number of units	4
Type	Vertical, single-stage centrifugal; bottom suction and horizontal discharge
Design pressure, psig	2,485
Design temperature, °F	650
Operating pressure, nominal, psig	2,235
Suction temperature, °F	564.5
Design capacity, gpm	111,400
Design head, ft	363
Hydrostatic test pressure (cold), psig	3,110
Motor type	AC induction, single speed
Motor rating, hp	12,230 (cold)
Principal design parameters of the steam generators	
Number of units	2
Type	Vertical U-tube with integral moisture separator and economizer
Tube material	SB-163 Ni-Cr-Fe alloy
Shell material	SA-533, Grade B, Class I, and SA-516, Grade 70
Tube-side design pressure, psig	2,485
Tube-side design temperature, °F	650
Tube-side design flow per steam generator, lb/hr	82×10^6
Shell-side design pressure, psig	1,255
Shell-side design temperature, °F	575
Tube-side operating pressure, nominal, psig	2,235
Shell-side, operating pressure, maximum psig	1,155
Maximum moisture at outlet at full load, %	0.25
Tube-side hydrostatic test pressure (cold), psig	3,110
Steam pressure at full power, psia	1,070
Steam temperature at full power, °F	552.9
Steam flow at full power, per steam generator, lb/hr	8.59×10^6

Table 1-1. Summary of Perkins Plant characteristics (continued)

Containment system parameters	
Type	Steel spherical containment, cylindrical concrete shield building with hemispherical domed roof
Design parameters	
Inside diameter, ft	195
Height, ft	195
Free volume, ft^3	3.3×10^6
Reference incident pressure, psig	49.5
Steel thickness, in.	
Vertical wall	1-5/8"
Hemispherical head	~1.5
Concrete thickness, ft	
Vertical wall	3
Dome	3
Containment leak prevention and mitigation systems	Leaktight penetration, automatic isolation
Engineered safety features	
Emergency core-cooling system	
Number of high-head pumps	2
Number of low-head pumps	2
Number of safety injection tanks	4
Containment heat-removal system	
Number of pumps	2
Emergency power	
Number of diesel-generator units	2
Instrumentation and control systems	
Reactor protection system	
Number of manual switches	2 sets of 2 each
Automatic initiation parameter, channels/logic	2 of 4 logic for each trip

Table 1-1. Summary of Perkins Plant characteristics (continued)

Waste management system	
Miscellaneous liquid-waste system	
Waste tank	
Number	2
Capacity of each tank, gal	15,000
Waste condenser	
Number	1
Capacity, gpm	20
Waste condensate ion exchanger	
Number	1
Capacity, gpm	50
Secondary liquid-waste system	
Secondary waste tank	
Number	2
Capacity of each tank, gal	15,000
Secondary waste ion exchanger	
Number	2
Capacity, gpm	50
Gaseous-waste system	
Gas decay tank	
Number	3
Design pressure, psig	380
Capacity of each tank, ft^3	700
Solid-waste system	
Spent-resin tank	
Number	1
Capacity, gal	4,600
Concentrate tank	
Number	2
Capacity of each tank, gal	1,000
Mixer package	
Number	1
Capacity for waste, gpm	20[a]

[a]20-gpm mixer, three 10-gpm pumps.

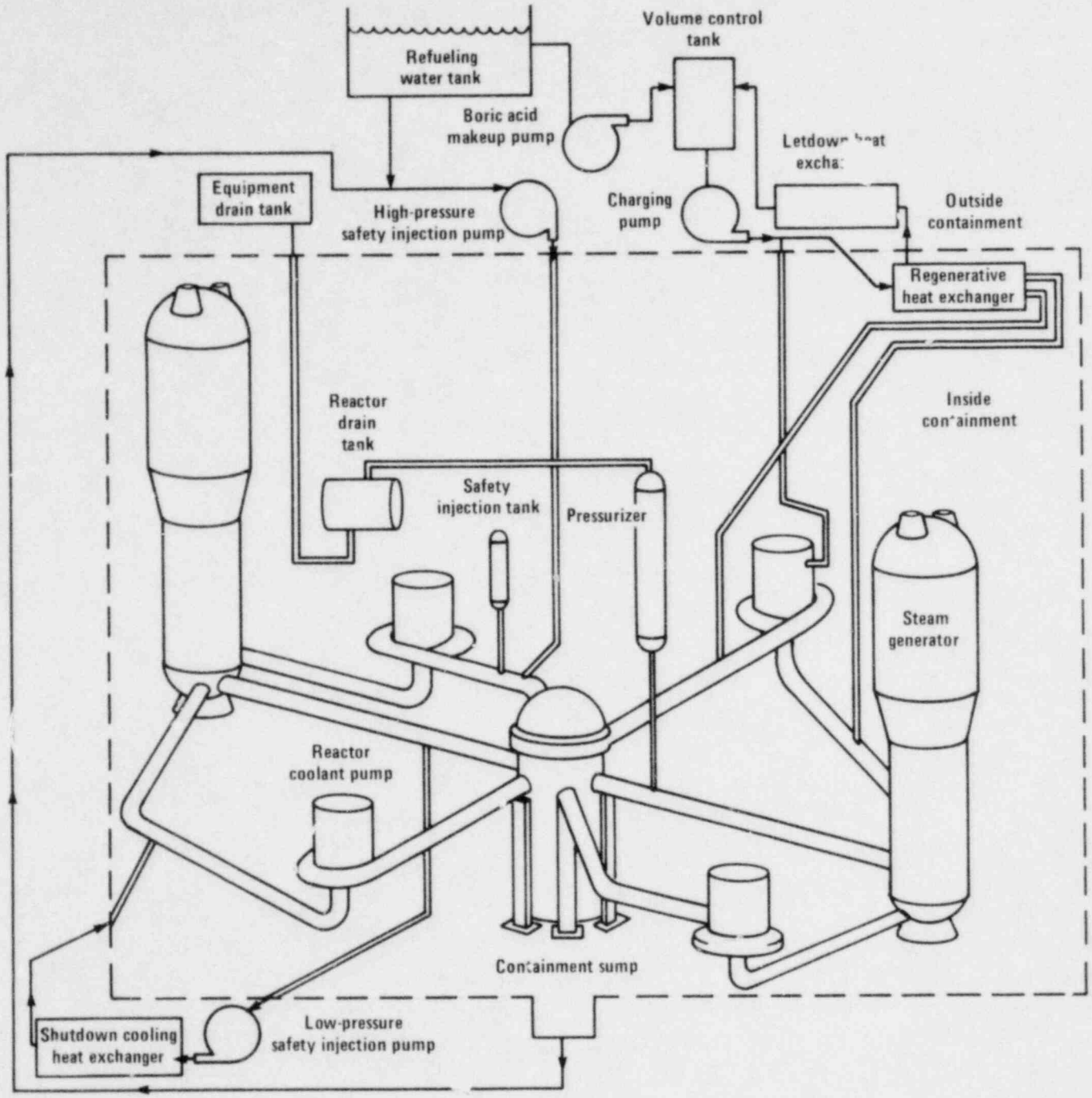


Figure 1-1. CESSAR design scope.

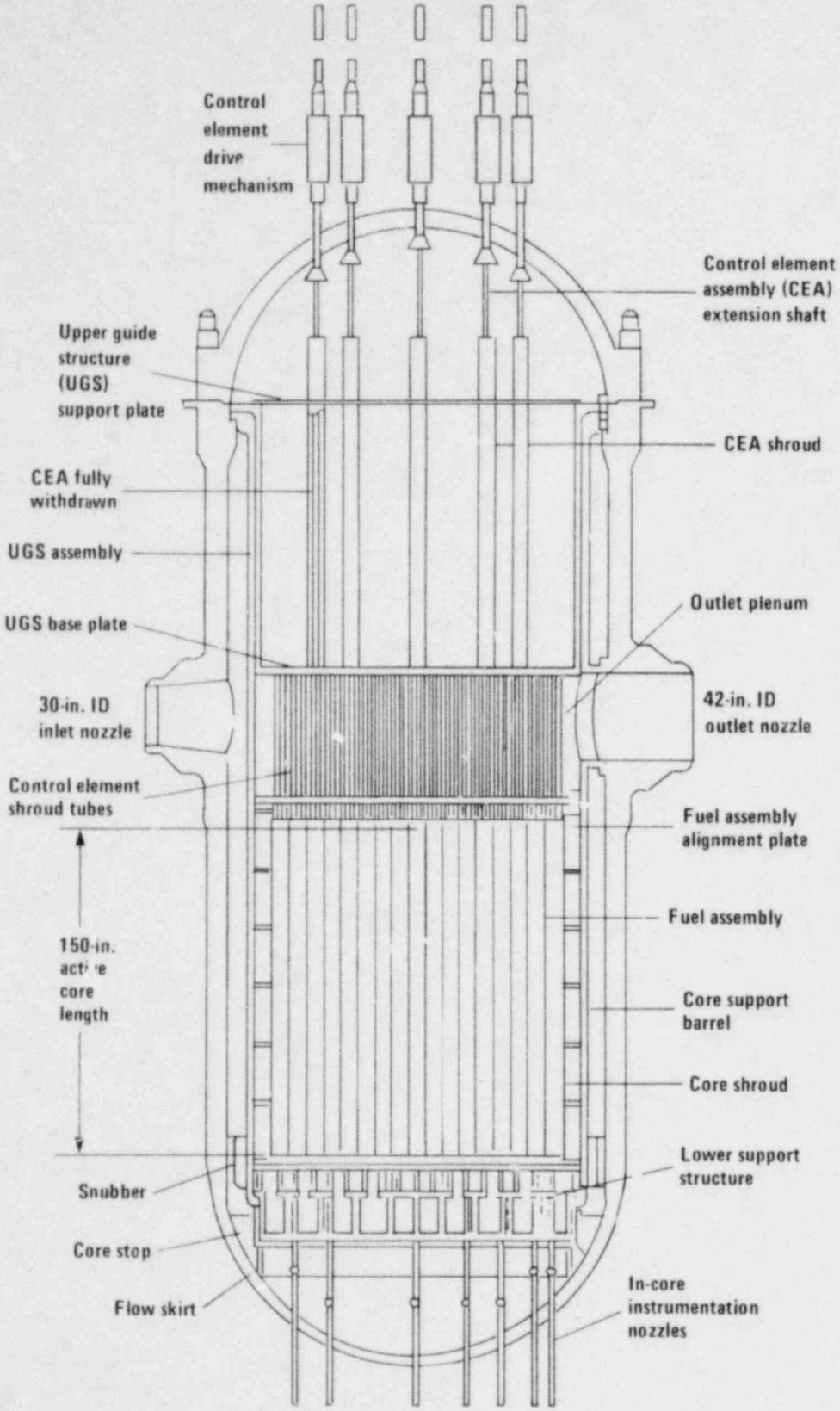


Figure 1-2. Reactor vertical arrangement.

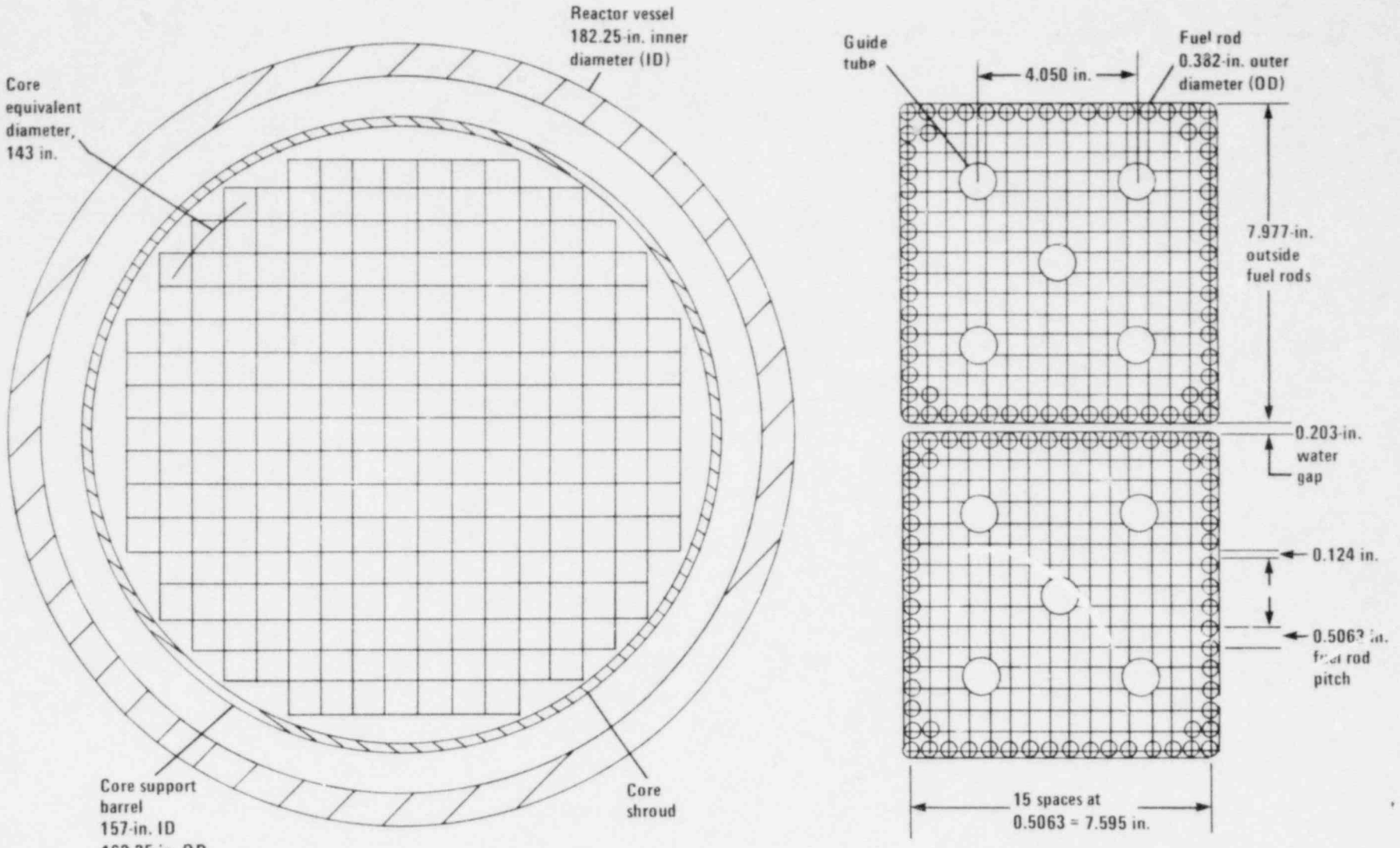


Figure 1-3. Reactor core cross section: 241 fuel assemblies.

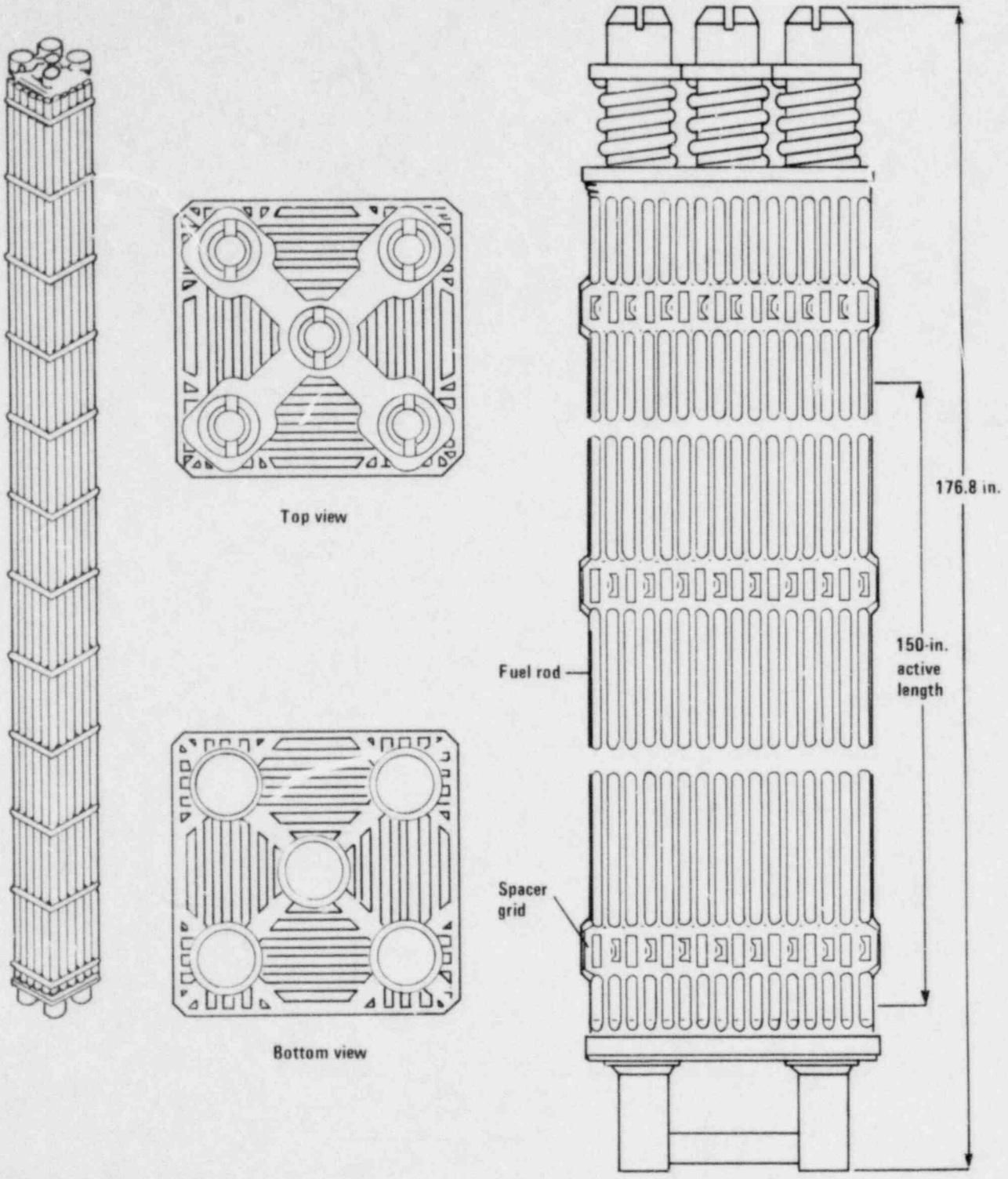


Figure 1-4. Fuel assembly.

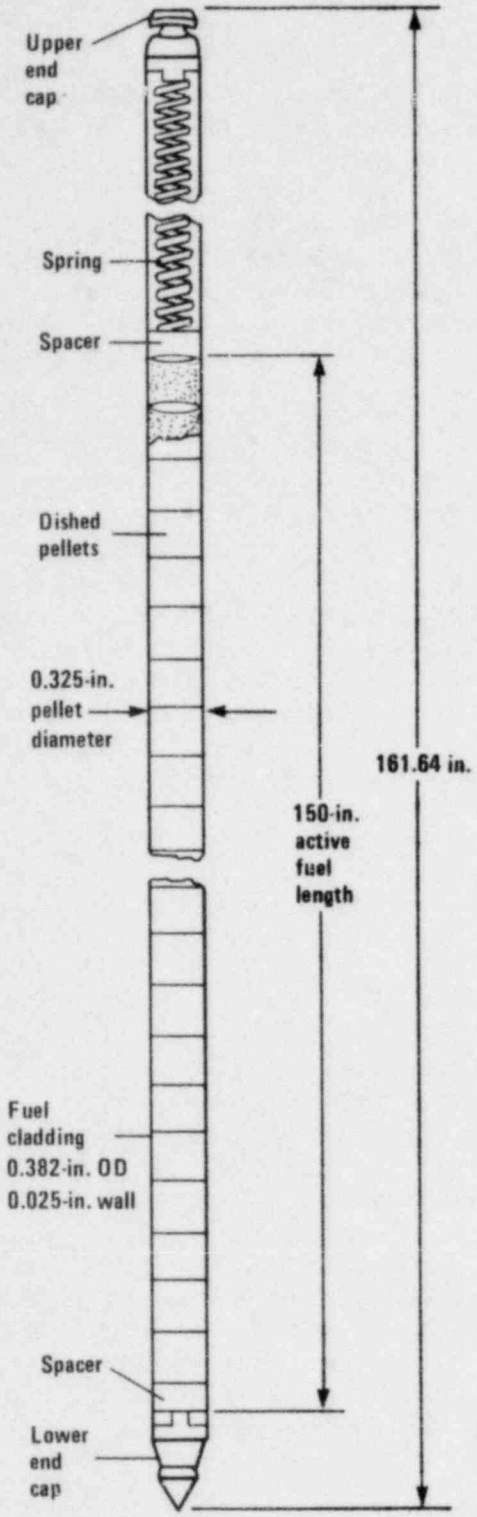


Figure 1-5. Fuel rod.

1.2 REACTOR-COOLANT SYSTEM

The reactor-coolant system consists of two closed reactor-coolant loops. Each loop includes a steam generator and two pumps. The water is circulated through the reactor vessel and core and two loops by the pumps. The water heated by the reactor flows through the "hot leg" to two steam generators, where heat is transferred to the secondary (steam) system, and is then pumped back to the reactor through the "cold leg." An electrically heated pressurizer with a safety-valve system is connected to one of the loops to establish and maintain the pressure. The major components of the reactor-coolant system have integral supports and snubbers of standard design. These supports are provided for the steam generators, the pumps, and the pressurizer.

1.2.1 STEAM GENERATORS

The steam generator consists of a vertical U-tube heat exchanger (see Figure 1-6) in which the heated water from the reactor enters near the bottom, passes through thousands of U-shaped tubes. first upward and then downward, and finally leaves near the same elevation at which it entered. The upper section of the steam generators contains equipment to separate the small quantities of liquid water droplets that are invariably present in the raw steam. The lower section of the steam generator, in which boiling occurs, is called the evaporator section; the upper section, in which the steam is separated from suspended droplets, is called the steam-drum section. Since the steam generators are large and operate under high pressure, they are fabricated from thick steel plate. The steam generators are therefore some of the most massive components of the plant.

An economizer section on the steam generators improves the overall heat transfer. Multiple feed nozzles allow the flow distribution in the economizer to be optimized for each power level.

1.2.2 PRIMARY COOLANT PUMPS

The pumps circulate the water between the reactor and the steam generator (see Figure 1-7). The system employs more than one loop and pump to meet its cooling requirements (i.e., redundancy in number though not in capacity), and each pump has a capacity greater than that required to accommodate the removal of the decay-heat load immediately after a reactor "scram."

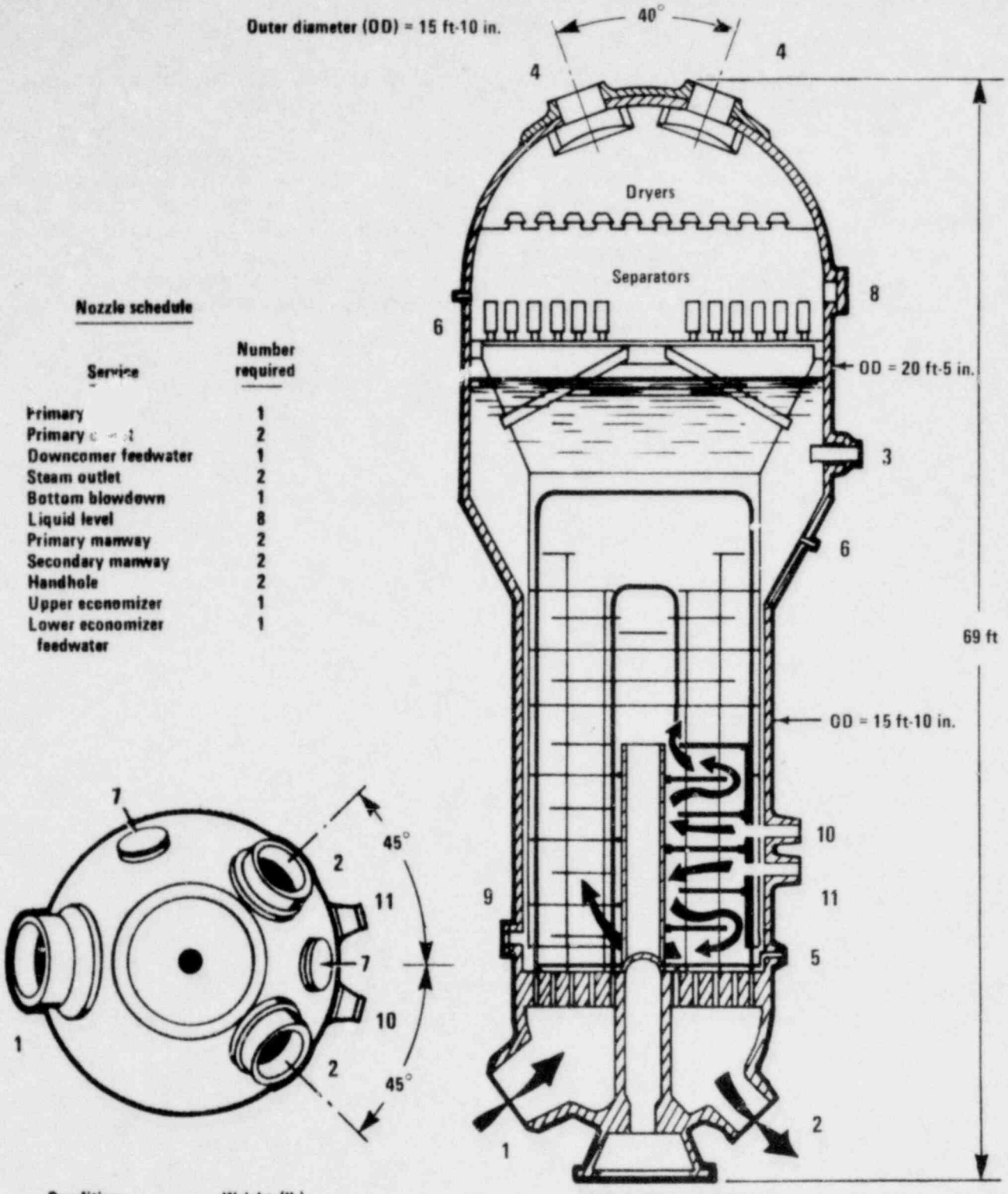
Each pump has four vertical and four horizontal support legs, each attached by a spherical joint to a rigid structural column. The columns are placed to allow unrestricted uniform motion from thermal expansion, but to limit seismic or LOCA-induced motion. Together with appropriate snubbers and stops, they will hold the pump in place after a break in either the inlet or the discharge line and during the full "design" earthquake (safe-shutdown earthquake).

1.2.3 PRESSURIZER

The NSSS is equipped with a pressurizer to maintain the required coolant pressure during steady-state operation, to limit pressure changes caused by the thermal expansion and contraction of coolant as plant loads change, and to prevent coolant pressure from exceeding the design pressure.

The pressurizer contains electric immersion heaters, multiple safety and relief valves, a spray nozzle, and appropriate valves and instruments. The lower portion

of the pressurizer contains liquid water; the upper portion contains steam. The pressurizer is connected by a surge line (pressure stabilizer) joining the pressurizer to the hot leg of one of the reactor coolant loops. When the plant electric load is decreased, the temperature of the primary coolant rises. This positive pressure surge in the primary system results in automatic operation of the spray system in the top of the pressurizer; this condenses some of the steam, keeping the pressure below the operating pressure of the relief valves. During a negative pressure surge caused by an increased plant electrical load, the electric heaters are turned on and generate sufficient steam inside the pressurizer to keep the primary-system pressure above the minimum allowable limit. A pressurizer is shown schematically in Figure 1-8.

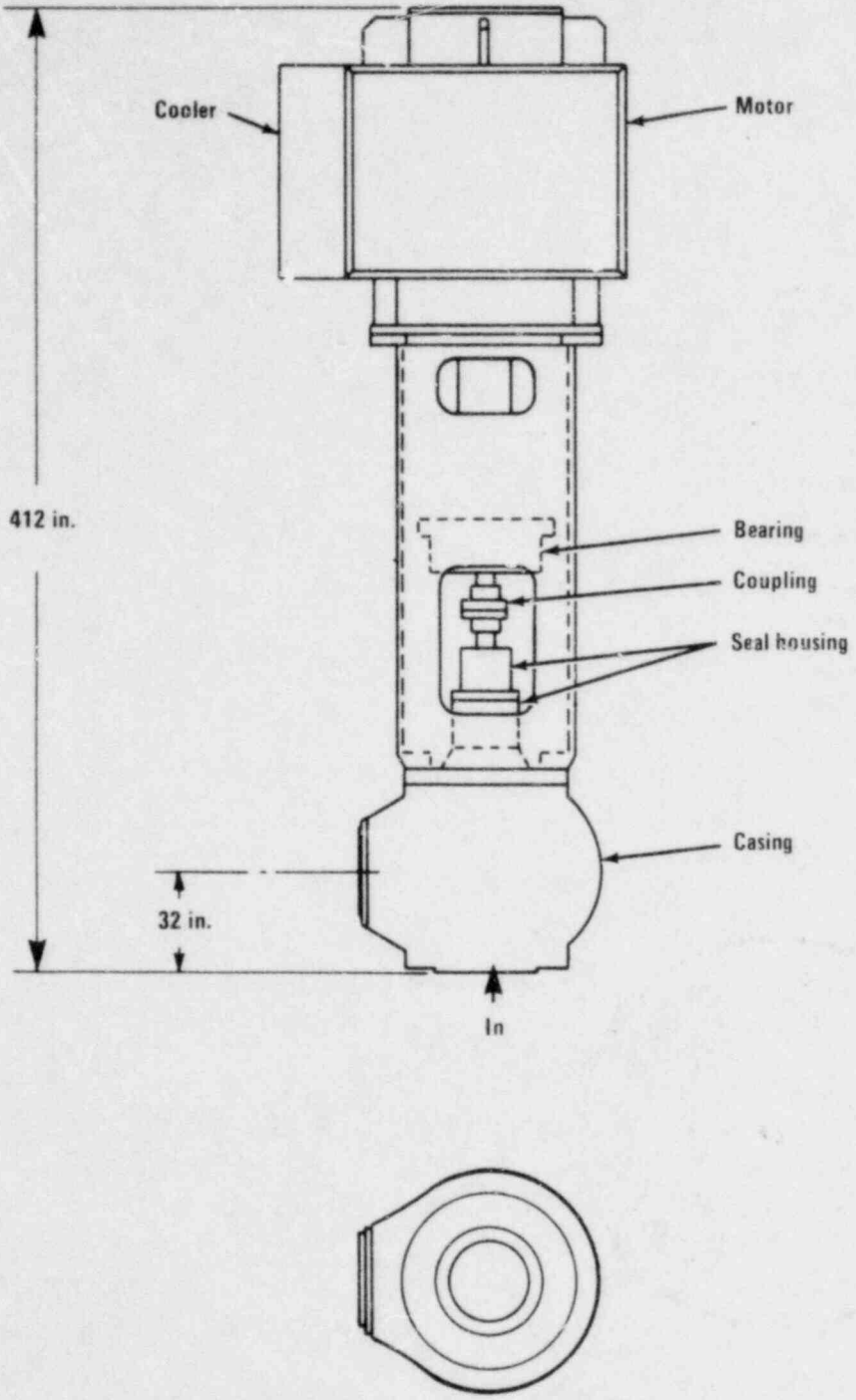


Nozzle schedule

No.	Service	Number required
1	Primary	1
2	Primary [illegible]	2
3	Downcomer feedwater	1
4	Steam outlet	2
5	Bottom blowdown	1
6	Liquid level	8
7	Primary manway	2
8	Secondary manway	2
9	Handhole	2
10	Upper economizer feedwater	1
11	Lower economizer feedwater	1

Condition	Weight (lb)
Dry	1,428,900
Flood	2,220,000
Shipping	1,570,000
Normal operating	1,725,000 (full load)

Figure 1-6. Steam generator.

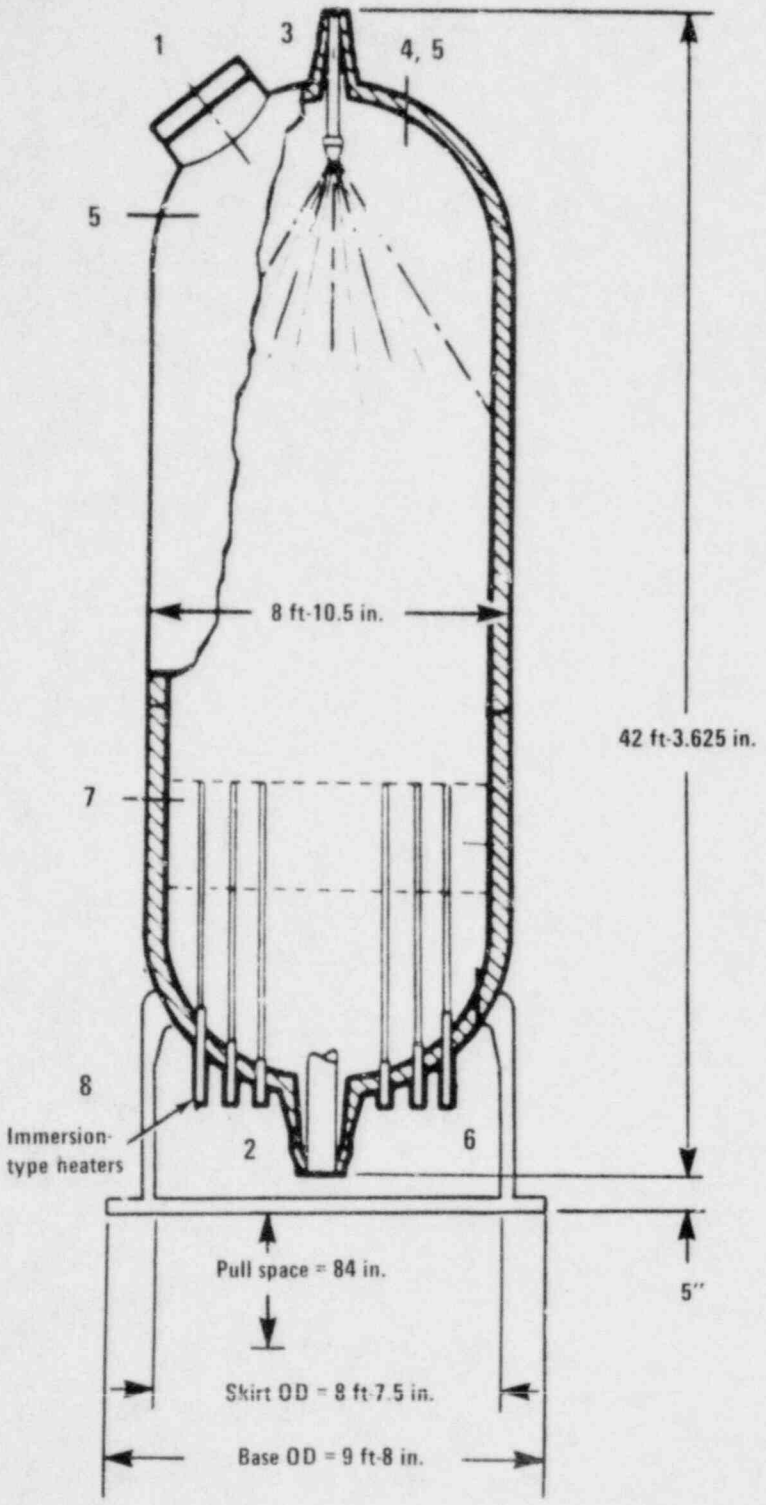


Item	Weight (lb)
Pump, dry	111,700
Pump, flooded	118,000
Motor	106,000
Operating assembly	222,200

Figure 1-7. Reactor coolant pump.

Nozzle schedule

No.	Service	Number required
1	Manway	1
2	Surge	1
3	Spray	1
4	Safety valve	4
5	Instrument	4
6	Instrument	2
7	Temperature	1
8	Heater	36



Condition	Weight (lb)
Dry	229,150
Flooded	341,300
Shipping	275,000
Normal operating	268,240

Abbreviation: OD = outside diameter.

Figure 1-8. Pressurizer.

1.3 ENGINEERED SAFETY FEATURES

The plant design incorporates redundant engineered safety features. In conjunction with the containment, these systems ensure that the offsite radiological consequences of any credible accident up to and including a double-ended break of the largest reactor-coolant pipe, will not exceed the guidelines of 10 CFR Part 100 or other appropriate performance criteria. The engineered safety features include the following:

1. Emergency core-cooling system
2. Containment spray and iodine removal systems
3. Combustible-gas control system
4. Containment isolation systems
5. Auxiliary feedwater system
6. Containment heat-removal systems
7. Annulus ventilation system
8. Habitability systems

1.3.1 EMERGENCY CORE-COOLING SYSTEM

An emergency core-cooling system (ECCS) is one of the engineered safety features provided to localize, control, mitigate, and terminate postulated accidents, including a loss-of-coolant accident. The ECCS includes four safety injection tanks and independent and redundant low- and high-pressure safety injection trains designed to automatically inject highly borated water into each of the four cold legs. This system ensures core cooling and protection for the complete size range of postulated primary and secondary coolant pipe-breaks.

1.3.2 CONTAINMENT SPRAY AND IODINE REMOVAL SYSTEMS

Two 100% capacity spray and iodine-removal systems provide spray to the containment environment for (a) cooling and reducing the pressure of the containment atmosphere, and (b) removing iodine after a postulated loss-of-coolant accident, if required.

The containment spray supplies borated water to cool and reduce pressure in the containment. The system is designed so that with one spray pump, one set of spray nozzles, and one shutdown cooling heat exchanger in operation, adequate cooling is provided. The pumps take suction initially from the refueling-water tank. Long-term cooling is based on suction from the containment sump through the recirculation lines.

The iodine-removal subsystem delivers an aqueous solution of sodium hydroxide from the spray chemical storage tank to two redundant suction lines of the containment spray pumps. Its rate of injection is regulated to give a suitable spray-water pH.

1.3.3 COMBUSTIBLE-GAS CONTROL SYSTEM

After a loss-of-coolant accident, the containment hydrogen recombiner system is used to prevent the concentration of hydrogen in the containment from reaching the lower flammable limit of 4% by volume. The system consists of two full-capacity, independent, parallel loops, each loop having the capability of keeping the containment hydrogen concentration below the limit of 3% by volume.

After a loss-of-coolant accident, both recombiner loops are started. The containment gas enters the loops through the suction headers and is then drawn into the

which are located in the auxiliary building. Hydrogen and oxygen from the containment atmosphere are catalytically recombined to form water vapor. The resulting mixture of water vapor and gas is piped back to the containment, completing the recombination cycle.

1.3.4 CONTAINMENT ISOLATION SYSTEM

The containment isolation system provides the means of isolating fluid systems that pass through containment penetrations so as to confine to the containment any radioactivity that may be released in the containment after a postulated design-basis accident. The system is required to function after a design-basis accident to isolate non-safety-related fluid systems that penetrate the containment.

1.3.5 AUXILIARY FEEDWATER SYSTEM

The auxiliary feedwater system provides an independent means of supplying water to the system. It insures that a heat sink is always available to the reactor-coolant system by maintaining an adequate water inventory in the steam generators.

1.4 PROTECTION, CONTROL, AND INSTRUMENTATION SYSTEMS

1.4.1 REACTOR PROTECTION SYSTEM

The reactor protection system consists of sensors, calculators, logic circuits, and supporting equipment for monitoring selected NSSS conditions. Redundancy, diversity, independence, and separation of reactor-protection circuits are provided in accordance with NRC criteria.

Conditions inside the reactor are normally maintained within acceptable limits by the characteristics of the reactor itself, by the reactor-regulating system, by dissolved boric acid, and by operating procedures. In addition, in order to prevent unsafe conditions for plant equipment or personnel, the reactor protection system initiates a reactor trip if any one of the selected parameters reaches its preset limit. Four independent channels normally monitor each of the selected parameters. The reactor protection system logic is designed to initiate protective action whenever the signal from any two of these four channels reaches the preset limit. Should this occur, the power supply to the magnetic-jack control-element-drive mechanism is interrupted, releasing the CEAs and allowing them to drop into the core and shut down the reactor. Redundancy is provided in the reactor protection system to insure that no single failure will prevent protective action when it is required. The protection system is completely independent of, and separate from, the control system.

System 80 provides "two out of four" trip logic. This system allows an instrument channel to be taken out of service indefinitely for maintenance with the plant still fully protected by the remaining "two out of three" logic. Spurious trips during instrument maintenance caused by nonredundancy are thus eliminated.

1.4.2 REACTOR CONTROL SYSTEM

The reactor is controlled in a combination of two ways: by boric acid in the reactor coolant and by the CEAs. Boric acid is used to control reactivity changes associated with large but gradual changes in water temperature, core xenon, fuel burnup, and power levels. Additions of boric acid also provide an increased shutdown margin during the initial fuel loading and refuelings. The movement of the CEAs controls reactivity during shutdown or power changes. The CEAs are actuated by control-drive mechanisms mounted on the reactor-vessel head. The control-drive mechanisms are designed to permit rapid insertion of the CEAs into the reactor core by gravity. The motion of the CEAs can be initiated manually or automatically.

The reactor-regulating system provides for adjustment of the reactor power in response to turbine load. The NSSS can follow a ramp change from 15 to 100% power at a rate of 5% per minute and at greater rates over smaller load changes up to a step change of 10%, except as limited by xenon. This control is normally accomplished by automatic movement of CEAs in response to a change in reactor-coolant temperature, with manual control capable of overriding the automatic signal at any time. A temperature controller compares the existing average reactor-coolant temperature with the value corresponding to the power called for by the temperature control program. If the temperatures differ, the CEAs are adjusted to bring them within the prescribed control band. Regulation of the reactor-coolant temperature in accordance with this program maintains the secondary steam pressure within operating limits and matches reactor power to load demand.

The pressure in the reactor-coolant system is controlled by regulating the temperature of the coolant in the pressurizer, where steam and water are held in thermal equilibrium. Steam is formed by the pressurizer heaters or condensed by the pressurizer spray as necessary to control pressure and accommodate expansion and contraction of the reactor coolant resulting from reactor-system temperature changes.

Overpressure protection for the reactor-coolant system is provided by spring-loaded safety valves designed in accordance with Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code and connected to the pressurizer. The discharge from the pressurizer safety valves is released under water in the reactor drain tank to insure condensation of the discharge. A rupture disk venting to the containment atmosphere is provided for overpressure protection if the safety-valve discharge exceeds tank capacity.

A turbine-control system is provided to regulate steam flow to the turbine as a function of system load. In the event of turbine trip, bypass systems release steam to the condenser or to the atmosphere. These systems are designed to reduce the sensible heat in the reactor-coolant system, maintain the steam-generator pressure during hot standby, and permit turbine trip without the steam-generator safety valves being opened when the condenser is available.

A water-level control system regulates the flow of feedwater to the steam generator. An auxiliary feedwater system is provided to insure flow to the steam generators in the event the main feedwater supply is inoperable.

The C-E System 80 is supplied with a reactor-cutback system to reduce power rapidly without trip by automatically dropping selected CEAs. Use of this system permits loss of feedpump or loss of a reactor-coolant pump without trip. Turbine trip at 100% power without reactor trip can be accomplished with a normal steam-dump capacity of 55%.

1.4.3 INSTRUMENTATION AND MONITORING SYSTEMS

The nuclear instrumentation includes out-of-core and in-core neutron-flux detectors. Eight channels of out-of-core instrumentation monitor the neutron flux and provide reactor-production and control signals during startup and power operation. Two of the channels monitor the neutron flux through the startup range, and four channels monitor the neutron flux from the startup range through the full-power range. The latter channels are used for protection. Two additional channels monitor the power range and provide control signals to the reactor regulating system.

Signals from the in-core detectors, together with other inputs, are fed to a designated core-monitoring computer that continuously generates values for the linear heat rate, the ratio of departure from nucleate boiling, and the axial power shape index. These serve to guide the operator in avoiding undesirable or prohibited power distributions.

A second, larger, plant computer performs more general functions: NSSS and BOP monitoring, logging, and alarming; NSSS output calculations; turbine, condenser, and feedwater-heater calculations; and trending. Should the core-monitoring computer be unavailable, the plant computer automatically assumes the core-monitoring functions. Neither computer is necessary for successful short-term plant operation.

The in-core instrumentation consists of thermocouples and self-powered neutron detectors to provide

1. Information on neutron-flux distribution and temperature in the core
2. Calibration of the out-of-core detectors

The process instrumentation monitoring includes the critical channels that are used for protective action. Monitoring of temperature, pressure, flow, and liquid level are provided as required, as inputs to the protection system inputs and to keep the operating personnel informed of plant operating conditions. The boric acid concentration in the reactor coolant is also monitored and displayed in the control room.

The plant gaseous and liquid effluents are monitored to insure that they remain within applicable radioactivity limits.

1.5 ELECTRICAL SYSTEMS AND EMERGENCY POWER

Each nuclear unit has two redundant and independent electric-power distribution systems to supply electric power to the redundant engineered safety systems equipment. Each of these two electric power distribution systems per unit has three power supplies:

1. The 230-kV transmission network through one of the unit's independent offsite power circuits
2. The 230-kV transmission network through the unit's other independent offsite power circuit
3. An independent diesel-generator unit arranged to supply its own distribution system

Power for the station auxiliaries is normally supplied from the generator bus through two full-sized auxiliary transformers. Each nuclear unit is provided with a preferred power supply consisting of two independent offsite power circuits capable of supplying power to engineered safety systems and a standby supply consisting of two independent onsite emergency diesel-generator units.

A manual tie to the auxiliary power system of another unit can be initiated, in the event that one of the preferred power circuits is unavailable because of equipment maintenance or failure.

1.6 FUEL-HANDLING AND STORAGE SYSTEM

A fuel-handling system is provided for the safe handling of fuel assemblies and CEAs for refueling and maintenance. The system provides for the assembly, disassembly, and storage of the reactor-vessel head and internals; it includes the following:

1. A refueling machine
2. A fuel-transfer carriage
3. Tilting machines
4. A fuel-transfer tube
5. A spent-fuel-handling machine in the fuel-handling building
6. Various devices used for handling the reactor-vessel head and internals

New fuel is stored dry in vertical racks in the fuel-handling building. Room is provided for storing one-third of a core. The spacing of the rack and fuel assembly precludes criticality.

The fuel pool, a reinforced-concrete structure lined with stainless steel, has storage capacity for one and one-third cores. Spent-fuel assemblies are stored in vertical racks so spaced as to preclude criticality with no credit taken for the borated pool water.

Cooling and purification equipment is provided for the fuel-pool water and may also be used for cleaning up refueling water after each fuel change in the reactor.

1.7 COOLING WATER AND OTHER AUXILIARY SYSTEMS

1.7.1 CHEMICAL AND VOLUME CONTROL SYSTEM

The chemical and volume control system (CVCS) controls the purity, volume, and boric acid content of the reactor-coolant system.

The purity of the coolant in the reactor-coolant system is controlled by the continuous purification of a bypass stream. Water removed from the reactor-coolant system is cooled in the regenerative heat exchanger; it then flows to the letdown heat exchanger, and through a filter and a demineralizer where corrosion and fission products are removed. It is then sprayed into the volume control tank and returned by the charging pumps to the regenerative heat exchanger, where it is heated before being returned to the reactor-coolant system.

The CVCS automatically adjusts the amount of reactor coolant to maintain a programmed level in the pressurizer. The level program partially compensates for changes in specific volume resulting from changes in coolant temperature and controlled leakage from the reactor-coolant-pump seals.

The CVCS controls the boric acid concentration in the coolant by a "feed and bleed" method. The purified letdown stream is diverted to a boron-recovery section, and either concentrated boric acid or demineralized water is sent to the charging pumps. The diverted water stream is processed by ion exchange and degasification and flows to a concentrator. The concentrator bottoms are sent to the refueling-water tank for reuse as boric acid; the distillate is deionized and stored for reuse in the reactor-makeup-water tank.

1.7.2 SHUTDOWN COOLING SYSTEM

The shutdown cooling system is used to reduce the temperature of the reactor coolant at a controlled rate from 350°F to a refueling temperature of approximately 135°F and to maintain its temperature at the proper level during refueling and extended-shutdown operations.

The shutdown cooling system uses the low-pressure safety injection pumps and containment spray pumps to circulate the reactor coolant through two shutdown cooling heat exchangers, returning it to the reactor-coolant system through the low-pressure injection header.

The component cooling-water system serves as a heat sink for the shutdown cooling heat exchangers.

1.7.3 COMPONENT COOLING-WATER SYSTEM

The component cooling-water system removes heat from the various auxiliary systems. Corrosion-inhibited demineralized water is circulated by the system through all components of the NSSS that require cooling water. During reactor shutdown, component cooling water is also circulated through the shutdown heat exchangers. The component cooling-water system provides an intermediate barrier between the reactor-coolant system and the intake cooling-water system.

1.7.4 SECONDARY CHEMISTRY CONTROL SYSTEM

The secondary chemistry control system (SCCS) continuously monitors the chemical composition of the condensate, feed, and steam-generator waters, continuously injects chemicals into them, and continuously processes steam-generator blowdown.

It monitors the main-steam-line and the condensate-pump discharge for sodium-ion concentration. A comparison of these concentrations provides an indication of condenser leakage. System instruments monitor additive concentrations at the condensate-pump discharge and the steam-generator feedwater inlet; comparison of these concentrations provides an indication of dissolved-oxygen concentration entering the steam generator. A conductivity monitor at the condensate-pump discharge provides another means of determining the total dissolved solids in condensate.

The blowdown recycle portion of the SCCS controls the concentration of additives and impurities in the steam-generator secondary-side water by continuous removal of contaminants via the blowdown line. The blowdown is regeneratively cooled and then purified by filtration and ion exchange. The purified blowdown is then returned to the feed train. If a reactor-coolant leak develops in the steam generator, making the blowdown radioactive, an additional mixed-bed ion exchanger is placed in service to insure complete radionuclide removal. Exhausted ion-exchange resin is sluiced to a regeneration subsystem in which the cation and anion resins are regenerated in separate tanks. The resins are remixed after regeneration and stored until needed.

1.7.5 PROCESS SAMPLING SYSTEM

The process sampling system is a means of obtaining samples from the reactor-coolant and auxiliary systems for laboratory analysis. Sample points from the reactor-coolant system include samples from the hot leg, pressurizer surge line, and pressurizer steam space. Safety injection system samples include those from the shutdown cooling suction line and the high-pressure ECCS pump main flow lines. Chemical and volume control system sample points have been provided for the purification filter inlet, purification filter outlet (purification ion-exchanger inlet), and purification ion-exchanger outlet. The remaining sample points are from each steam-generator blowdown. Sample points are located between all pieces of process equipment in both the secondary and liquid-waste management systems.

1.7.6 COOLING TOWERS

Closed-cycle cooling towers are provided to dissipate heat discharged by the plant.

1.7.7 AIR-HANDLING SYSTEMS

Separate ventilation systems are provided for the containment vessel, the control room, the reactor auxiliary building, and the diesel-generator building. A purge system is provided for the containment-vessel atmosphere.

1.7.8 PLANT FIRE-PROTECTION SYSTEM

The fire-protection system is common to all units and supplies water to fire hydrants, deluge systems, and hose racks in various areas of the plant.

Noncombustible and fire-resistant materials are used throughout the plant, particularly in areas containing critical portions of the plant such as the containment, control room, cable-spreading room, and rooms containing components of the engineered safety features.

A number of portable fire extinguishers are placed at key locations for use in extinguishing limited fires.

1.8 RADIOACTIVE-WASTE MANAGEMENT SYSTEMS

The radioactive-waste management systems provide all of the equipment required to collect, process, monitor, and discharge radioactive liquid, gaseous, and solid wastes that are produced during reactor operation.

1.8.1 LIQUID-WASTE MANAGEMENT

The miscellaneous-liquid-waste management system (MLWMS) collects and provides controlled treatment for potentially radioactive liquid wastes. The design objective is to protect personnel and the environment by providing monitoring, containment, and treatment systems for all plant effluents to insure that the releases of each radionuclide in liquids are below the concentrations specified in 10 CFR 20 and as low as practicable.

1.8.2 GASEOUS-WASTE MANAGEMENT

The gaseous-waste management system protects the plant personnel, the general public, and the environment by providing means for collecting, storing, and monitoring potentially radioactive gaseous waste. Design releases, both inside and outside the plant, are well below the concentrations specified in 10 CFR 20 and as low as practicable.

1.8.3 SOLID-WASTE MANAGEMENT

The function of the solid-waste management system is to process potentially radioactive solids and concentrated liquid wastes in preparation for shipment off the site. Inputs to the system include waste-concentrator bottoms, spent resins, chemical reagent waste, spent filter cartridges, and miscellaneous low-activity solids.

1.9 STEAM AND POWER-CONVERSION SYSTEM

The steam and power-conversion system for each unit is designed to remove heat energy from the reactor coolant, deliver it in the form of steam to the turbine-generator, and convert it to electrical energy. The closed feedwater cycle condenses the steam and heats feedwater for return to the steam generators.

An auxiliary feedwater system provides an independent means of supplying water to the steam generators. The nonconvertible heat energy in the exhaust steam from the turbine is dissipated to the atmosphere through a closed-cycle cooling-tower system.

1.10 CONTAINMENT AND SHIELD BUILDING

The containment, shown in Figure 1-9, is a 195-foot-diameter spherical steel shell with a wall thickness of 1-5/8 inches. This shell is supported in, but not anchored to, a spherical depression in an intermediate floor of the shield building, which is also referred to as the reactor building. The shield building is a reinforced-concrete cylinder with a spherical dome and totally encloses the containment. The outer edge of the containment-support floor is at plant elevation 92.0 feet (the plant grade elevation is at 100.0 feet). All containment leakage after postulated accidents will be collected in the annulus above elevation 92.0 feet, either by direct leakage into the annulus above elevation 92.0 feet or through a leak-chase system consisting of a network of steel channels welded over containment welds and penetration-seal welds.

An annulus ventilation system will continuously circulate air from the annulus through engineered-safety-feature filter systems at a rate of about 16,000 cubic feet per minute (cfm) for each redundant train after a vacuum of about 0.5 inch of water gauge is drawn by exhausting air from the annulus through the plant vent during the first 80 seconds after a postulated LOCA. After the vacuum is achieved, air will be exhausted at a rate of 400 cfm or less, as needed to match the inflow to the annulus. The inflow will be made up of outward containment leakage, inward leakage through the shield, and upward leakage through the containment support floor.

Space below the containment and inside the shield building is occupied by engineered-safety-feature equipment, including emergency core-cooling-system equipment, containment-spray-system equipment, and shutdown-cooling-system equipment. Some of the containment penetrations terminate in areas below the containment; others pass through the annulus above elevation 92.0 feet and terminate outside the shield building. Since the containment-support floor is not a fluid seal, postulated, but unlikely, pipe breaks in the regions below the containment could result in external pressures on the containment. The containment is designed to withstand these pressures without the use of vacuum-relief devices. Guard pipes are provided around high-energy lines that traverse the annulus. Although unlikely, cracks in moderate-energy lines within the annulus could cause flooding of the spaces below the containment-support floor. The facility is designed to prevent these effects from impairing the function of the containment and other engineered safety features.

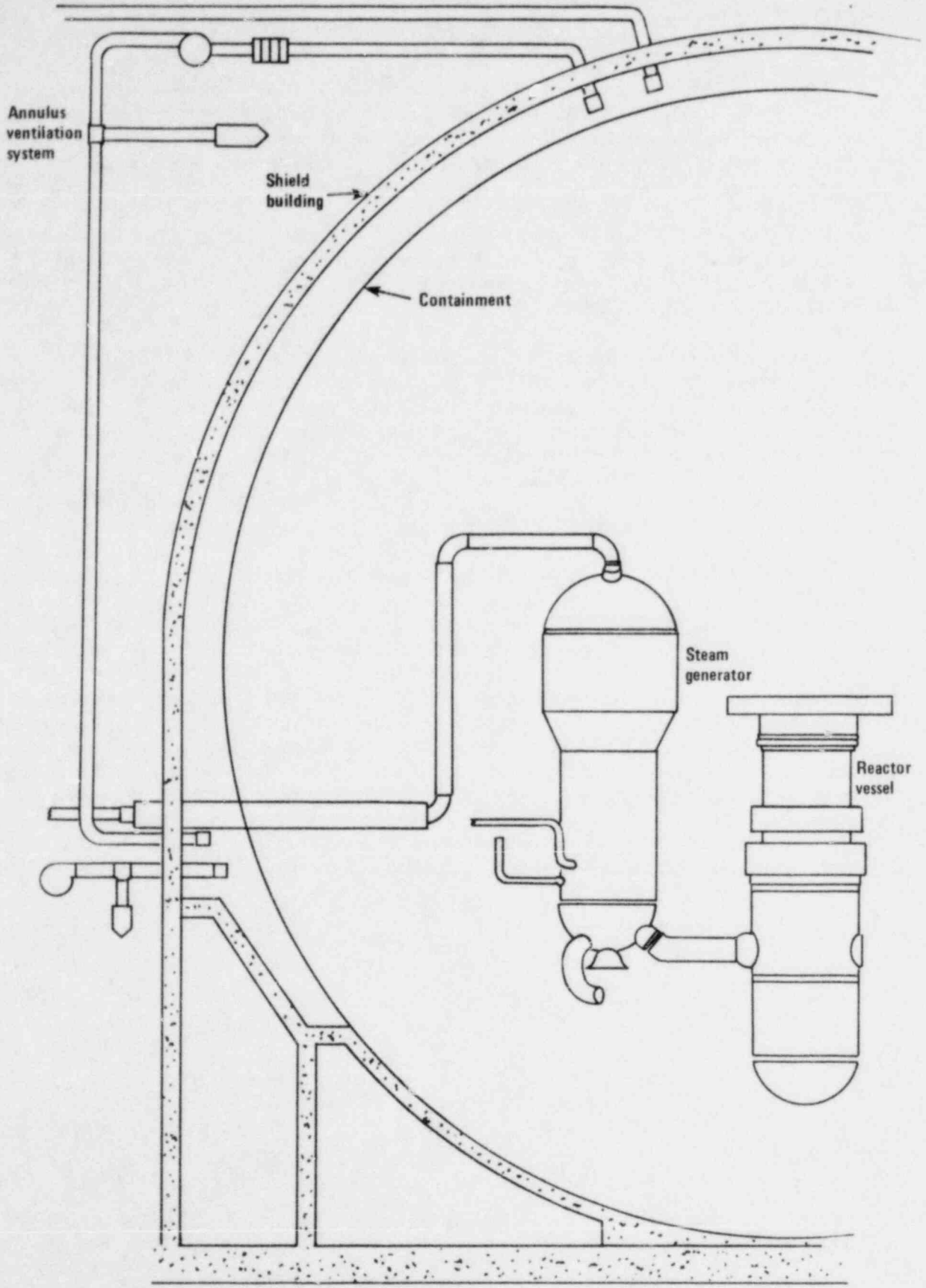


Figure 1-9. Containment and shield building.

1.11 OTHER MAJOR STRUCTURES

The auxiliary building is immediately adjacent to the shield building and includes fuel-handling areas, auxiliary equipment, the control room, and a nonseismic Category I control annex that will be supported on portions of the seismic Category I auxiliary building. The end of the turbine building abuts on this control annex in such a way that an extension of the turbogenerator axis will pass through the center of the containment.

There are nine cooling towers of the circular mechanical-draft type for primary cooling and two smaller ones to reject heat from the nuclear-service-water system. Makeup to the nine main towers and the two nuclear-service-water cooling towers is provided by pumping water from the makeup intake structure in the nuclear-service-water pond located immediately south of the cooling towers. This pond also serves as an intake sedimentation basin for water pumped from the river intake structure.

Two nuclear-service-water pump structures are located between the nuclear-service-water cooling towers and Unit 1. Each houses three pumps, one for each unit, to pump water to a component-cooling-water heat exchanger in one of two component-cooling loops for each unit.

1.12 ULTIMATE HEAT SINK

An alternative nuclear-service-water pond is formed in the upper portion of the nuclear-service-water pond by an underwater weir. The nuclear-service-water pond is connected by underground pipes to the nuclear-service-water pump structures. Water pumped from these structures through underground pipes to each unit can be discharged through underground pipes back to a discharge ditch that will discharge into the alternate nuclear-service-water pond. The complex of the two ponds and two cooling towers is the ultimate heat sink and will provide cooling capability even after severe natural phenomena and failure of man-made structures.

REFERENCES FOR CHAPTER 1

1. Combustion Engineering, Inc., System 80--Preliminary Safety Analysis Report, Standard PWR-NSSS, Docket No. STN 50-470.

2. Duke Power Company, Project 81, Preliminary Safety Analysis Report, Perkins Nuclear Station, Docket Nos. STN 50-488, STN 50-489, and STN 50-490.

Chapter 2

ONCE-THROUGH, LOW-ENRICHMENT URANIUM-235 FUEL, 30 MEGAWATT-DAYS PER KILOGRAM (PWR LEU (5)-OT)

2.1 DESCRIPTION

This reactor/fuel cycle combination is a standard pressurized-water reactor (PWR) using 3% low-enriched uranium oxide pellet fuel achieving 30 MWd/kg average burnup and operating in a once- through cycle. Spent fuel will be stored at the reactor site or away-from-reactor storage facility. Ultimately, the spent fuel will be sent to a geologic spent-fuel repository. Low-level waste from fabrication will be sent to a shallow land disposal site.

The fuel-cycle facilities associated with this reactor/fuel cycle combination, shown in the mass-flow diagram of Figure 2-1, are discussed in the following sections of Volume VII:

Enrichment	Section 3
Fuel fabrication 1	Section 4.1
Spent fuel storage	Section 6.3
Waste disposal 1	Section 7.1
Waste disposal 3	Section 7.3

For the purposes of the Nonproliferation Alternative Systems Assessment Program (NASAP) alternatives assessment, separate calculations were performed by Combustion Engineering, Inc., for this fuel cycle. The generalized reactor-performance characteristics are summarized in Table 2-1, and the reactor-design data are summarized in Table 2-2. Additional data on fuel management are presented in Section 2.1.4.

2.1.1 FUEL MECHANICAL DESIGN

2.1.1.1 Design Bases

For a complete description of the design bases of the fuel mechanical design, see Section 4.2.1.1 of Reference 1.

2.1.1.2 Design Description

For a complete description of the fuel design, see Section 4.2.1.2 of Reference 1. Table 2-3 contains a summary of selected core mechanical design parameters for the nuclear steam supply system of the Perkins nuclear power station.

2.1.1.3 Design Evaluation

For a complete description of the design evaluation, see Section 4.2.1.3 of Reference 1.

2.1.1.4 Testing and Inspection Plan

For a complete description of the testing and inspection plan, see Section 4.2.1.4 of Reference 1.

2.1.2 FUEL NUCLEAR DESIGN

2.1.2.1 Design Bases

For a complete description of the design bases of the fuel nuclear design, see Section 4.3.1 of Reference 1.

2.1.2.2 Description

For a complete description of the fuel nuclear design, see Section 4.3.2 of Reference 1. Table 2-4 contains a summary of selected design data for the nuclear steam supply system of the Perkins nuclear power station.

2.1.2.3 Analytical Methods

For a complete description of analytical methods, see Section 4.3.3 of Reference 1.

2.1.2.4 Nuclear Design Changes

For a complete description of the nuclear design changes, see Section 4.3.4 of Reference 1.

2.1.3 FUEL THERMAL-HYDRAULIC DESIGN

2.1.3.1 Design Bases

For a complete description of the bases for the fuel thermal-hydraulic design, see Section 4.4.1 of Reference 1.

2.1.3.2 Description

For a complete description of the thermal-hydraulic design, see Section 4.4.2 of Reference 1. Table 2-5 contains a summary of selected hydraulic and thermal design parameters for the fuel used at the Perkins nuclear power station.

2.1.3.3 Evaluation

For a complete evaluation of the thermal-hydraulic design, see Section 4.4.3 of Reference 1.

2.1.3.4 Testing and Verification

For a complete description of the testing and verification program of the thermal-hydraulic design, see Section 4.4.4 of Reference 1.

2.1.4 FUEL MANAGEMENT

Fuel-management information is summarized in Table 2-6. The isotopic distribution of the fuel inventory for the beginning and the end of the equilibrium cycle is listed in Tables 2-7 and 2-8, respectively. The reactor charge and discharge data for a 30-year lifetime are given in Tables 2-9 and 2-10, respectively.

The material flow diagram for the reference LWR once-through fuel cycle, shown in Figure 2-1, was obtained from Tables 2-9 and 2-10 by multiplying all values by 1,000/1,270 (see Table 2-1) to obtain a normalization to 1,000 MWe.

Table 2-1. Generalized reactor-performance specifications for the NASAP PWR reference design

Reactor thermal power output (gross), MW	3,817
Electrical power output, MW[a]	
Gross	1,344
Net	1,270
Plant heat rate, Btu/kW-hr	10,212
Core design and performance parameters	
Core heat output, MW	3,800
Core volume, liters	40,050
Core loading, kg	
Heavy metal	99,313
Fissile fuel	2,201
Conversion ratio	0.59
Average discharge exposure, MWd/MTHM[b]	30,390
Peak discharge exposure, MWd/MTHM[a]	55,000
Fuel type	Oxide
Reactor-inlet temperature, ^{o}F	565
Reactor-outlet temperature, ^{o}F	621
End-of-cycle excess reactivity	0

[a]Depends on specific plant design features; these values assume mechanical-draft cooling.

[b]Heavy-metal charged.

Table 2-2. Reactor-design data for the NASAP PWR reference design

Geometric information	
Core height, cm	381.0
Number of core enrichment zones (nominal)	3
Number of assemblies	241
Equivalent diameters, cm	365.8
Pins per assembly	236
Pin pitch-to-diameter ratio	1.325
Overall assembly length, cm	406.4
Lattice pitch, cm	1.288
Assembly material	Oxide fuel with Zircaloy-4 cladding
Cladding parameters	
Cladding outside diameter, mils	382.7
Cladding wall thickness, mils	25
Cladding material	Zircaloy-4
Fissile inventory at beginning of equilibrium cycle, kg	1,907
External fissile inventory, kg	NA
Fissile loss, kg/cycle	1,032
Specific power, kW/kg fissile	1,990
Power density, kW/kg HM	38.3

Table 2-3. Perkins core mechanical design parameters

Fuel assemblies	
Rod bundle arrangement	16 x 16
Design	CEA
Rod pitch, in.	0.5063
Cross-section dimensions, in.	7.98 x 7.98
Fuel weight (as UO_2), lb	256,520
Total weight, lb	317,131
Number of grids per assembly	12
Fuel rods	
Number of locations	56,876[a]
Outside diameter, in.	0.382
Diametral gap, in.	0.007
Cladding thickness, in.	0.025
Cladding material	Zircaloy-4
Fuel pellets	
Material	UO_2 Sintered
Diameter, in.	0.325
Length, in.	0.390
Control assemblies	
Cladding material	Ni-Cr-Fe alloy
Cladding thickness, in.	0.035
Core structure	
Core barrel inside and outside diameter, in./in.	157/162.25

[a]Some of the rod locations are occupied by burnable poison rods.

Table 2-4. Perkins nuclear design data

Structural characteristics	
Core diameter, in. (equivalent)	143
Core height, in. (active fuel)	150
H_2O/V, unit cell (cold volume ratio)	3.57
Number of fuel assemblies	241
Number of UO_2 rods per assembly	
Batch A	236[a]
Batch B	236
Batch C	236
Performance characteristics	
Loading technique	Three-batch mixed central zone
Fuel discharge burnup, MWd/MTU	
Average first cycle	13,740
First-core average	23,200
Fuel enrichment, wt% U-235	
Region 1	1.9
Region 2	2.4
Region 3	2.9
Control characteristics	
Effective multiplication (beginning of life, rods out, no soluble boron)	
Cold, zero power, clean	1.169
Hot, zero power, clean	1.133
Hot, equilibrium xenon, full power	1.071
Control-element assemblies (CEAs)	
Material	B_4C
Number of control assemblies (full/part length)	81/8[b]
Number of absorber rods per CEA (or RCC) assembly	4, 8, or 12
Total rod worth (hot), %	10.0
Boron concentrations, ppm	
To shut reactor down with no rods inserted, clean, cold/hot	960/980[c]
To control at full power with no rods inserted, clean/equilibrium xenon	820/560
Kinetic characteristics range over first cycle[d]	
Moderator temperature coefficient, per °F	-0.4×10^{-4} to -2.1×10^{-4}
Moderator pressure coefficient, per psi	$+0.49 \times 10^{-6}$ to $+2.55 \times 10^{-6}$
Moderator void coefficient, per % void	-0.26×10^{-3} to -1.35×10^{-3}
Doppler coefficient, per °F	-1×10^{-5} to -1.8×10^{-5}

[a] In the first core, some UO_2 rods may be replaced by burnable poison rods.
[b] Locations are provided for eight additional CEAs.
[c] Figures take into account the equivalent worth of shim rods.
[d] Hot, operating.

Table 2-5. Perkins fuel hydraulic and thermal design parameters

Total core heat output, MWt	3,817
Total core heat output, Btu/hr	$13,000 \times 10^6$
Heat generated in fuel, %	96.5
System pressure, nominal, psia	2,250
System pressure, minimum steady state, psia	2,200
Hot channel factors, overall heat flux, F_q	2.35
Departure from nucleate boiling ratio at nominal conditions (W-3)	2.22
Coolant flow, lb/hr	
Total flow rate	164×10^6
Effective flow rate for heat transfer	157.4×10^6
Effective flow area for heat transfer, ft^2	60.8
Average velocity along fuel rods, ft/sec	16.6
Average mass velocity, $lb/hr\text{-}ft^2$	2.59×10^6
Plant temperatures, ^oF	
Nominal inlet	565
Maximum inlet due to instrumentation error and deadband	568
Average rise in vessel	56
Average rise in core	58
Average temperature in core	594
Average temperature in vessel	593
Hot channel outlet	653
Heat transfer at 100% power	
Active heat-transfer surface area, ft^2	69,000[a]
Average heat flux, $Btu/hr\text{-}ft^2$	182,200
Maximum heat flux, $Btu/hr\text{-}ft^2$	425,700
Average thermal output, kW/ft	5.34
Maximum thermal output, kW/ft	12.5
Fuel center temperature, ^oF	
Maximum at 100% power	3,420
Maximum at overpower	3,740
Thermal output, kW/ft at maximum overpower	14
Engineering heat-flux factor	1.03

[a]Based on eight burnable poison rods per fuel assembly replacing fuel rods.

Table 2-6. Fuel-management information (PWR UO_2 once-through standard)

Average capacity factor, %	75
Approximate fraction of core replaced annually	One-third
Lag time assumed between fuel discharge and recycle reload, years	2
Fissile-material reprocessing loss fraction, %	1
Fissile-material fabrication loss fraction, %	1
Yellowcake requirements, ST/GWe	
Initial core	408
Annual equilibrium reload requirement	194
30-year cumulative requirement	6,128
Separative-work requirements, 10^3 SWU/GWe	
Initial core	212
Equilibrium reload	118
30-year cumulative requirement	3,632
Requirements for special fuel materials (fissile Pu, U-233, etc.), kg HM/GWe	
Initial load	0
Annual equilibrium charge, discharge	0
30-year cumulative requirement	0
Other data for proliferation-resistance assessment	
Fuel element weight, kg	650
Fresh and discharge fuel radiation level at 1 meter at 90 days, R/hr	
Fresh fuel	Air: 0.020
Discharge fuel	Air: 20,000 Water: 200
Discharge-fuel energy-generation rate after 90-day cooling (watts/element)	12,600

Table 2-7. Fuel inventory at the beginning of equilibrium cycle

Isotope	Fresh fuel, zone 1	Once-burnt fuel, zone 2	Twice-burnt fuel, zone 3
Th-232			
Pa-233			
U-232			
U-233			
U-234			
U-235	1,027.71	660.49	429.01
U-236		16.68	105.25
U-238	33,230.71	32,973.51	32,716.37
Pu-238		0.35	1.74
Pu-239		125.22	161.42
Pu-240		23.96	51.25
Pu-241		10.49	28.78
Pu-242		1.06	6.39
Fission products			
Other isotopes			
Am-241		0.114	0.536
Cm-242		--	--
Np-237		3.58	8.98

Table 2-8. Fuel inventory at the end of equilibrium cycle

Isotope	Fresh fuel, zone 1	Once-burnt fuel, zone 2	Twice-burnt fuel, zone 3	Zone 4
Th-232				
Pa-233				
U-232				
U-233				
U-234				
U-235	660.49	429.01	273.83	
U-236	16.68	105.25	127.46	
U-238	32,973.51	32,716.37	32,447.87	
Pu-238	0.35	1.74	4.31	
Pu-239	125.22	161.42	174.65	
Pu-240	23.96	51.25	71.90	
Pu-241	10.49	28.78	43.09	
Pu-242	1.06	6.39	15.31	
Fission products	388.58	768.16	1,064.53	
Other isotopes				
Am-241		0.114	0.536	0.482
Cm-242		--	--	--
Np-237		3.58	8.98	14.71

Table 2-9. Reactor charge[a] data for years 1 through 9

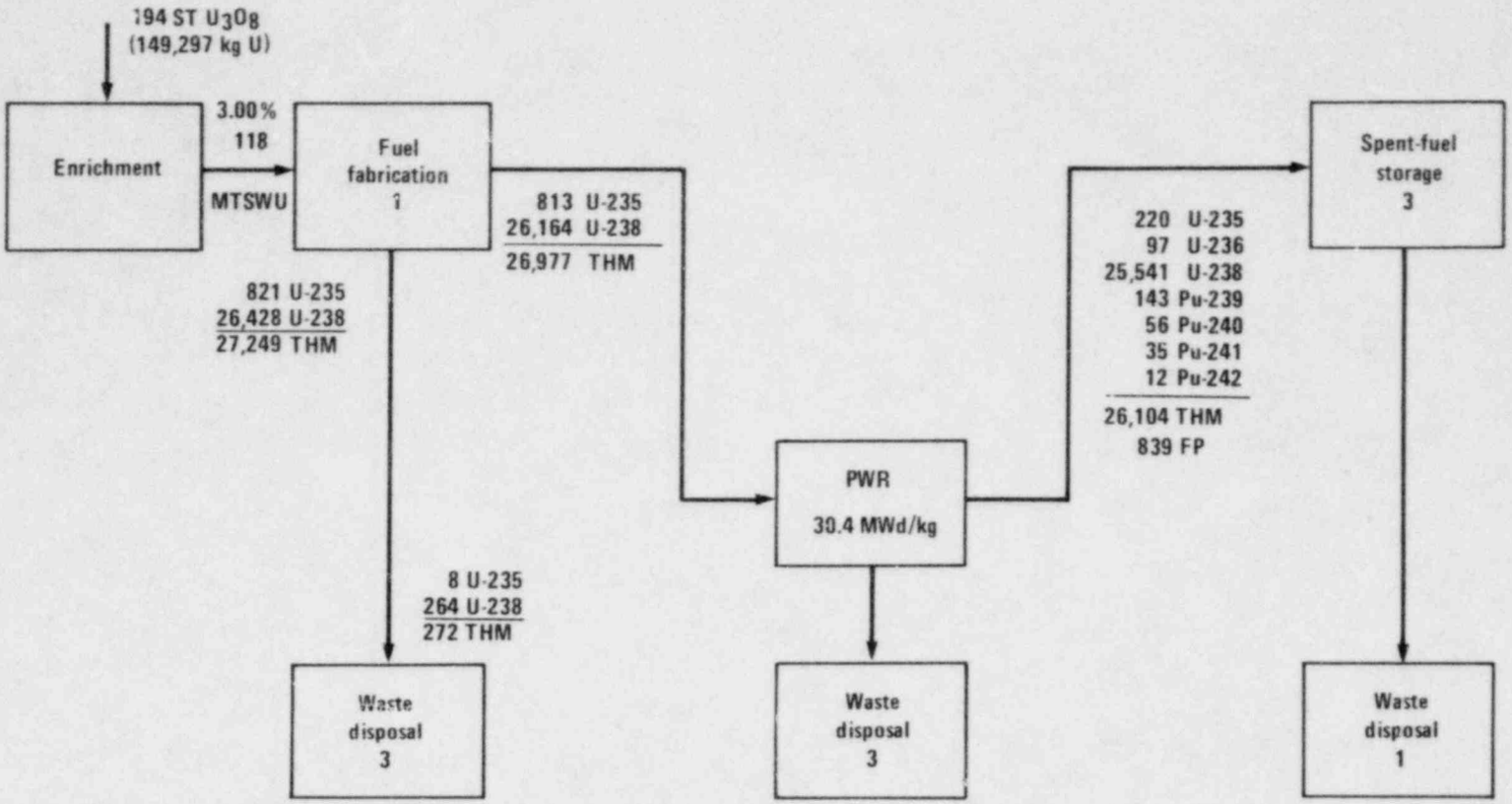
Year	Th-232	Pa-233	U-232	U-233	U-234	U-235	U-236	U-238	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	Np-237	Total heavy metal
1						2,201		97,112							99,313
2						1,071		33,048							34,119
3						1,028		33,517							34,545
4						954		33,135							34,119
5						1,058		33,061							34,119
6						1,026		33,519							34,545
7						1,028		33,091							34,119
8						1,028		33,091							34,119
9						1,041		33,504							34,545

[a]In kilograms of heavy metal, for a 1,270-MWe (net) reactor.

Table 2-10. Reactor discharge[a] data for years 1 through 11

Year	Th-232	Pa-233	U-232	U-233	U-234	U-235	U-236	U-238	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	Np-237	Total heavy metal
1						243	56	33,158		137	42	20	4		33,658
2						214	81	30,948		145	58	31	10		31,487
3						244	111	31,302		166	69	40	14		31,945
4						304	127	32,277		183	70	44	14		33,018
5						271	124	32,706		183	73	45	16		33,417
6						261	118	32,349		179	71	44	15		33,036
7						295	126	32,282		182	70	44	15		33,017
8						274	123	32,718		183	72	45	15		33,431
9						278	123	32,305		181	71	44	15		33,015
10						278	123	32,303		181	71	44	1[c]		33,015
11						282	124	32,706		183	72	45	15		33,427

[a]In kilograms of heavy metal, for a 1,270-MWe (net) reactor.



Notes:

1. Mass flows are in kilograms per 0.75 GWe-yr.
2. Abbreviations: FP, fission products; MTSWU, metric tons of separative-work units; THM, total heavy metal.

Figure 2-1. Mass flows for the reference LWR once-through fuel cycle PWR LEU(5)-OT.

2.2 SAFETY CONSIDERATIONS

2.2.1 GENERAL

Most nuclear power stations in the world, operational or planned, are light-water reactors (LWRs), either pressurized-water reactors (PWRs) or boiling-water reactors (BWRs). Sixty-six LWRs are currently licensed for operation in the United States and an additional 128 are under construction or planned; a similar number of LWRs are in operation, under construction, or planned in other countries. Safety standards, regulatory requirements, and licensing procedures have all evolved with time. Safety concerns have ranged, for example, from reactivity transients and shutdown systems to blowdowns in containment, to severe design-basis accidents and mitigating systems, to the performance of actual materials, systems, and people. The primary safety concerns of one epoch have been superseded in considerable measure by those of later times. Successive plateaus of technical understanding are achieved as solutions are found to earlier problems. Design studies, research, operating experience, and regulatory imperatives all contribute to the increased understanding and thus to the safety improvements adopted and accepted.

The primary assurance of safety depends on a high degree of reliability and predictability obtained by the application of rigorous standards in the design, construction, and operation of the nuclear facility and by extensive quality assurance actions. In addition, in accordance with the "defense-in-depth" concept, safety features and engineered safeguards systems are provided to prevent or to accommodate the consequences of accidents postulated to occur in spite of these measures.

The U.S. approach has been to rely on the defense-in-depth philosophy in the design of reactors. This concept requires that reactor systems tolerate a spectrum of operating transient and accident conditions while maintaining barriers to the release of fission products.

Defense in depth includes the following:

1. Designing for safety in normal operation and maximizing the ability to tolerate malfunctions through intrinsic features of sound conservative design, construction, selection of materials, quality assurance, testing, and operation. Margins are incorporated into the plant by adhering to regulatory requirements and the many accepted codes and standards of organizations such as the American Nuclear Society, the American Society of Mechanical Engineers, the American Society for Testing and Materials, and the Institute of Electrical and Electronics Engineers.
2. Anticipation that some abnormal incidents will occur during plant life and that some provision should be made to terminate such incidents and to limit their consequences to acceptable limits, even though important components or systems fail. Even under these conditions there are still significant margins provided as a result of utilizing conservative design practice and accepted codes and standards.
3. Providing protection against extremely unlikely events, which are not expected to occur during the life of a single plant, assuming failures of consequence-limiting equipment. From an analysis of these postulated events, features and equipment are designed into the plant to control the postulated events and to insure that there is no undue risk to the public.

The NRC regulations, as stated in Section 50.34 of 10 CFR 50, require each applicant requesting a construction permit or operating license for a nuclear power plant or a fuel-reprocessing plant to provide an analysis and evaluation of the design and performance of the structures, systems, and components of the facility, with the objective of assessing the risk to public health and safety resulting from operation of the facility. These analyses are to establish (a) the margins of safety during normal operations and transient conditions anticipated during the life of the facility, and (b) the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of accident consequences.

The conditions analyzed range from relatively trivial events that result in essentially no risk to the public (such as releases within the criteria for routine operation) and might occur with moderate frequency to accident situations that have a theoretical potential for large consequences but are very unlikely. Since it is not practical to consider all possible accidents in detail, the spectrum of potential accidents, ranging in severity from trivial to very serious, is divided into nine classes. Examples of these classes of accidents are presented in Table 2-11 (Ref. 2).

The radiological environmental effects are calculated for each of the above classes using reasonable assumptions, justifiable calculational models and techniques, and realistic assessments of environmental effects. The environmental impact of the nuclear facility is evaluated in relation to the natural background radiation already present.

2.2.1.1 Frequency Classification

The range of accidents considered can be categorized into three groups described as follows:

A. Events of moderate frequency (anticipated operational occurrences) leading to abnormal radioactive releases from the facility.
B. Events of small probability with the potential for small radioactive releases from the facility.
C. Potentially severe accidents of extremely low probability, postulated to establish the performance requirements of engineered safety features and used in evaluating the acceptability of the facility site.

It is highly desirable, for both safety and economic reasons, that group A (moderate-frequency) events, such as partial loss of forced reactor-coolant flow, should result in reactor shutdown with no radioactive release from the fuel and with the plant capable of readily returning to power after corrective action. Analysis and evaluation of these moderate-frequency conditions offer the opportunity of detecting and correcting faults in a particular plant design that might otherwise lead to more serious failures. Safety is certainly enhanced if all those events that can be identified as having a reasonable chance of occurring are shown to be covered by features designed to preclude and to prevent their occurrence and significant damage.

The second group of events, such as a complete loss of forced reactor-coolant flow or partial loss of reactor coolant from small breaks or cracks in pipes, must be shown to present minimal radiological consequences. The actual occurrence of such accidents may, however, prevent the resumption of plant operation for a considerable time because of the potential for failure of the cladding of some fuel rods and the resulting requirement for replacement and cleanup.

Evaluation of these accidents must show that under accident conditions the engineered safety features and containment barriers function effectively to eliminate (or reduce to an insignificant level) the potential for radioactive releases to the environment. In this way, assurance is gained that these unlikely events would lead to little or no risk to public health and safety. These studies also show the effectiveness of safety features designed into the facility to cope with unlikely accidents and show the margins of safety that exist in the design by indicating the type of failures that can be accommodated.

To provide additional defense in depth, extremely unlikely accidents of group C are postulated in spite of their low probability and the steps taken to prevent them. One of these hypothetical accidents is the loss of reactor coolant resulting from postulated major ruptures in the primary coolant system piping (LOCAs). Another is a postulated control-system failure that causes control-rod withdrawal at maximum speed; the resulting rapid power increase beyond design limits could damage the reactor fuel. An accident postulated during refueling is the dropping of an irradiated fuel assembly and consequent damage to the reactor fuel. Accidents postulated for PWRs include system transients resulting from major ruptures in the secondary system piping.

Each of these accidents could result in damage to the fuel-rod cladding and the release of radioactive material from the reactor fuel. A portion of this radioactive material could be transported through leakage paths in the containment barriers, and some portion of it could leak out into the environment. Each type of accident is analyzed to establish that adequate safety features have been engineered into the plant, in the form of passive barriers or active systems, to limit the consequences of a release of fission products from the reactor fuel, and to show that the maximum radiological doses would not exceed the values specified in 10 CFR 100, even under highly pessimistic assumptions.

Experience in such analyses has shown that, for many LWRs, the potential accident that results in the largest calculated potential radiological consequences to the public is the LOCA in which a major failure of one of the large coolant pipes in the primary-coolant system is postulated, along with degraded performance of systems designed to counteract its consequences.

2.2.1.2 Analysis Parameters

For the CESSAR-80 analysis parameters, see Section 15.1.2 in Reference 1.

2.2.1.3 Trip Settings

For the CESSAR-80 safety-related trip settings, see Section 15.1.3 in Reference 1.

2.2.1.4 Radiological Parameters

For the CESSAR-80 radiological parameters, see Section 15.1.4 in Reference 1.

2.2.1.5 Computer Programs

For the CESSAR-80 computer programs used in the safety analysis, see Section 15.1.5 in Reference 1.

2.2.2 GROUP A EVENTS

For details on the safety analysis of Group A events, see Section 15.2 in References 1 and 3.

2.2.3 GROUP B EVENTS

For details on the safety analysis of Group B events, see Section 15.3 in References 1 and 3.

2.2.4 GROUP C EVENTS

For details on the safety analysis of Group C events, see Section 15.4 in References 1 and 3. The offsite doses calculated for design-basis accidents at the Perkins nuclear power station are summarized in Table 2-12.

Table 2-11. Reactor facility classification of postulated accidents and occurrences

No. of Class	Description	Example(s)
1	Trivial incidents	Small spills Small leaks inside containment
2	Miscellaneous small releases outside the containment	Spills Leaks and pipe breaks
3	Radwaste-system failures	Equipment failure Serious malfunction or human error
4	Events that release radioactivity into the primary system	Fuel defects during normal operation Transients outside expected range of variables
5	Events that release radioactivity into the secondary system	Class 4 and heat-exchanger leak
6	Refueling accidents inside the containment	Drop of fuel element Drop of heavy object onto fuel Mechanical malfunction or loss of cooling in transfer tube
7	Accidents to spent fuel outside the containment	Drop of fuel element Drop of heavy object onto fuel Drop of shielding cask--loss of cooling to cask, transportation incident on site
8	Accident-initiation events considered in design-basis evaluation in the safety analysis report	Reactivity transient Rupture of primary piping Flow decrease--steam-line break
9	Hypothetical sequences of failures more severe than Class 8	Successive failures of multiple barriers normally provided and maintained

Table 2-12. Design-basis-accident calculated offsite doses[a] for the Perkins nuclear power station

Accident	2-hr dose at 2500-ft exclusion radius (rem)		30-day dose at 5-mile low-population zone (rem)	
	Thyroid	Whole body	Thyroid	Whole body
Steam-generator tube rupture	7.1×10^{-1}	3.9×10^{-1}	5.4×10^{-2}	3.2×10^{-2}
Loss of external power and/or turbine trip				
30-min release	7.2×10^{-2}	Less than 10^{-4}	5.5×10^{-3}	Less than 10^{-3}
60-min release	1.3×10^{-1}	Less than 10^{-3}	9.9×10^{-3}	Less than 10^{-3}
3-hr release	3.2×10^{-1}	Less than 10^{-3}	2.5×10^{-2}	Less than 10^{-3}
Loss of normal AC power to station auxiliaries	3.4×10^{-1}	Less than 10^{-3}	2.6×10^{-2}	Less than 10^{-3}
Waste gas decay tank rupture	--	5.0	--	1.2×10^{-2}
Rupture of major steam line	3.6×10^{-1}	Less than 10^{-3}	2.8×10^{-2}	Less than 10^{-3}
Design-basis loss of coolant	178	7.8	40	2.2
Fuel-handling accident	5.4	0.51	4.6×10^{-1}	4.4×10^{-2}
Reactor-coolant-waste-tank rupture	9.1×10^{-2}	--	7.0×10^{-3}	--

[a]Conservative estimates.

2.3 ENVIRONMENTAL CONSIDERATIONS

This section addresses the environmental factors associated with land use, water use, thermal discharges and radioactivity releases to the environment resulting from normal reactor operation. The material is organized to reflect two objectives: (a) to identify the information on which the environmental assessment is made and (b) to discuss the significance of the environmental factors that relate to a particular alternative. The environmental information is organized in a form generally consistent with NRC Regulatory Guide 4.2, Revision 2 (Ref. 4).

2.3.1 SITE AND ENVIRONMENTAL INTERFACES (R.G. 4.2/2.0)*

In general, LWRs have been found to be environmentally acceptable at a variety of sites within the United States, although the plant features necessary to achieve acceptability vary widely from site to site.

The environmental effects of NASAP alternatives will be compared with those of the reference LWR by characterizing the differences in normal radioactive effluents and evaluating the projected doses to individuals. From the projected differences, judgments will be presented on the need for new removal technology, larger site areas, and other possible mitigating measures. The comparisons relate to the reference LWR and do not specifically address the question of "as low as reasonably achievable." The comparisons do assess the potential difficulties that each NASAP alternative may encounter from the viewpoint of environmental licensing.

The estimated impacts of each alternative reactor will be presented in terms of a comparison to (i.e., as fractions or multiples of) the estimated impacts of the reference LWR and in terms of the fractions of the impacts attributable to various isotopes.

For the reference LWR, the impacts are presented in terms of the fractions of the impacts (i.e., of the total doses from noble gases, iodines, and particulates and from liquid pathways) attributable to various isotopes. These relative impact values are not strongly dependent on site parameters.

As noted, the reference LWR has been found to be environmentally acceptable. The following sections present descriptions of a model or typical site whose general characteristics are representative of those that have received licensing acceptance.

2.3.1.1 Geography and Demography (R.G. 4.2/2.1)

To characterize environmental factors for the reference LWR (Ref. 5), a reference site is needed. One aspect of that reference site is demography and land and water use.

The reference site for this study was chosen to accept the Combustion Engineering, Inc. (C-E), System 80 plant and to reflect the siting criteria expected for power plants in the period 1985-2000. The reference site is considered to contain two 1,250-MWe plants for estimating both land and water uses. Its characteristics are based mainly on information in two reports published by the U.S. Atomic Energy Commission (AEC), WASH-1355 (Ref. 6) and WASH-1258 (Ref. 7).

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*Topics in this subsection are identified with the corresponding subsections of Regulatory Guide 4.2; for example, Section 2.3.1 of this document corresponds to Section 2.0 of Regulatory Guide 4.2.

Reference 6 (WASH-1355) reviews the site characteristics of some 61 reactor sites for their water and land requirements, including annual water consumption, area occupied, site transmission corridors, and miscellaneous land uses. After this review, a design envelope was developed. Reference 7, a study of some 64 reactor sites, indicated an average site-boundary distance of 0.46 mile (741 meters). On the basis of this study a reference site-boundary distance of 0.50 mile (805 meters) was used for the dose analyses. Table 2-13 presents the envelope of land-use requirements for the reference LWR plant. It is presented for three commonly used cooling-system modes.

Water-use requirements for a reference site are also presented in Reference 6. These requirements include service water and cooling water, consumptive water uses, discharge flow rates, and temperature difference across the main condenser. The values for these parameters (for the above three cooling-system modes) are presented in Table 2-14.

As discussed below, the cooling-tower system is expected to be the one most commonly used in future plants, and the reference site is therefore based on that cooling mode.

Offsite land uses also affect radiological doses. For this study, no typical offsite land-use parameters (e.g., the distance to the nearest cow) were established. The calculations were done by using the site boundary as the nearest place at which all pathways were active. This, however, was simply a calculational tool to allow relative dose values to be determined. Typically, the various pathways (especially the cow-milk pathway) will be 1 to 2 miles from the site boundary in order that radiological doses will be within Appendix I limits.

2.3.1.2 Meteorology (R.G. 4.2/2.3)

To assess normal-operation radioactive effluents, meteorological characteristics of the reference site are needed. Since only normal-operation effluents are included under "environmental considerations," only annual average meteorological parameters are necessary. (Accident meteorology as it affects safety considerations is addressed in Section 2.2.)

The AEC calculated "typical" annual average χ/Q, depleted χ/Q, and D/Q values for over-land diffusion based on meteorological data available for each of several nuclear power plant sites (13 river sites, 6 lakeshore sites, and 6 seashore sites -- see Sections 6.4 and 6.13 of Reference 7). A typical χ/Q value for a 22.5^{o} sector was calculated by averaging the meteorological data over all of the sectors that resulted in over-land trajectories. These results (including depleted χ/Q and D/Q values) have been multiplied by a factor of 2 for this application on the basis of the AEC conclusion that the typical ratio of the average-sector χ/Q value to the maximum-sector χ/Q is 2:1. Maximum-sector values of χ/Q, depleted, χ/Q and D/Q for a "typical" site are summarized in Table 2-15. These values are based on a release at a height of 10 meters, which is equivalent to a ground-level release for the distances (0.5 to 4.5 miles) of interest for this application.

A composite of the "typical" sites is used to represent dispersion conditions at a reference site for this study. Table 2-16 summarizes the χ/Q, depleted χ/Q, and D/Q values for the reference site. These values are in the general range of values presented for the AEC "typical" sites.

2.3.1.3 Hydrology (R.G. 4.2/2.4)

In order to assess the environmental consequences of liquid releases from the reference LWR, some basic hydrologic characteristics of a reference site are needed.

Nuclear power sites are generally located on waterways adequate to provide condenser cooling capacity. Sites can be classified into the following three types: (a) river, including smaller lakes; (b) lakeshore (very large lakes including the Great Lakes); and (c) ocean. According to Reference 7 (Volume 1, Chapter 6), approximately 60% of U.S. nuclear sites are on a river, 15% on a lakeshore, and 25% on a seashore. Since most existing sites are on a river, the reference site for the NASAP study is a river site.

The dilution of radioactivity is affected not only by the type of site but also by the cooling mode used by a power plant: cooling tower, once-through, or a combination of both. Figure 6.3 of Reference 7 illustrates the effect of cooling modes on dispersion factors in a river. After approximately 1,000 meters, the dispersion is almost independent of the cooling mode. Since dispersion at this distance is independent of cooling mode and since future power plant sites will probably be required to have cooling towers, the cooling-tower mode was selected. For this reason the reference site is on a river and uses cooling towers.

To determine the dose from liquid radioactive effluents, dispersion of the effluents must be calculated for the location of the maximum-dose individual. A review of five existing sites on rivers indicated a large variation in dilution factors. They ranged from 1 (no dilution) to cases in which a user pathway was conservatively assumed to be close to the discharge. Regulatory Guide 1.109 (Revision 0 of Reference 8) indicated for high-velocity discharges that a conservative dilution factor of 10 was acceptable to the NRC.

2.3.2 REACTOR AND STEAM-ELECTRIC SYSTEM (R.G. 4.2/3.2)

The Perkins nuclear station, which was selected as a basis for comparison, is a present-generation PWR plant that has received NRC review and is capable of meeting current regulations, including Appendix I to 10 CFR 50.

Basic parameters that describe the plant are given in Table 2-17.

2.3.3 STATION LAND USE

Figure 2-2 is an artist's sketch showing station land use, and Figure 2-3 is a layout diagram. These figures are of a four-unit (approximately 1,000 MWe each) plant and therefore indicate land usage corresponding to four times the 1,000-MWe reference unit for this study. Figure 2-4 shows a diagram of the reference LWR (C-E System 80). Table 2-18 gives areas for land usages associated with typical LWR plants.

2.3.4 STATION WATER USE (R.G. 4.2/3.3)

The largest single use of water is for makeup to the heat-dissipation system. Much smaller amounts are required for the plant (after demineralization) and for laundry, showers, and sanitary facilities. Water use is tabulated in Table 2-19.

2.3.5 HEAT-DISSIPATION SYSTEM (R.G. 4.2/3.4)

There are several types of heat-dissipation system that may be used, depending on site conditions and other factors. The wet natural-draft cooling tower with fresh-water makeup was assumed for this report.

A typical natural-draft cooling tower for a 1,000-MWe unit would have a single shell with a height of about 550 feet and a maximum shell diameter of about 410 feet. Heat is dissipated to the atmosphere by evaporation and by sensible-heat transfer. Heat dissipation by evaporation is larger, but the balance between the two depends on air temperature and humidity. The average rate of water use, therefore, varies from month to month. Blowdown is required to limit the concentration of solids in the circulating water. For the "typical" plant discussed here, a maximum concentration factor of 5 is used, though other values are frequently found. Tower design data are shown in Table 2-20 for a site in the north central United States.

Circulating water is periodically chlorinated to control algae and other slime-forming microorganisms. Typically, chlorine is added as required to achieve a residual free chlorine content of 0.5 to 1.0 ppm for 1 to 2 hours per day. The cooling-tower blowdown may have a small residual free chlorine content during periods of chlorination.

2.3.6 RADWASTE SYSTEMS AND SOURCE TERMS

Sources of radioactivity, release paths, and processing systems are described briefly in the following sections. Quantities of radioactivity released, taken from Reference 4, are also tabulated below. Table 2-21 lists the principal assumptions and plant parameters that were used in the calculations. These parameters are for 1,300 MWe, and the results were then normalized to 1,000 MWe.

2.3.6.1 Source Term (R.G. 4.2/3.5.1)

The sources of radioactivity in the plant are fission products and materials in the reactor core and coolant that become activated by neutron irradiation. Small amounts of fission products are released to the reactor coolant through defects in the fuel cladding, and activated core materials are released to the coolant by corrosion. Two isotopes of particular interest are carbon-14 and tritium. Carbon-14 is produced by an (n,p) reaction of nitrogen-14 and by an (n,α) reaction of oxygen-17. Tritium is produced by ternary fissions and by the reaction of neutrons with boron and lithium in the reactor coolant. Radioactivity is removed from the reactor coolant by cleanup in the chemical and volume control system (CVCS) and by fluid removal from the system by leakage and by the shim bleed stream. Figure 2-5 is a block diagram showing the potential paths for radioactivity removal from the reactor-coolant system. The leakage paths serve as sources of radioactivity to other plant systems.

Figure 2-6 shows the steam and power-conversion system components that are most important from the standpoint of radioactivity in the system and releases to the environment. Noble gases and small amounts of iodine that leak into the steam generator are carried out with the steam, pass through the turbine and condenser, and are removed from the condenser by the air-removal system. A filter system removes most of the iodine, leaving the noble gases and a small amount of iodine to be discharged into the atmosphere. Noble gases and iodine also reach the atmosphere directly in a small amount of steam leakage. Nonvolatile radionuclides collect in the steam-generator liquid. They are removed in the blowdown stream, which goes to the condenser and there it mixes with the condensate. About 65% of the condensate stream passes

through the condensate-polishing demineralizer as it is returned to the steam generator. The nonvolatile radionuclides are collected in the condensate-polishing demineralizers.

Figure 2-7 shows the boron-recycle system which collects and processes for recycling water from the reactor-coolant system. Radioactivity in the processed stream is removed by the pre-holdup ion exchanger, the gas stripper, the boric acid concentrator, and the boric acid condensate ion exchanger. Either effluent stream from the boric acid concentrator (or both) may be routed to the waste-disposal system as required for tritium control or other purposes.

2.3.6.2 Liquid-Radwaste System (R.G. 4.2/3.5.2)

The miscellaneous-liquid-waste system (Figure 2-8) processes liquid wastes from the sources described above as well as from other sources: laundry and shower wastes, equipment drains, and floor drains. Laundry and shower wastes and condensate from the containment coolers are collected and monitored. If there is no significant radioactivity, these wastes are filtered and discharged with the laundry and shower wastes. If significant activity is detected, these streams are routed to the equipment discussed below for processing.

Waste for processing is collected in the waste tanks and passed through particulate and carbon filters to remove oil and other organics. It then goes to an evaporative waste concentrator. The concentrates (bottoms) are sent to the solid-waste-handling system for solidification and disposal. The distillate is passed through an ion exchanger and is then stored in a waste-condensate tank for monitoring and discharge. Turbine-building drainage is collected and discharged. The quantities of important radionuclides, calculated with the GALE computer code, are shown in Table 2-22. Assumptions for these calculations, such as flow paths, are shown on the figures.

Discharges from the miscellaneous-liquid-waste system are normally directed to the body of water on which the plant is sited.

2.3.6.3 Gaseous-Waste System (R.G. 4.2/3.5.3)

The gaseous-waste system is shown in Figure 2-9. Compressed storage is provided for gases removed from the gas stripper of the boron recovery system, the volume-control tank, and the reactor drain tank. The gas from the first two is hydrogen containing small (volumetrically) amounts of fission products. The gas from the reactor drain tank is nitrogen cover gas, displaced as the tank is filled. A recombiner is provided to allow removal of hydrogen and/or oxygen from the stored gases. The gases from the gas stripper are collected and compressed. The hydrogen is removed in the recombiner to leave a small volume of fission-product gases that is returned to one of the storage tanks for long-term holdup. The gases from the volume-control tank can be processed similarly.

Nitrogen cover gas displaced by filling the reactor drain tank is compressed in the gaseous-waste system. Hydrogen can be removed by recombination and the nitrogen stored for reuse as a cover gas.

In addition to these major sources of radioactive gases, there are the leakage paths discussed earlier. These are small leaks from the reactor-coolant system to the containment, small leaks of reactor coolant to the auxiliary building, and small leaks from the reactor-coolant system to the steam and power-conversion system.

The containment is equipped with an internal, recirculating filter system containing particulate, absolute, and charcoal filters. This system removes particulates and iodine before containment purge. The containment is vented and/or purged through similar filter systems.

The ventilation system of the auxiliary building also contains particulate, absolute, and charcoal filters. This system filters air exhausted from areas that might become contaminated by reactor-coolant leakage. Most of the gaseous activity leaking into the steam and power-conversion system will be contained in air removed from the condenser.

This effluent is also filtered by particulate, absolute, and charcoal filters. Total gaseous releases of radioactivity have been calculated (Ref. 5). These results are shown in Table 2-23.

2.3.6.4 Solid Radwastes (R.G. 4.2/3.5.4)

Materials transferred to the solid-radwaste system for disposal include spent demineralizer resins and evaporator concentrates. These will be solidified for off-site disposal. Other solid wastes (contaminated clothing, paper, and filters) are also sent off the site for disposal. It is estimated that a total of 1,050 fifty-five-gallon drums will be shipped off the site for disposal each year.

2.3.6.5 Comparison with Predicted Releases from Other Studies

Several other studies have been made of potential releases of radioactive material from normal operation of nuclear power plants. References 9, 10, and 11 describe three of these. These studies have considered a variety of reactor and plant designs, assumptions, calculational techniques, and other topics. The results, in terms of liquid and gaseous releases, are shown in Tables 2-24 and 2-25. For comparison, the actual release experience of H. B. Robinson Unit 2 (Ref. 12) is also shown. These data are the average of the first years of operation. It should be noted that Robinson is an older unit and has substantially less installed waste-treatment capability than was assumed for the other cases.

A considerable variation in predicted releases of some radionuclides is seen in the tables. Overall, however, reasonable agreement between the studies is found, considering the differences in assumptions and calculational methods. Wide variations among operating plants and from year to year for the same plant are also found. Thus, releases of radioactivity cannot be precisely predicted, but expected ranges are defined. From experience, it is reasonable to conclude that the range of releases defined by the various studies is representative of releases from present-generation plants or is conservative with respect to actual releases.

2.3.7 CHEMICAL AND BIOCIDAL WASTES (R.G. 4.2/3.6)

The primary sources of chemical and biocidal wastes are the cooling-tower blowdown stream and the chemical effluents from regeneration of demineralizers that treat makeup water. The cooling-tower blowdown stream contains dissolved solids that entered in the makeup stream and are concentrated by evaporation in the cooling towers. This stream will also intermittently contain a small chlorine residual from chlorination of the condenser cooling water. This is discussed in Section 2.1.2.5.

Acid and caustic soda solutions are used for demineralizer regeneration. These wastes are held up and neutralized before discharge. They contain no radioactivity.

2.3.8 EFFECTS OF OPERATION OF THE HEAT-DISSIPATION SYSTEM (R.G. 4.1/5.1)

The temperature of the blowdown water is primarily a function of the wet-bulb temperature of the air drawn into the cooling towers. For example, for the Perkins (Ref. 5) plant and site in North Carolina, the estimated temperature of the blowdown was 4-15°F above the river temperature in summer and up to 3°F above the river temperature in winter. The effect of this heat (e.g., the amount of the aqueous environment that is heated 3°F or more above ambient) and its impact on the aquatic ecosystem depend on the receiving water body. For the purpose of this NASAP comparison study, more or less heat released to the water body will make siting more or less easy.

Another effect of the heat-dissipation system is water consumption; the effect was considered in Section 2.1.2.4 above. The effects of biocide treatment necessary in the operation of cooling towers are considered in Section 2.1.2.10.

There are several other effects of cooling-tower operation, including increased fogging, increased icing, increased precipitation, aesthetic effect of the tower and the visible plume, terrestrial ecosystem impact of drift and shadowing by the plume, impingement of fish on the intake, and entrainment of small aquatic life forms. These effects are generally minor; therefore the anticipated differences in these minor effects among the alternatives in this NASAP study are not expected to be of importance.

2.3.9 RADIOLOGICAL IMPACT FROM ROUTINE OPERATION (R.G. 4.2/5.2)

A characterization of normal operating effluents in terms of their contribution to doses to individuals in the nearby vicinity of the site is useful so that comparisons of the significance of differences in normal effluents between the reference LWR reactor and each of the NASAP alternatives may be made. Computational techniques consistent with NRC Nuclear Regulatory Guide 1.109 (Ref. 8) were used. It should be noted that groupings of radionuclides have been made from two standpoints. The first is radionuclides with similar environmental dispersion characteristics (e.g., noble gases, particulates, and iodines). The second is radionuclides that make a significant contribution to the individual dose. The comparison of population dose was not made because the radionuclides that contribute most to the individual dose are also most likely to contribute most to population dose. The results given here for the reference LWR are intended only to characterize the contribution of specific radionuclides to typically predicted individual doses around sites in the United States at this time. The results are indicated in the form of fractional contribution to the total individual dose rather than the absolute value of those doses since the objective is to compare the significance of one radionuclide relative to others. This process also negates the need to identify more specific aspects of the model site and yet provides the opportunity to characterize the significance relative to dose of increases or decreases in radioactive effluents for alternatives within the NASAP evaluation other than the reference LWR.

2.3.9.1 Exposure Pathways (R.G. 4.2/5.2.1)

The pathways by which man can be exposed to radiation from a nuclear power plant are shown in Figure 2-10. The exposure pathways can be grouped into those associated with liquid emissions, those associated with gaseous emissions, and those

involving exposure to direct radiation from the plant or from radioactive materials during transport to and from the plant.

In order to evaluate the impact of the alternatives on the liquid emission doses, it is necessary to identify pathways that could potentially be affected. For this study, it was assumed that a maximum-dose individual would be involved in the following activities downstream of the plant:

1. Drinking water
2. Eating fish grown in the immediate area
3. Participating in shoreline activities such as picnicking and shoreline fishing

Other pathways were not considered since they contribute an extremely small amount of the maximum dose to the individual.

Gaseous releases from a nuclear power plant may result in the exposure of an individual through the following pathways:

1. Air submersion
2. Inhalation
3. Ground-plane exposure from the deposition of radioiodines and particulates
4. Ingestion of food from such sources as
   (a) Vegetables and fruits subject to the direct deposition of particulates and radioiodines
   (b) Cow's milk or goat's milk containing particulates and radioiodines transferred to the milk by ingestion of fresh or stored forage by the animal
   (c) Meats containing particulates and radioiodines transferred to the animal by ingestion of fresh or stored forage

All of these pathways were considered to be present in the vicinity of the model site. Each pathway was considered to be present at the site boundary (805 meters) for which the dose analyses were performed.

2.3.9.2 Dose-Rate Estimates for Man (R.G. 4.2/5.2.4)

As discussed previously, it was assumed that a dilution factor of 10 was applicable at all liquid-pathway locations. Using the models and usage factors discussed in Regulatory Guide 1.109 (Ref. 8) and implemented in the computer code LADTAP, the maximum individual dose was evaluated. The largest total-body dose was found for the adult, and the largest organ dose, the thyroid dose, was found for the infant. Table 2-26 presents the breakdown of the dose by radionuclide.

Doses resulting from the release of gaseous effluents were calculated at the site boundary (805 meters) assuming that an infant, child, teenager, and adult all reside at this location and assuming that all potential gaseous pathways existed at the site boundary. The doses were calculated for noble gases, and for radioiodines and particulates. The doses are presented in Tables 2-27, 2-28, and 2-29, as percent contribution by radionuclide for various organs. The dose contributions to a child were presented because a child may receive exposure from all pathways (i.e. ground-plane exposure, inhalation, and ingestion of meat, vegetables, and milk). An infant, on the other hand, can be exposed only by the ingestion of milk and by inhalation. Table 2-27 gives the contribution by isotope to the total-body and skin dose. Tables 2-28 and 2-29 present

the contribution of various radioiodines and particulates to the organ doses of a child and an infant, respectively.

Direct external radiation exposure to people outside the site of a typical nuclear power plant is insignificant and well within the requirements of 10 CFR 20 and 40 CFR 190. This is because plant shielding generally limits exposure rates at the outside of building walls to less than 2.5 mrem/hr. A representative value for direct external radiation exposure to a person located at the boundary of a typical nuclear plant site is less than 014 mrem/yr at a distance of 2,500 feet. This result is based on calculations made for the Perkins nuclear power station (Units 1, 2, and 3). The source of radioactivity considered in these calculations is the residual radioactivity present in water stored in storage tanks outside the plant buildings.

2.3.10 EFFECTS OF CHEMICAL AND BIOCIDAL DISCHARGES

Liquid-effluent content must comply with the Environmental Protection Agency (EPA) regulations; these effluents include those from demineralizer systems, wastewater and waste solutions from cleaning operations, boiler blowdown, and cooling-tower blowdown. Available technology generally allows necessary cleanup of these liquids. In the case of the cooling-tower blowdown, a problem sometimes arises because the cooling towers concentrate existing pollutants in the cooling-water body; the cooling-tower blowdown may therefore contain an unacceptably high concentration. For the purpose of this NASAP comparison study, the effects of chemicals and biocides will probably not be important and have therefore not been included in the study.

2.3.11 OCCUPATIONAL EXPOSURE

Compilations and studies of historical data (Refs. 13-15) show that the workers in PWR plants are exposed to an integrated radiation dose that averages 400 to 500 man-rem/per unit. Most of this dose is incurred in maintenance and repair activities and much smaller amounts in reactor operation, waste processing, and refueling (see Table 2-30). These data also show significant numbers of individuals with exposures in the range 1 to 10 rem/yr. Exposures of this magnitude may be expected for the unit discussed here.

Table 2-13. Land-use requirements for LWR sites[a]

Land use (reactor phase)	Area (acres)		
	Cooling tower	Cooling pond	Once through
Site land requirements (total)	1,100 ± 900	11,000 ± 2,600	1,100 ± 900
Transmission route	1,800 ± 2,200	1,800 ± 2,200	1,800 ± 2,200
Disrupted land surface (site)	350 ± 610	4,800 ± 2,600	350 ± 610
Committed land	130 ± 100	130 ± 100	130 ± 610

[a]For two 1,250-MWe units.

Table 2-14. Water-use requirements for LWR sites[a]

Water use (reactor phase)	Cooling tower	Cooling pond	Once through
Cooling-water supply, cfs	89 ± 54	1,200 ± 1,700	4,000 ± 1,000
Service-water supply, cfs	170 ± 40	170 ± 40	170 ± 40
Discharge flow rate, cfs	210 ± 110	1,800 ± 1,900	4,200 ± 1,100
Consumptive use, cfs	47 ± 4	51 ± 12	77 ± 15
Temperature difference (ΔT) across condenser, ^{o}F	24 ± 11	22 ± 5	24 ± 8

[a]For two 1,250-MWe units.

Table 2-15. Maximum sector annual average dispersion estimates for a typical site

Downwind distance (miles)	χ/Q (sec/m^3)			Depleted χ/Q (sec/m^3)			D/Q (m^{-2})
	Lake	Sea	River	Lake	Sea	River	
0.5	3.8×10^{-6}	6.6×10^{-6}	8.4×10^{-6}	3.6×10^{-6}	6.2×10^{-6}	7.9×10^{-6}	2.8×10^{-8}
1.5	6.6×10^{-7}	1.4×10^{-6}	1.8×10^{-6}	5.7×10^{-7}	1.2×10^{-6}	1.6×10^{-6}	4.2×10^{-9}
2.5	3.0×10^{-7}	6.4×10^{-7}	9.6×10^{-7}	2.5×10^{-7}	5.2×10^{-7}	7.9×10^{-7}	2.0×10^{-9}
3.5	1.7×10^{-7}	3.8×10^{-7}	5.6×10^{-7}	1.3×10^{-7}	3.0×10^{-7}	4.4×10^{-7}	1.2×10^{-9}
4.5	1.2×10^{-7}	2.4×10^{-7}	4.0×10^{-7}	9.0×10^{-8}	1.8×10^{-7}	3.0×10^{-7}	7.2×10^{-10}

Table 2-16. Maximum sector annual average atmospheric dispersion estimates for the reference site

Downwind distance (miles)	χ/Q (sec/m^3)	Depleted χ/Q (sec/m^3)	D/Q (m^{-2})
0.5	1.3×10^{-5}	1.1×10^{-5}	1.3×10^{-7}
1.5	1.5×10^{-6}	1.1×10^{-6}	1.1×10^{-8}
2.5	5.0×10^{-7}	3.6×10^{-7}	3.0×10^{-9}
3.5	2.6×10^{-7}	1.7×10^{-7}	1.3×10^{-9}
4.5	1.7×10^{-7}	1.1×10^{-7}	7.9×10^{-10}

Table 2-17. Model plant parameters

Type	PWR
Fuel cycle	Once through
Burnup, MWd/MT	30,000
Base reactor, MW	3,817
Electrical output, MW	1,270
Normalized electrical output, MW	1,000
Heat rate, Btu/kW-hr	10,293
Heat-dissipation rate at 1,000 MW, Btu/hr	6.7×10^9

Table 2-18. Land-use requirements for the reference LWR site[a]

Land use	Acres
Site land requirements (total)	1,100 $\pm$ 900
Transmission route	1,800 $\pm$ 2,200
Disrupted land surface (site)	350 $\pm$ 610
Committed land (station)	130 $\pm$ 100

[a]For two 1,250-MWe units as described in WASH-1355 (Ref. 7).

Table 2-19. Water use

Use	Quantity (gpm)
Makeup to cooling-tower system (maximum)	14,500
Makeup to cooling-tower system (average)	8,500
Input to laundry, hot showers, sanitary and potable water	3
Input to demineralized-water system	140
Demineralized-water system waste	10

Table 2-20. Heat-dissipation system design data

Type of cooling tower	Wet natural draft
Heat-dissipation rate (maximum, full power), Btu/hr	6.7×10^9
Evaporation and drift (maximum, full power), gpm	11,500
Evaporation and drift (annual average), gpm	6,800
Blowdown (maximum), gpm	3,000
Blowdown (annual average), gpm	1,700

Table 2-21. Principal parameters and conditions used in calculating releases of radioactive material (3,817 MWt)

Operating-power fission-product source term, %	0.25
Primary system	
Mass of coolant, lb	5.7×10^5
Letdown rate of CVCS, gpm	84
Shim bleed rate, gpm	3.1
Leakage rate to secondary system, lb/day	110
Leakage rate to auxiliary building, lb/day	160
Leakage rate to containment building, lb/day	240
Frequency of degassing for cold shutdowns (per year)	2
Secondary system	
Steam flow rate, lb/hr	1.7×10^7
Mass of steam/steam generator, lb	1.8×10^4
Mass of liquid/steam generator, lb	1.6×10^5
Secondary coolant mass, lb	2.8×10^6
Rate of steam leakage to turbine building, lb/hr	1.7×10^3
Dilution flow, gpm	4.0×10^3
Containment-building volume, ft^3	3.3×10^6
Frequency of containment purges per year	4
Recirculation system	
Flow rate, cfm	1.8×10^4
Operating period per purge, hr	16
Mixing efficiency, %	70
Iodine partition factors (gas/liquid)	
Leakage to containment building	0.1
Leakage to auxiliary building	0.005
Steam leakage to turbine building	1
Steam generator (carryover)	0.01
Main condenser air ejector	0.0005

Table 2-21. Principal parameters and conditions used in calculating releases of radioactive material (3,817 MWt) (continued)

	Decontamination Factors (Liquids)		
	Boron recycle	MLWMS[b]	SGB/VCC[a] (condensate treatment)
I	1×10^5	1×10^4	1×10^2
Cs, Rb	2×10^4	1×10^5	1×10^1
Mo, Tc	1×10^5	1×10^6	1×10^4
Y	1×10^4	1×10^5	1×10^3
Others	1×10^6	1×10^5	1×10^2

	All nuclides except iodine	Iodine
Decontamination factors[c]		
Waste evaporator	10^4	10^3
BRS evaporator	10^3	10^2

	Cation[d]	Anion[d]	Cs, Rb
Mixed-bed demineralizer (Li_3BO_3)	10	10	2
Mixed-bed demineralizer (H^+OH^-) DF	$10^2(10)$	$10^2(10)$	2(10)
Cation demineralizer	$10^2(10)$	1(1)	10(10)
Anion demineralizer	1(1)	$10^2(10)$	1(1)
Powdex	10(10)	10(10)	1(10)

Removal by plateout	Removal factor
Mo, TC	10^2
Y	10

[a]Steam-generator blowdown/volatile coolant chemistry.

[b]Miscellaneous liquid waste management system.

[c]For two demineralizers in series, the decontamination factor for the second demineralizer is given in parentheses.

[d]Does not include Cs, Mo, Y, Rb, Tc.

Table 2-22. Liquid radioactive source term normalized to 1,000 MWe

Isotope	Ci/yr	Isotope	Ci/yr
Br-82	0.00007	Cs-138	0.00002
Br-83	0.0001	Ba-139	0.00004
Rb-86	0.00004	Ba-140	0.002
Sr-89	0.0002	La-140	0.0001
Sr-91	0.00006	Ce-141	0.00002
Y-91m	0.00002	Ce-143	0.00001
Y-91	0.0001	Pr-143	0.00002
Zr-95	0.00002	Ce-144	0.00005
Nb-95	0.00002	Pr-144	0.00002
Mo-99	0.0003	Nd-147	0.00001
Tc-99m	0.0003	Na-24	0.0001
Ru-103	0.00002	P-32	0.00002
Rh-103m	0.00002	P-33	0.0001
Te-125m	0.00001	Cr-51	0.0003
Te-127m	0.0001	Mn-54	0.00006
Te-127	0.0002	Mn-56	0.001
Te-129m	0.0005	Fe-55	0.0003
Te-129	0.0003	Fe-59	0.0002
I-130	0.0004	Co-58	0.003
Te-131m	0.0005	Co-60	0.0004
Te-131	0.0001	Ni-65	0.00002
I-131	0.14	Nb-92	0.00006
Te-132	0.01	Sn-117m	0.00002
I-132	0.01	W-185	0.00002
I-133	0.1	W-185	0.0005
I-134	0.00007	Np-239	0.0002
Cs-134m	0.00003		
Cs-134	0.01	All others	0.0001
I-135	0.02		
Cs-136	0.005	Total (except tritium)	0.3
Cs-137	0.01		
Ba-137m	0.01	Tritium	270

Note: Isotopes with discharges of less than 10^{-5} Ci/yr per unit are not identified but are included in the "all others" term.

Table 2-23. Gaseous radioactive source term normalized to 1,000 MWe

Isotope	Ci/yr
Kr-83m	1
Kr-85m	11
Kr-85	380
Kr-87	2
Kr-88	14
Kr-89	1
Xe-131m	44
Xe-133m	80
Xe-133	7,200
Xe-135m	1
Xe-135	50
Xe-137	1
Xe-139	1
I-131	0.05
I-133	0.06
Tritium	580
C-14	6
Particulates	0.05

Table 2-24. Comparison of liquid releases

Isotope	Perkins, 1,300 MWe	GESMO,[a] 1,000 MWe	EPA,[b] PWR, 1,000 MWe	Robinson 2, 665 MWe
I-131	0.18	0.052	0.21	
Te-132	0.01	0.00073		
I-132	0.01	0.0023		
I-133	0.1	0.043	0.1	1.0 (total for all isotopes)
Cs-134	0.01	0.025	0.6	
I-135	0.02	0.011		
Cs-137	0.01	0.033	0.5	
Ba-137m	0.01	0.0084		
Tritium	350	240	380	410

[a]Data from Reference 9.
[b]Data from Reference 10.

Table 2-25. Comparison of gaseous releases

Isotope	Perkins, 1,300 MWe	Palo Verde,[a] 1,300 MWe	GESMO,[b] 1,000 MWe	EPA,[c] 1,000 MWe	Robinson, 665 MWe
Kr-85	494	1,040	470	800	
Kr-88	19	26	23	28	
Xe-131m	57	27	82		
Xe-133	9,420	318	12,000	2,600	1,300
Xe-135	69	28	86		(total
I-131	0.068	0.0095	0.025	0.043[d]	for all
I-133	0.08	0.013	0.023	0.022[d]	isotopes)
Tritium	760	350	1,100	----	
C-14	8	----	8	----	

[a]Data from Reference 11.
[b]Data from Reference 9.
[c]Data from Reference 10.
[d]Elemental only--organic releases 1.3 and 1.1 Ci/yr (I-131 and I-133, respectively).

Table 2-26. Contributions to dose due to liquid effluents

Isotope	Percentage	
	Adult total body	Infant thyroid
Tritium	67	3
I-131	1	87
I-133	(a)	8
Cs-134	15	(a)
Cs-136	1	(a)
Cs-137	8	(a)
Others	8	2
Total	100	100

[a]Less than 1%.

Table 2-27. Contribution of noble-gas emissions to total-body and skin doses

Isotope	Contribution (%) to organ dose	
	Total body	Skin
Kr-83m	(a)	(a)
Kr-85m	(a)	(a)
Kr-85	(a)	10
Kr-87	(a)	1
Kr-88	8	4
Kr-89	1	(a)
Xe-131m	(a)	(a)
Xe-133m	1	2
Xe-133	85	79
Xe-135m	(a)	(a)
Xe-135	4	3
Xe-137	(a)	(a)

[a]Less than 1%.

Table 2-28. Contributions of radioiodines and particulates to the thyroid dose of a child

Isotope	Contribution (%) to thyroid
I-131	91
I-133	1
Tritium	3
C-14	5

Table 2-29. Contribution of radioiodines and particulates to thyroid dose of an infant

Isotope	Contribution (%) to thyroid
I-131	91
I-133	1
Tritium	3
C-14	5

Table 2-30. Distribution of radiation exposure by activity (1975 data)

Activity	Percentage of exposure
Reactor operations	11
Maintenance	72
In-service inspection	3
Waste processing	7
Refueling	8

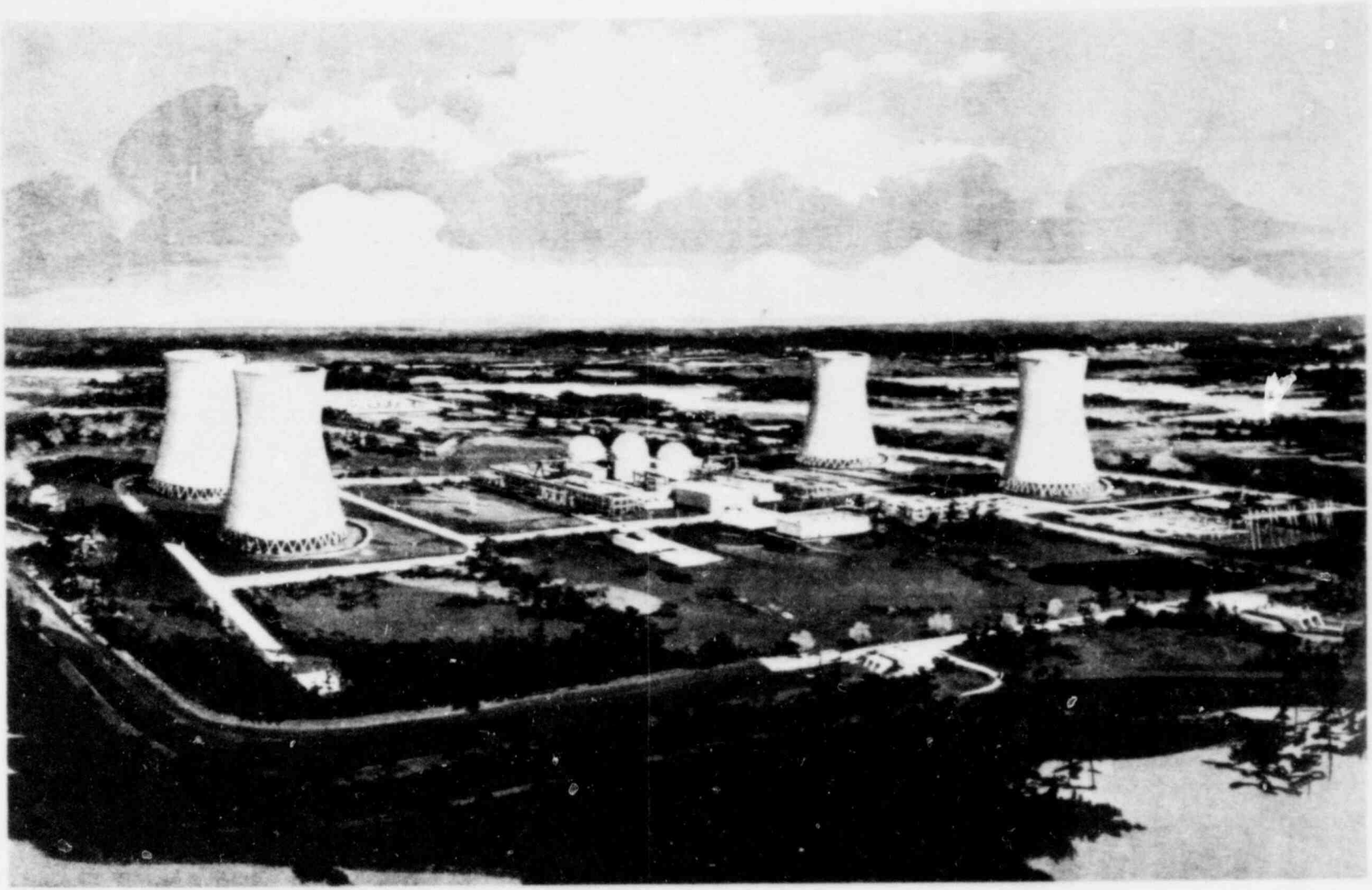


Figure 2-2. Artist's sketch of a typical four-unit LWR plant.

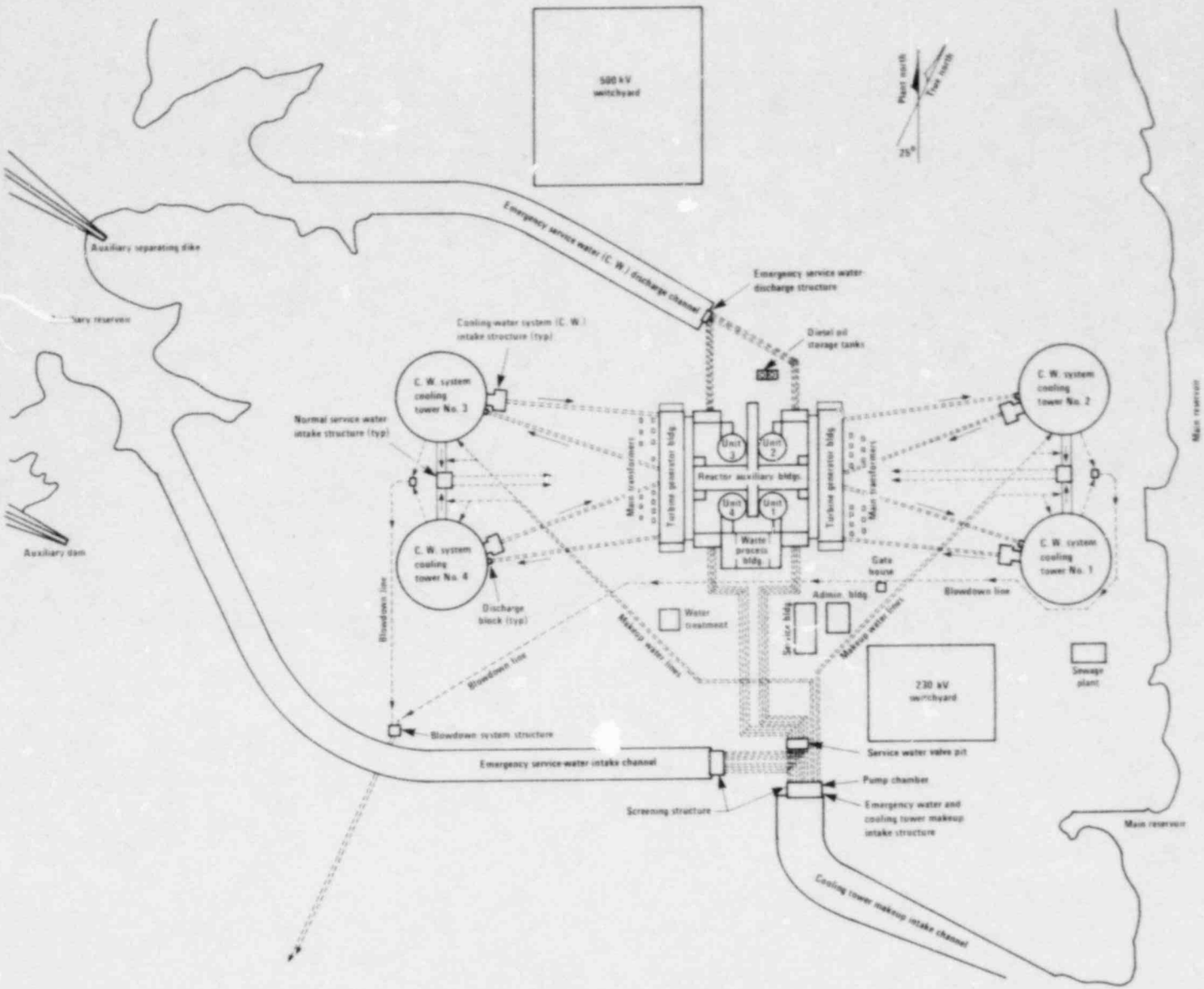


Figure 2-3. Site layout for a typical four-unit LWR plant.

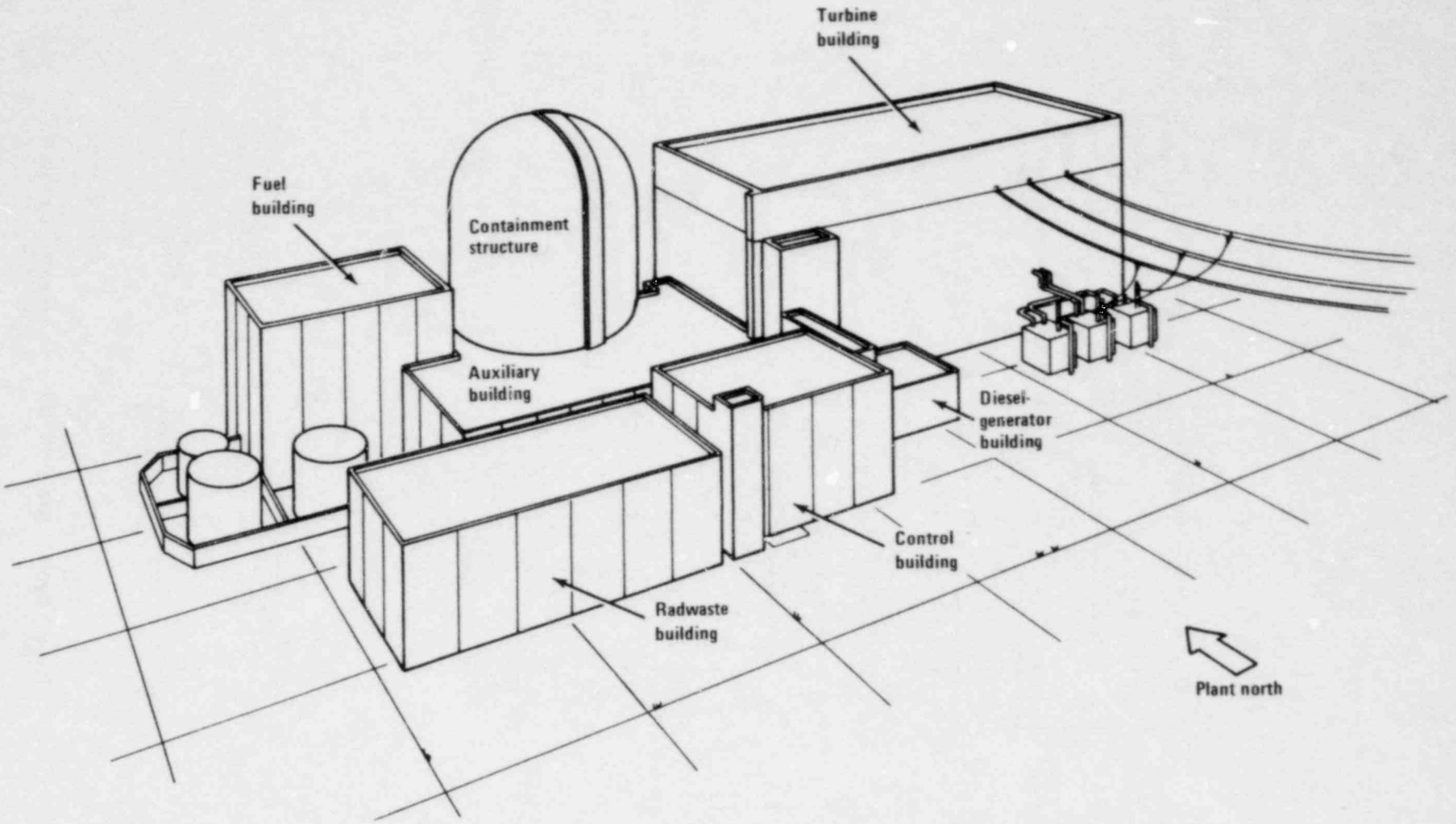


Figure 2-4. Combustion Engineering System 80 plant.

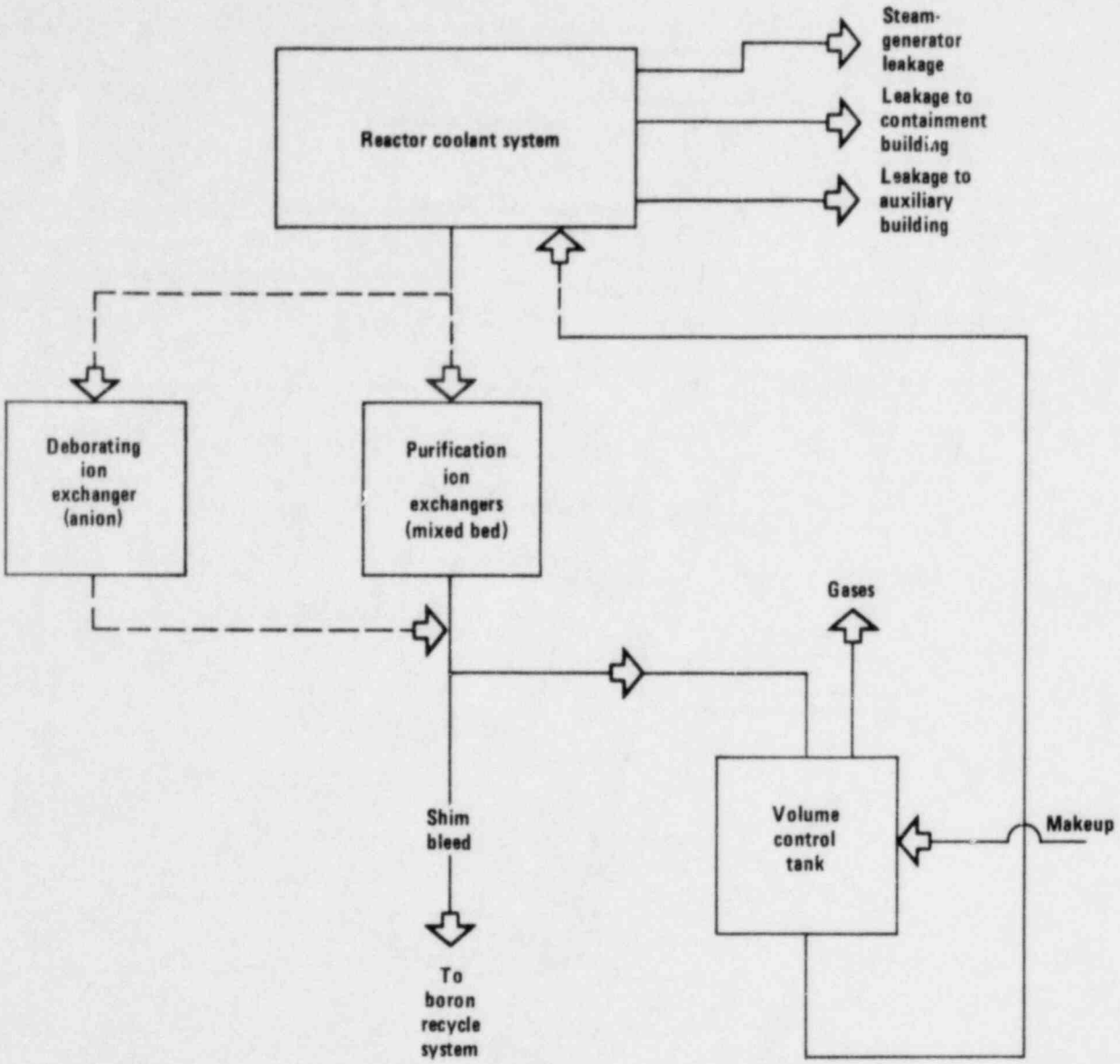
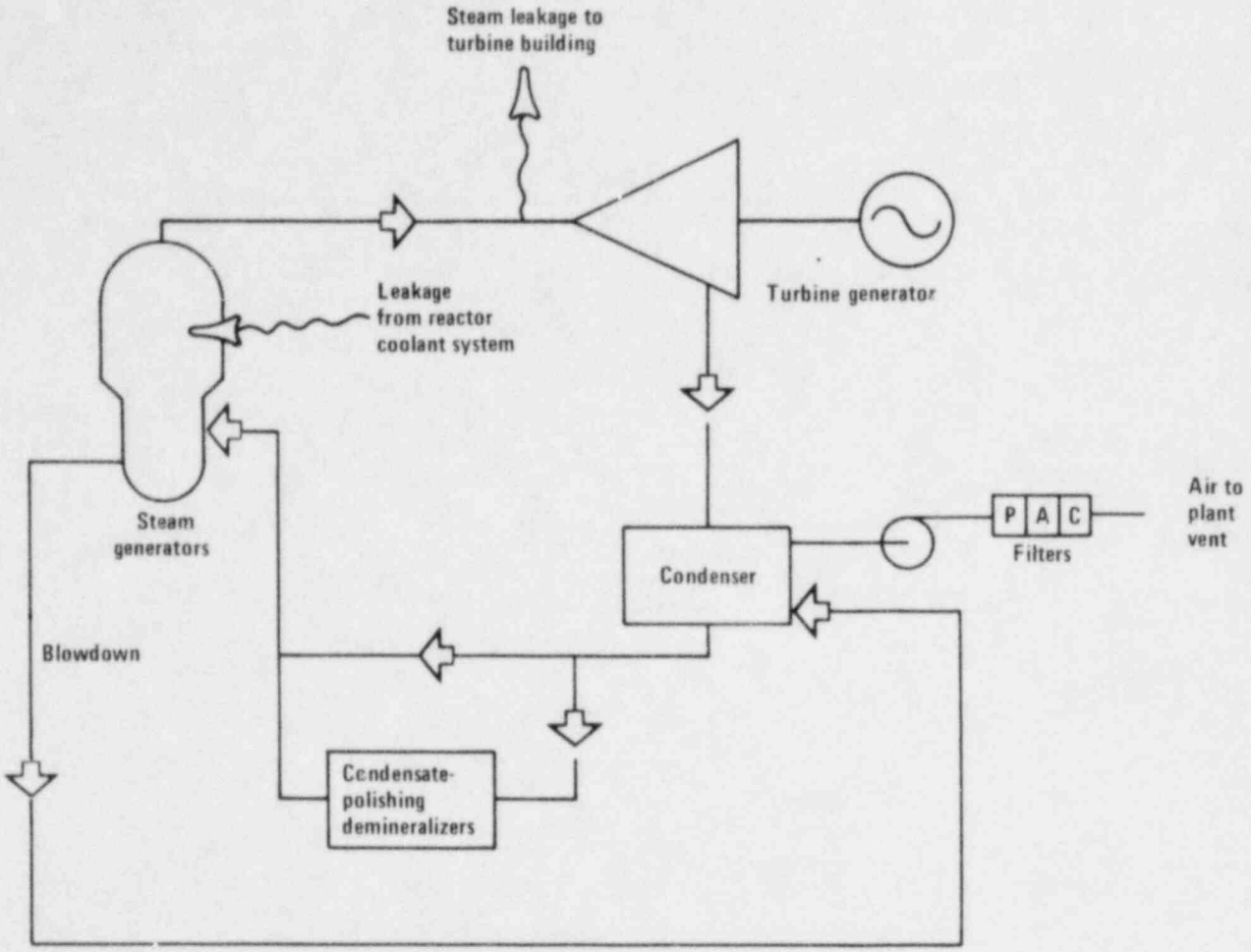


Figure 2-5. Reactor coolant and chemical and volume control system.



Legend:

P = particulate
A = air
C = carbon

Figure 2-6. Steam and power conversion system with sources of radioactivity.

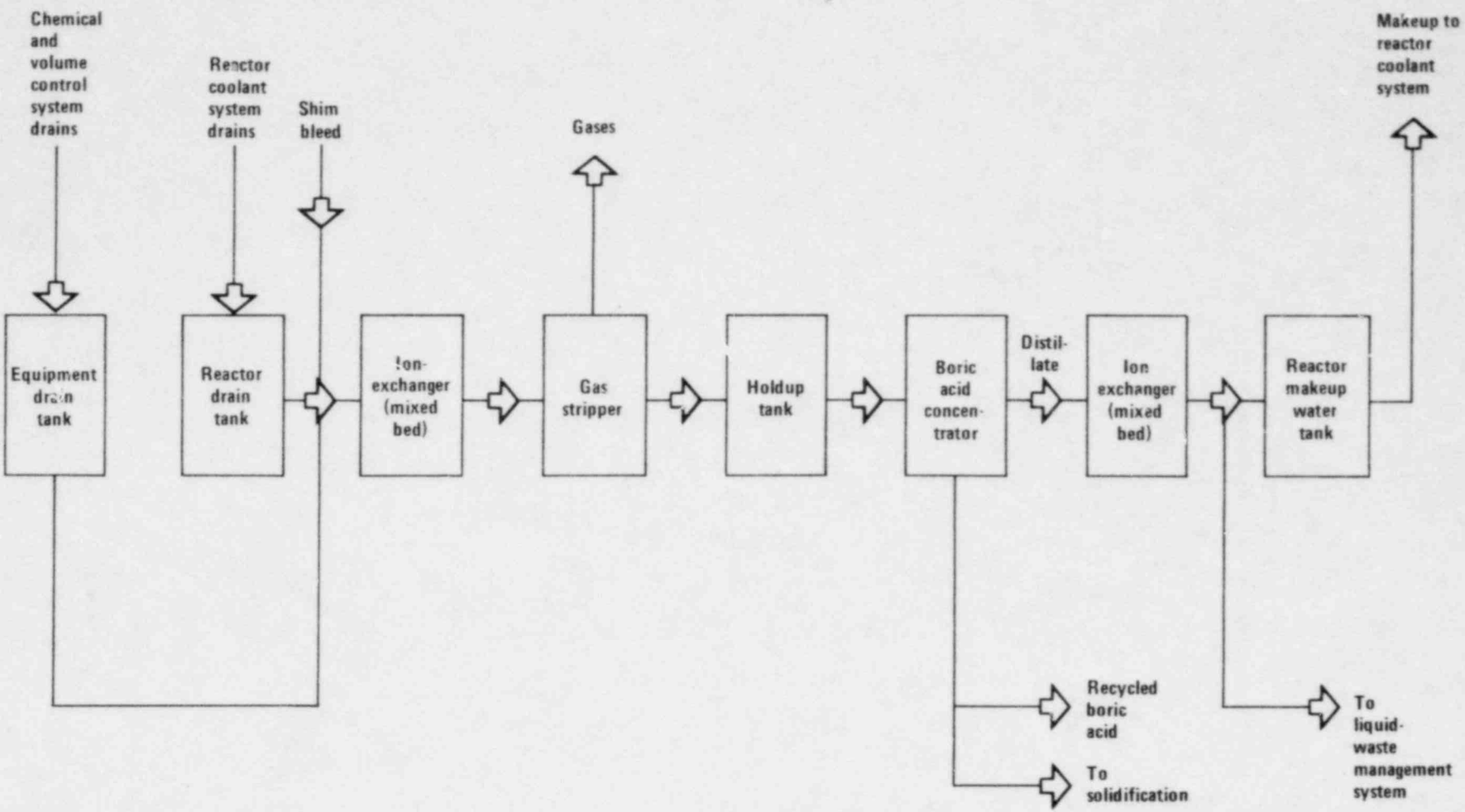


Figure 2-7. Boron recycle system.

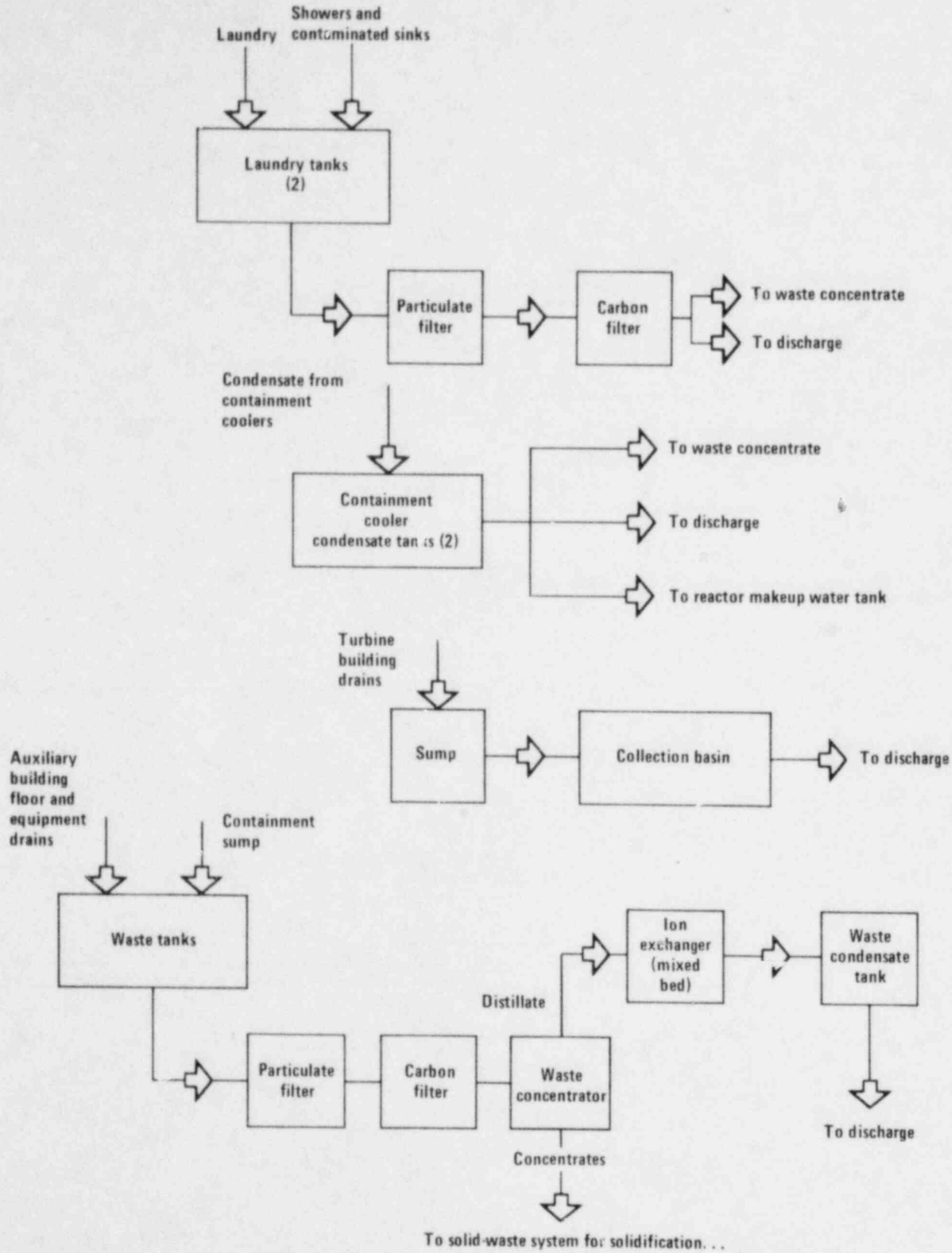
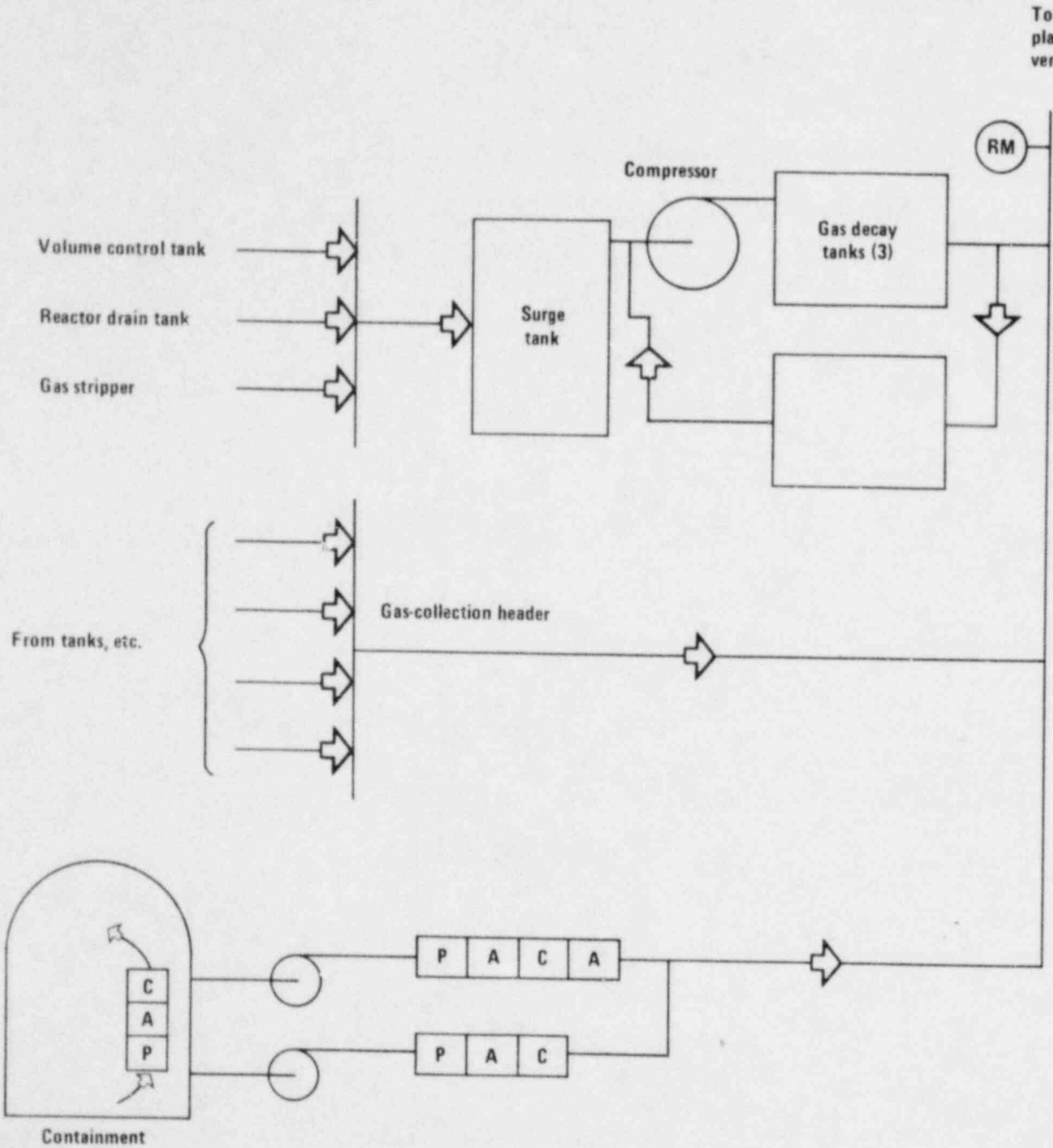


Figure 2-8. Miscellaneous-liquid-waste management system.



Abbreviations: RM = radiation monitor; P = particulate; A = air; C = carbon

Figure 2-9. Gaseous-waste management system.

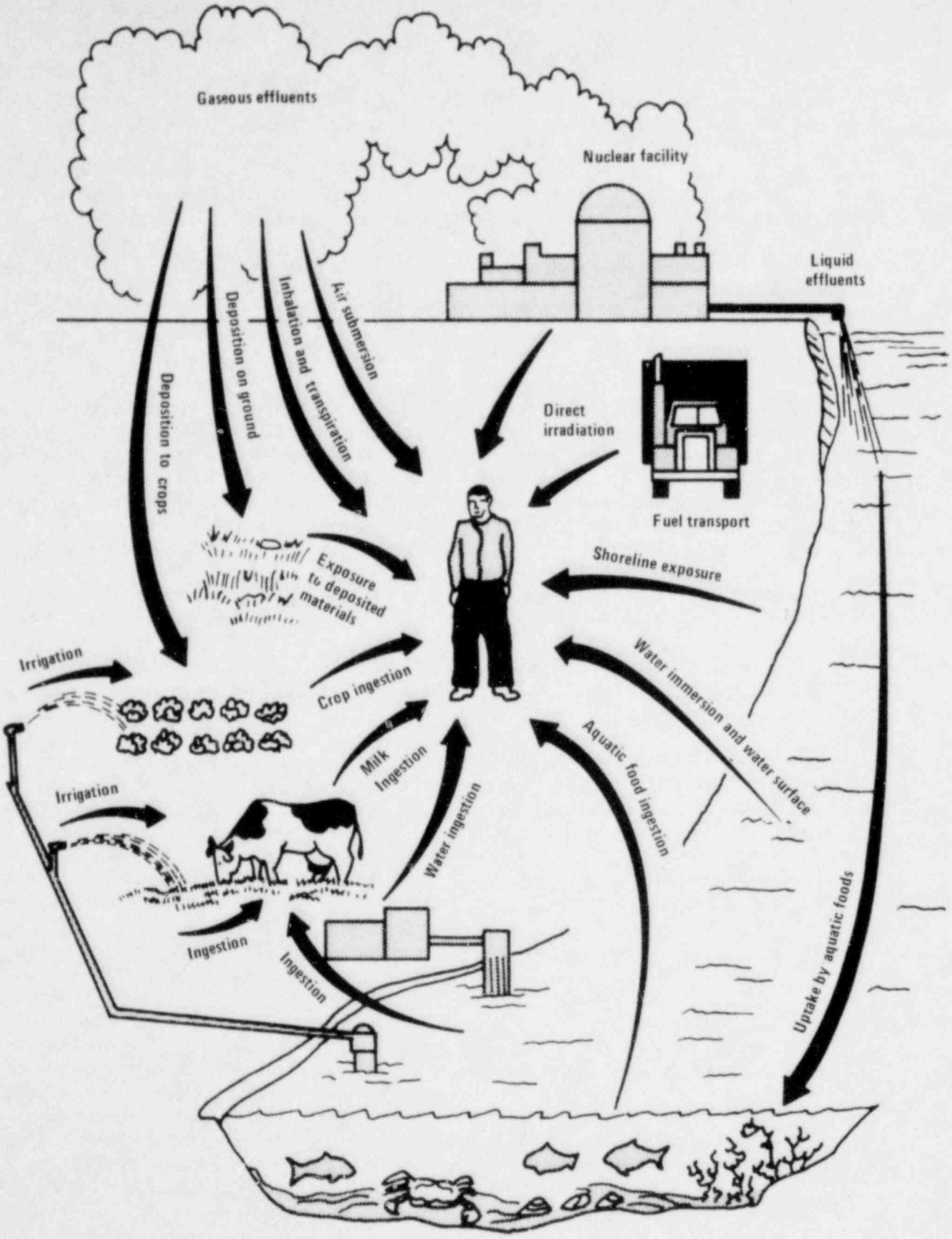


Figure 2-10. Exposure pathways to man.

2.4 LICENSING STATUS AND CONSIDERATIONS

There are currently 72 LWRs with operating licenses in the United States. Construction permits have been issued for an additional 88 units.

The reference system, Combustion Engineering, Inc.(C-E), System 80, and the reference power plant, Perkins nuclear power station, have been reviewed by the NRC staff and by the Advisory Committee on Reactor Safeguards (Refs. 16-19).

Several outstanding items and generic issues have been identified in these reviews. Combustion Engineering, Inc., has initiated a program to resolve the various issues, and satisfactory solution is expected. The Duke Power Company is also in the process of resolving the various issues raised with respect to the Preliminary Safety Analysis Report for the Perkins station.

Some aspects of LWR design, construction, and operation are still under review by the NRC and the Advisory Committee on Reactor Safeguards (ACRS), and are subjects of ongoing NRC and industry safety research projects. These aspects were reviewed by the NRC in "Unresolved Safety Issues" (NUREG-0510, February 1979). The issues are not considered significant enough to require plant shutdown for immediate modification; however, they are subjects for industry and NRC actions to improve LWR safety. The issues pending completed NRC action are discussed below.

1. Water hammer. The concern is that water hammer, which has occurred in many LWR 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. An NRC interim report on the principal aspects of this subject was scheduled for issuance in spring 1979, and completion of the investigation, for 1980.

2. Asymmetric blowdown loads on the reactor coolant system. Blowdown due to events such as a primary coolant piping rupture would impose large, nonuniformly distributed loads on the reactor vessel, reactor vessel internals, and other components in the reactor coolant system. The potential for such asymmetric loads was identified a few years ago and was not considered in the original design of some PWRs. The NRC has requested all operating PWR licensees to assess the adequacy of the reactor vessel supports with respect to these loads. It will review these analyses and will conduct a pipe-break probability study. All these efforts are to be completed in 1979. BWRs are also susceptible to these asymmetric loads, and plans are being made to resolve this issue for BWRs as well.

3. Pressurized-water reactor steam generator tube integrity. The complex, corrosion-related phenomena of tube denting, stress corrosion, tube/support plate interactions, and tube cracking have severely affected several PWR plants. While the NRC staff 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 NRC staff requirements and guidelines. These tasks are scheduled for completion in early 1980.

4. 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 the Mark I and Mark II plants. The NRC staff determined that a detailed reevaluation of these containment system designs was required. Analyses made by industry and accepted by the NRC for the Mark I design have established that the plants could continue to operate safely, pending decisions on longer term approaches. Any needed plant modifications are scheduled for implementation by December 1980.

The analysis of the Mark II plants was completed in October 1978. The NRC staff is evaluating confirmatory experimental and analytical programs to assess the margin for selected loads. The Mark II program leading to conclusions regarding modifications to be made to the plants is scheduled for completion in October 1980.

5. Anticipated transients without scram. This issue concerns (1) the possible failure of the reactor control rod shutdown system to scram after an anticipated transient, and (2) the possibility that anticipated transients without scram (ATWS) may be sufficiently low that it can continue to be excluded from the design basis (as it has been in the past).

   In April 1978, the NRC concluded that the reliability of current reactor scram systems necessary to meet the safety objectives was not demonstrated, and that means of mitigating the consequences of ATWS events should be provided in plant designs. Alternative means of reducing the probability or the consequences of ATWS events are being evaluated by NRC staff, so that recommendations can be provided in 1979.

6. BWR nozzle cracking. The issue is whether or not the cracking that has occurred in the feedwater nozzles of 21 of the 23 BWRs and in the nozzle of the control rod drive 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 cause considerable personnel exposures. The NRC staff issued interim guidance to operating plants in July 1977 and is continuing to review development of design modifications and crack-detection techniques. This effort is scheduled for completion in late 1979.

7. Reactor vessel materials toughness. The results from a reactor vessel surveillance program indicate that up to 20 older operating PWRs were fabricated with materials that will have marginal toughness after comparatively short periods of operation. An NRC task scheduled for completion in July 1979 is to evaluate material degradation mechanisms resulting from neutron irradiation and determine the appropriate licensing criteria and corrective action for the low toughness of reactor vessels in these older plants.

8. Fracture toughness and potential lamellar tearing of PWR steam generator and reactor coolant pump supports. The concern is that, although these supports are designed for worst-case accident conditions, poor fracture toughness could cause the supports to fail during accidents. The NRC generic study of fracture toughness is to be completed in August 1979, and the study of lamellar tearing, in 1981.

9. Systems interactions in nuclear plants. The ACRS requested that the NRC staff evaluate the reactor safety systems from a multi-disciplinary point of view, to identify undesirable interactions among plant systems. The concern is that these interactions may be overlooked because the plant design and analysis frequently are fragmented and assigned to teams of functional engineering specialists without adequate design integration. Sandia Laboratories is evaluating current NRC review procedures for adequacy in this regard, and this work is to be completed in September 1979.

10. 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. There are questions regarding the capability of equipment in older plants to perform under accident conditions and also regarding the adequacy of tests and analyses of equipment in newer plants to qualify their equipment. The NRC 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 NRC and industry are developing guidelines for implementing the applicable standard. This effort is scheduled for completion in 1979. Further efforts will involve review of qualification programs of industry.

11. Residual heat removal shutdown requirements. It is essential that a power plant be able to go from hot-standby to cold-shutdown conditions, under all conditions, when that course of action is deemed best for safety. There is some uncertainty that this can be accomplished in some plants, under all accident conditions, using the plant safety systems--as the recent experience with the Three Mile Island-2 plant has shown. The NRC staff reviewed this issue and made changes to the NRC Standard Review Plan for plants. In addition, guidelines for residual heat removal system design requirements were incorporated into Regulatory Guide 1.139, expected to be issued in final form in late 1979 or early 1980.

12. Control of loads near spent fuel. The concern is that plant operating procedures and designs may not protect adequately against 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 cause offsite releases to exceed the guideline values in 10 CFR 100. NRC is reevaluating current requirements and procedures to develop a revision to the Standard Review Plan, which can then be used to implement changes in currently operating plants and new plants. The task of developing criteria is expected to be completed in 1979, and changes will then be implemented on a plant-specific basis.

13. Seismic design criteria. A number of plants have construction permits and operating licenses that were issued before current NRC regulations and regulatory guidance were in place. To ensure that these plants do not present an undue risk to the public, the NRC is reviewing the seismic design of various plants. It is also reviewing the seismic design sequence to determine their 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. A major part of the effort

is scheduled for completion in September 1979; the remaining portion, related to earthquake ground motion near the earthquake source, is to be completed in 1981.

14. Pipe cracks at BWRs. Pipe cracking has occurred in the heat-affected zones of welds in the primary system piping in BWRs since the mid-1960s. This 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 NRC to reestablish a Pipe Crack Study Group to address the pertinent issues, including 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 PWRs. The Study Group completed its report in January 1979. In addition, the NRC is conducting several generic technical reviews that focus on improving piping inspection techniques and requirements.

15. 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 (ECCS) pumps would recirculate this water to the ECCS and 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, NRC staff are continuing to study the behavior of pipe insulation debris under accident conditions. Regulatory Guide 1.79 addresses the testing of the recirculation function, and the NRC staff believes that pumps tested in accordance with this Guide resolve this issue. Study of the issue is continuing to provide further guidance on generic design and review of emergency sumps.

16. Station blackout. The issue is whether or not nuclear power plants should be designed to accommodate a complete loss of all alternating current (AC) power including offsite sources and onsite emergency diesel generators. Long-term loss of AC power at PWRs, accompanied by loss of auxiliary feedwater pumps, could result in an inability to cool the reactor core. Current NRC requirements demand that diverse power drives be provided for the redundant auxiliary feedwater pumps, normally accomplished by an AC-powered electric motor and a redundant steam turbine-driven pump. There is concern regarding the design adequacy of the plants licensed prior to adoption of the current requirements. An initial survey of operating plants shows that all plants have some capability for accommodating an extended loss of AC power. The NRC is continuing its further review of older plants to determine if any additional requirements are needed, e.g., the time for retaining this core-cooling capability. A schedule for completing study of this issue has not yet been established.

Subsequent to the NRC report to Congress in January 1979, which described the currently unresolved safety issues discussed above, a major incident occurred at the Three Mile Island nuclear power plant site in Pennsylvania. This incident led to major reviews of the licensing criteria, design requirements, operating procedures, and operator qualifications by the NRC, the presidentially-appointed 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 NRC reviews are the Lessons-Learned Task Force by the staff scheduled for completion by November 1979, 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 at the end of October 1979. 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 NRC report (NUREG 0578, TMI-2, Lessons-Learned Task Force Report, July 1979) by the Nuclear Regulatory Commission disclosed a number of areas of design, analysis, and plant operations that the Task Force recommends 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 NRC 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:

1. Making inert all boiling-water reactor containments
2. Providing the capability to install an external recombiner at each light-water reactor facility
3. Revising the limiting conditions for operation based on the total loss of safety-system availability through human or operational errors

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:

1. 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.

2. System design requirements. The adequancy 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 (a) the classification and requirements for non-safety-grade systems and components, (b) operator interactions, and (c) postaccident design requirements.

a. Non-Safety System Classification and Requirements

Non-safety systems and components 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.

b. 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 an adverse operator action which occurred several times during the Three Mile Island accident, and will be reexamined.

c. Postaccident Design Requirements

The ongoing reexamination of design requirements for postaccident operations includes the availability of postaccident monitoring instrumentation, provisions for storage and treatment of large quantities of radioactive liquid and gaseous wastes, and procedures for handling other anticipated postaccident problems on site.

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

2.5 RESEARCH AND DEVELOPMENT

2.5.1 GENERAL

The current research and development programs in the United States have the following four major objectives:

1. Improve the information base for conducting accident analyses and for evaluating the effects and consequences
2. Improve the performance of engineered safety features either by improvements in existing designs or by new and innovative designs
3. Improve plant availability
4. Improve standardization in plant design, construction, and operation.

The major sponsors of research and development in the United States are the NRC, the U.S. Department of Energy, the Electric Power Research Institute, and the reactor industry (primarily Babcock & Wilcox, Combustion Engineering, Inc., the General Electric Company, and Westinghouse Electric Corporation).

The existing reactor safety research programs (Ref. 20), including those of the NRC, may be divided into the following general categories:

1. Thermal-hydraulic and system-interaction tests: Experiments designed to further elucidate the physical phenomena that occur in postulated accidents. These experiments help the model developers confirm and improve techniques for analyzing the safety systems of commercial nuclear power plants.
2. Fuel-behavior tests: Experiments designed to provide a better physical understanding of the behavior of nuclear fuel rods under normal and abnormal conditions. These experiments are also used in the development of analytical models.
3. Primary-system integrity studies: Experimental and analytical efforts designed to improve the understanding of the metallurgical and mechanical response of the primary-system pressure boundary of a reactor during normal and accident conditions.
4. Computer-code development: An analytical program designed to provide better mathematical models and computer codes for calculating the behavior of nuclear power plants under postulated accident conditions.
5. Reactor operational safety research: A research effort on the operational safety aspects of nuclear power reactors.
6. Site safety research: An experimental and analytical effort designed to provide a better understanding of the influence of siting on the safety of nuclear power plants.
7. Risk assessment: Primarily an analytical program in which techniques are developed and used to assess the risk associated with the use of nuclear power.
8. Earthquake-related research: Research on the quantification of the safety margins in current seismic design and the ability of structures and components to withstand earthquake-induced motion and forces. Geologic and seismological studies of regional seismicity in the eastern United States and the Pacific Northwest; and miscellaneous studies of soil properties, etc.

The extent to which the U.S. Government, the Electric Power Research Institute, and industry are involved in research and development is summarized in Reference 20.

The NRC safety research program may be summarized in the following categories (Ref. 20):

I. Projects for immediate pursuit based on evaluation of potential for improving the safety of light-water reactors:

a. Alternative containment concepts
b. Alternative decay heat removal concepts
c. Alternative emergency core cooling concepts
d. Improved in-plant accident responses
e. Advanced seismic designs

II. Topics receiving significant attention in the NRC regulatory process and a confirmatory research program. These are to be reexamined for completeness in scoping studies:

a. Nondestructive examination and on-line monitoring
b. Reduced occupational exposure
c. Improved reactor shutdown systems
d. Protection against sabotage
e. New siting concepts
f. Improved offsite emergency response planning

III. Projects in which scoping studies should be conducted to determine whether research is warranted in the future:

a. Improved plant controls
b. Reactor vessel rupture controls
c. Core retention measures
d. Equipment for reducing radioactive releases
e. Improved plant layout and component protection

2.5.2 REFERENCE SYSTEM

In order to resolve the various issues raised by the NRC on the safety of the reference system, Combustion Engineering, Inc. has undertaken the test programs summarized in Table 2-31.

Table 2-31. Combustion Engineering test programs

	Test	Purpose of test
1.	16 x 16 Fuel assembly design tests	
	Upper guide structure and control-element-assembly buffer test	Verify structural and functional adequacy of the control-element-assembly guide-tube structure buffer design
	Components proof test	Verify scram characteristics, scram time and fuel uplift forces, and proof test the control-element assembly, control-element drive mechanism, guide structure, and fuel assembly
	Spacer-grid test	Verify structural characteristics
	Fuel-assembly static test	Verify lateral load deflection
	Fuel-assembly dynamic test	Verify pluck, pluck impact, vibratory and axial impact effects
	Reactor flow model test	Verify design hydraulic parameters
	Departure from nucleate boiling improvement test	Verify thermal performance capability
	In-core flow mixing test	Verify rate of intersubchannel energy transfer due to turbulent interchange and flow scattering of coolant
2.	Fuel-development programs	
	Densification program	Verify effects of fuel-processing methods and parameters on in-reactor densification at high linear power and burnup
3.	Loss-of-coolant accident refill and blowdown heat-transfer tests	
	Loss-of-coolant accident refill tests	Verify the capability of the emergency core-cooling system to recover the core after a loss-of-coolant accident
	Blowdown heat-transfer test	Verify the Dougall-Rosenow correlation; verify the transient critical heat flux and the post-critical heat flux heat-transfer coefficients
4.	Reflood test	Verify the reflood heat-transfer coefficients
5.	Iodine decontamination test	Verify Combustion Engineering's assumed iodine partition factors as described in CENPD-67
6.	Iodine spiking test	Develop a realistic and conservative model for the iodine spiking phenomenon
7.	Steam-generator program	Verify the analytical models used to predict transient and accident loads on the steam generator
8.	Core protection calculator program	Demonstrate the performance of the proposed core protection calculator system software and hardware

REFERENCES FOR CHAPTER 2

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17. Letter from W. Kerr, Chairman, Advisory Committee on Reactor Safeguards, to W. A. Anders, Chairman, U.S. Nuclear Regulatory Commission, subject: "Combustion Engineering Standard Safety Analysis Report--CESSAR-80," September 17, 1975.

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Chapter 3

ONCE-THROUGH LOW-ENRICHMENT, HIGH-BURNUP URANIUM FUEL (PWR LEU(5)-Mod OT)

3.1 DESCRIPTION

This reactor/fuel cycle combination is a pressurized-water reactor (PWR) using 4.3% low-enriched uranium oxide pellet fuel, modified to achieve 50 MWd/kg average burnup and using other means to decrease uranium requirements, operating on a once-through cycle. Spent fuel will be stored at the reactor site or in an away-from-reactor storage facility. Ultimately, the spent fuel will be sent to a geologic spent-fuel repository. Low-level wastes from fabrication will be sent to a shallow land disposal site.

The fuel cycle facilities associated with this reactor/fuel cycle combination, shown in the mass-flow diagram (Figure 3-1) are discussed in the following sections of Volume VII:

Enrichment	Section 3
Fuel fabrication 1	Section 4.1
Spent fuel storage	Section 6.3
Waste disposal 1	Section 7.1
Waste disposal 3	Section 7.3

For the Nonproliferation Alternative Systems Assessment Program (NASAP) alternatives assessment, Combustion Engineering, Inc. (C-E) performed a study to evaluate the benefits and the potential problems associated with increased fuel burnup in standard PWRs (Ref. 1). Increase in burnup has more potential for significantly reducing uranium requirements than any other alternative identified so far and accounts for most of the overall gains in uranium utilization anticipated for the light-water reactor (LWR). Furthermore, increased burnup could be implemented in the near term, can be readily backfitted into existing LWRs, and can be achieved with a modest research and development program.

Although simplified calculations indicate the potential benefits of high burnup, more detailed evaluations are required to confirm that the margin is adequate and that the anticipated gains in resource utilization can, in fact, be obtained for practical fuel-management schemes. It is also necessary to evaluate high-burnup fuel from the standpoint of fuel-rod design and performance. Ongoing U.S. Department of Energy (DOE) irradiation programs to demonstrate high-burnup fuel must be completed to resolve questions associated with fuel-rod performance.

The generalized reactor performance characteristics are summarized in Table 3-1; the reactor-design data are summarized in Table 3-2. Additional data on fuel management are presented in Section 3.1.4. The reactor that was used to generate the fuel-cycle data discussed in this report is the C-E System-80 PWR; similar performance, however, could be achieved with LWRs of other designs.

The development problems and potential benefits associated with high-burnup fuel cycles in boiling-water reactors (BWRs) are similar to those discussed in this chapter for the reference PWR. The principal difference in fuel management is that BWRs have a higher number of fuel assemblies (848 for a 3,800-MWt reactor) and use a longer fuel-residence time. Current BWR fuel-management schemes involve replacing 20 to

25% of the fuel assemblies at each annual refueling. High-burnup fuel cycles would replace about 15% of the fuel assemblies at each refueling.

3.1.1 FUEL MECHANICAL DESIGN

3.1.1.1 Design Bases

The bases (for the standard 30-MWd/kg burnup case) of the fuel mechanical design are described in Section 4.2.1.1 of Reference 2. However, several topics need to be investigated further in the light of higher burnup. These items are discussed in Section 3.1.1.3.

3.1.1.2 Design Description

Fuel design for the standard 30-MWd/kg burnup case is described in Section 4.2.1.2 of Reference 2. Table 2-3 contains a summary of selected Perkins NSSS core mechanical design parameters. These parameters and the design itself are subject to further investigation in view of the increased burnup as well as of the modified fuel-management scheme. The following must be reevaluated:

1. Fuel rod design
   (a) The required initial internal pressure and acceptable end-of-life pressure
   (b) Pellet-cladding interaction and the consequent effect on the design of the fuel-to-cladding gap
2. Burnable neutron-absorber rod
3. Fuel assembly
4. Mechanical, chemical, metallurgical, and thermal properties of the fuel-rod components, control-rod components, and fuel-assembly components
5. Enrichment: In order to increase the burnup from 30,000 to 50,000 MWd/MT, it is necessary to raise the feed enrichment by more than 1 wt% to 4.3%

In order to accommodate the fission-gas release at high burnup without exceeding the design limit for the internal fuel-rod pressure, it appears that a design change will be required. Several design modifications, such as increasing the gas-plenum volume or using annular fuel pellets, appear feasible. Choosing the parameters for a new fuel-rod design will have to wait until fission-gas-release data for higher burnup is available. Moreover, the ongoing DOE irradiation programs to demonstrate high burnup fuel will have to be completed before many of the questions associated with fuel-rod performance can be resolved.

3.1.1.3 Design Evaluation

There has been considerable experience with current fuel and core materials in LWRs. Well-established evaluation standards and regulatory requirements exist in the case of the reference PWR. Past LWR designs have used conservative quality control and design bases (see Section 4.2.1.3 of Reference 2).

If fuel is to be discharged after substantially higher burnup, it will be necessary to reestablish the adequacy of materials for operation. Some of the key factors to be considered are increased internal pressure; fuel-assembly dimensional changes; increased corrosion, hydriding, and fretting; increased fission-product release and consequent change in physical properties; and increased pellet-cladding interaction.

a. Fission-Product Release and Rod Internal Pressures

All modern PWR fuel rods are internally pressurized with helium to provide better gap conductance and resistance to cladding creepdown caused by the primary-system pressure. The fission gas released as the fuel is irradiated increases the internal pressure of the rod; at high burnups, the end-of-life rod pressures may exceed design limits.

Of concern is the possibility that, as a result of excessive fission-gas release, fuel-rod internal pressures will increase to such an extent that the cladding will expand away from the fuel into the coolant channel, impeding the flow of coolant and degrading the cladding-to-fuel gap conductance. This liftoff can be avoided by applying the conservative criterion that the internal rod pressure should not exceed the primary-system pressure (~2,200 psia) since this would clearly preclude liftoff. Somewhat higher pressures would be allowed when the strength of the cladding material and its ability to resist outward expansion are considered; with this "no liftoff" criterion, internal rod pressures of between 2,800 and 3,000 psia may be acceptable for a PWR. Predictions of fuel-rod internal pressures are typically performed by analytical models containing either a semiempirical respresentation of fission-gas release or analytical models benchmarked against experimental information. Unfortunately, only limited information is available on fission-gas release for high-burnup fuel.

b. Fuel-Assembly Dimensional Changes

Irradiation can cause both stress-free and stress-induced permanent dimensional changes in the components of the fuel assembly. These include axial growth of the fuel rods, poison rods, and control-rod guide tubes; fuel- and poison-rod bowing; fuel-assembly bowing; and retention grid relaxation. Although engineering evaluations and tests are needed to determine whether current fuel assemblies have adequate margin at high burnups and whether design changes are required, it appears likely that the effects of fuel-assembly dimension changes can be accommodated by providing adequate allowances for these dimensional changes in fuel-rod and assembly design.

c. External Cladding Corrosion

Under normal steady-state operation, current PWRs have not experienced excessive corrosion on the outside surface of fuel rods. The heat fluxes and residence times have been accommodated without serious consequences. If exposures and residence times are increased significantly, it is possible that a thicker layer of oxide and crud may develop. The increased insulation may raise the oxide-cladding interface temperature significantly to further accelerate the corrosion of the Zircaloy cladding. Because of this potential for accelerated Zircaloy corrosion, it will be necessary to develop further the data base for in-reactor corrosion of Zircaloy at typical power-reactor conditions for in-reactor residency times anticipated for extended-discharge-burnup fuel. However, presently available data, primarily from the extended exposure of Zircaloy tubing at the Shippingport PWR and from the Saxton plutonium experiments, suggest that Zircaloy corrosion may prove acceptable for the 5-year in-reactor residency times currently being considered for high-burnup fuel.

d. Change in Physical Properties

Increased accumulation of fission products and longer residence time would cause changes in the physical properties such as the cladding yield strength and the heat-transfer properties of fuel, gap, and cladding. Evaluation of the fuel-assembly performance would therefore require a data base on physical properties at higher burnups.

e. Pellet-Cladding Interaction

During irradiation, cladding creepdown and fuel swelling eventually lead to pellet-cladding contact. Once contact has occurred, the cladding is more susceptible to failure, because of pellet-cladding interactions (PCI), a mechanism that is possibly assisted by stress-corrosion cracking. Fuel becomes more susceptible to PCI failures on burnup increases because of the following:

1. Higher cladding stresses that result from the closure of the as-fabricated gap between fuel and cladding by clad creepdown and by pellet swelling and relocation
2. The increased availability of the fission products iodine and cesium, which are expected to be the primary corrosive species contributing to stress-corrosion cracking

The pellet-cladding interaction mechanism has been identified as a significant cause of fuel irradiated to conventional burnups (<30,000 MWd/MT). Thus, it is clear that pellet-cladding interactions will also be a problem when extended burnups are employed. Restrictions are currently placed on the operation of reactors to avoid failures from pellet-cladding interactions and the DOE and industry programs aimed at developing fuel immune to PCI-type failures are under way. Of concern is the question of whether the use of high-burnup fuel will require additional research and development programs, beyond those for conventional-burnup fuel, or further restrictions on plant operation.

Power-ramp tests for high-burnup fuel will have to be performed to establish whether or not the conditions within the fuel rod that can lead to PCI failures during a power transient become more adverse for high-burnup fuels. There are a number of reasons for believing that the propensity for PCI-type failures (upon a given power rate challenge) will not be significantly higher at extended burnups than for fuel irradiated to the burnup range of 20,000 to 30,000 MWd/MT:

1. The peak cladding stresses are not expected to be burnup dependent once cladding creepdown and pellet swelling and relocation have resulted in the onset of firm contact; the onset of such firm contact typically occurs after about two cycles of irradiation.
2. Corrosive fission products significant enough to result in stress-corrosion cracking failure at a critical cladding stress level are likely to be available after two cycles of irradiation.

Even if, as postulated above, the local conditions within a fuel rod that leave it susceptible to PCI-type failure upon a power-ramp challenge are no more adverse for high-burnup fuel, the probability for PCI-type failure will be larger for extended-burnup fuel if such fuel undergoes a larger number of power changes of sufficient magnitude to cause PCI-type failures during irradiation. For extended-burnup fuel, the length of time that the fuel pellet and the cladding are in contact increases considerably (from about 1 year for current designs to about 3 years for high-burnup designs). This increase in the pellet-cladding contact time, along with the increase in the number of power changes due to refueling and/or power maneuvers, will subject the fuel to a larger number of transients, which may increase the number of fuel pins experiencing PCI failure.

In summary, it appears that the tendency of fuel to fail by the PCI mechanism during a power ramp will not increase with increased burnup, but the probability of

fuel failure from pellet-cladding interactions may be greater because of the longer period of time during which the fuel is susceptible to PCI-type failures. Power ramp tests for high-burnup fuel will have to be performed in order to establish the acceptable power envelopes (i.e., the acceptable combinations of local burnup, power, power change, and rate of power change) for high-burnup fuel. Under the assumption that existing programs will develop fuel less susceptible to PCI-induced failure, it is thought that PCI failure should not preclude the operation of fuel to higher discharge burnups.

3.1.1.4 Testing and Inspection Plan

As indicated in Section 3.1.1.1, some modification to the design of either the fuel or the fuel rod will be required in order to achieve the higher batch-average exposure. Consequently, some changes in quality-assurance requirements must be worked out to accommodate the design changes (such as annular pellets and/or increased plenum size) before the commercial introduction of high-burnup fuel. The effort required, however, is not expected to be significant since detailed testing and inspection plans and procedures exist for the reference PWR design (see Section 4.2.1.4 in Reference 2).

3.1.2 FUEL NUCLEAR DESIGN

3.1.2.1 Design Bases

The design bases established for the reference System 80 (see Section 4.3.1 of Reference 2) would have to be reestablished for the high-burnup case because of the increased feed enrichment, higher discharge burnup, and the consequent change in fuel management and in fuel-rod design. Of concern are the following items:

1. Excess reactivity and fuel burnup
2. Core design lifetime and fuel-replacement program
3. Negative reactivity feedback and reactivity coefficient
4. Burnable poison requirements
5. Stability criteria
6. Maximum controlled-reactivity-insertion rates
7. Power-distribution control
8. Shutdown margins and stuck-rod criteria
9. Chemical-shim control
10. Maximum control-element-assembly speeds
11. Anticipated transients with failure to scram

3.1.2.2 Description

The nuclear characteristics of the core have to be reevaluated after specific design changes are established for the high-burnup case. Power distributions have to be computed to obtain the peak linear heat rate and the minimum departure-from-nucleate-boiling ratio by means of which realistic design limits can be specified. Also to be evaluated are

1. Reactivity coefficients
2. Control-element-assembly patterns and reactivity worths
3. Control requirements
4. Control and monitoring of power distribution
5. Criticality of fuel assemblies
6. Xenon stability
7. Vessel irradiation

3.1.2.3 Analytical Methods

Since the design changes envisioned are not expected to introduce drastic changes in the nuclear characteristics of the core, the analytic methods available to treat the reference case (see Section 4.3.3 of Reference 2) would be adequate for treating the high-burnup case.

3.1.3 FUEL THERMAL-HYDRAULIC DESIGN

3.1.3.1 Design Bases

As pointed out in Section 3.1.1.1, some design modification to the design of either the fuel or the fuel rod will be required in order to achieve the postulated 50,000-MWd/MT batch-average burnup. One of the key factors that influence the design change is the thermal margin adequacy. The intent is to adhere to the design bases of the reference system given in Section 4.4.1 of Reference 2 so that an increase in burnup does not pose any serious licensing or operating problems.

The changes in design and operation envisioned (see Section 3.1.1.1) are as follows:

1. Increase in the fuel-rod length and/or shortening of the active fuel column
2. Use of annular pellets
3. Use of duplex pellets
4. Increased residence time and discharge burnup
5. Increase from three-batch to five-batch fuel management

Increase in the fuel-rod length (to provide added plenum space) would have an insignificant effect on the thermal-hydraulic characteristics. However, shortening the active fuel column to provide plenum space in the current length would necessitate an increase in the average (and therefore the peak) linear heat rate. This would result in a higher peak temperature in the fuel and in the cladding. The margins for such events as the loss-of-coolant accident would therefore be reduced.

The use of annular pellets would lower the peak temperature in the fuel and also the fuel loading in the core. However, in order to maintain a constant number of megawatt-days, the discharge burnup would have to be increased even further. The central-region temperature of the duplex pellets is somewhat lower than that of solid pellets. Increase in residence time and discharge burnup affect the thermal-hydraulic characteristics of the fuel and the core. For example, an increase in fission-product buildup changes the thermal conductivity of the fuel and of the fuel-to-cladding gap. Confirmatory data are required.

Another effect to be considered is increased crud buildup on the cladding surface or corrosion which could alter the heat-transfer coefficient. However, presently available data, primarily from the extended exposure of Zircaloy tubing at the Shippingport PWR and from the Saxton plutonium experiments, suggest that Zircaloy corrosion may prove acceptable for the 5-year in-reactor residence times currently being considered for high-burnup fuel.

Fuel-assembly dimensional changes resulting from fuel swelling or rod bowing would alter the heat flow and temperature distributions. The effects of the dimensional changes are not expected to be serious, and it appears that adequate allowances in the design of fuel rod and assembly can be provided.

3.1.3.2 Description

A description of the thermal-hydraulic design and the design parameters cannot be provided until a specific design of the fuel-rod and core is decided on. Fuel and cladding temperature distributions (and peak temperatures) as well as the departure-from-nucleate-boiling ratio for a number of high-power rods have to be ascertained for steady-state conditions and for transients. Design values for hot-channel factors (see Section 4.4.2.3.3 of Reference 2) would also have to be determined for the modified design.

Core pressure drops and hydraulic loads are not expected to change significantly.

If fuel is to be discharged after higher burnup and longer residence time, it will be necessary to evaluate further uncertainties in the estimates of the following:

1. Departure-from-nucleate-boiling ratio
2. Pressure drop
3. Fuel and cladding temperatures
4. Enthalpy rise factor

3.1.3.3 Evaluation

The method of evaluating thermal-hydraulic design is not expected to change with an increase in discharge burnup. Evaluation procedure and results for the reference core are available in Section 4.4.3 of Reference 2. No significant change in the various computer codes is expected, although some physical property and correlation parameters may have to be modified. Flow-model tests may have to be performed for the modified core to obtain or verify hydraulic parameters, although these parameters are not expected to change significantly.

3.1.3.4 Testing and Verification

Current and planned testing and verification programs, as described in Section 4.4.4 of Reference 2, can be augmented readily to obtain information with respect to the high-burnup case. Specifically, further information regarding departure from nucleate boiling would be necessary in view of a possible increase in linear heat rate changes in the heat-transfer characteristics of the modified design. The component test programs should also be modified to reflect any changes in the fuel and control-element assemblies.

3.1.4 FUEL MANAGEMENT

At present, operating PWRs typically employ an annual refueling schedule in which approximately one-third of the fuel is replaced each year (three-batch fuel management) to obtain discharge fuel exposures usually varying between 30,000 and 33,000 MWd/MT. Most PWRs in the past have employed a fuel-management scheme that locates fresh fuel around the core periphery during the fuel's first cycle of irradiation. After the completion of the first cycle of irradiation, this fuel is moved in toward the core interior, where it resides for the second and third cycles of irradiation. However, PWRs are now beginning to use various forms of "low-leakage" fuel management in which all or part of the fresh fuel assemblies are initially loaded into the core interior. After the first cycle of irradiation, these fuel assemblies are moved to peripheral core locations and are then returned to the core interior for their last irradiation cycle. Placing the partially irradiated fuel assemblies in the peripheral locations (rather than

placing fresh fuel assemblies in these locations) reduces neutron leakage at the core periphery, resulting in a 2 to 4% reduction in yellowcake requirements.

The primary differences between the improved and standard PWR fueling practices lies in the higher discharge fuel exposure and smaller fraction of the core replaced at each refueling interval. In contrast to present PWR fueling practices, typically employing discharge fuel exposures between 30,000 and 33,000 MWd/MT, the discharge fuel exposure has been extended to 50,650 MWd/MT in the improved PWR design. This increase in discharge exposure allows the fraction of the core refueled to be reduced from 33 to 20% (i.e., a five-batch fuel management is employed in the improved design, rather than the current three-batch fuel management) while maintaining an annual refueling interval. Both the increased discharge exposure and the use of a larger number of fueling batches contribute to the improved resource utilization; about one-third of the reduction in uranium requirements can be attributed to the higher discharge burnup itself, with the remaining two-thirds attributable to the increase in number of refueling batches from three to five.

If conventional fuel management were to be employed (i.e., with fresh fuel located at the core periphery) an 11% reduction in uranium requirements would be realized, as compared to a similar conventional fuel-management scheme in a standard PWR employing three-batch fuel management and a discharge exposure of 30,400 MWd/MT. This improvement in uranium utilization is somewhat lower than the theoretically possible value, partly because of the increased neutron leakage. The neutron leakage is increased because a higher fresh-fuel enrichment is required to achieve the 50,600 MWd/MT discharge exposure (4.3 vs. 3.0 wt% for the standard case) and as a result the power density in the fresh-fuel assemblies is somewhat higher, thus contributing to higher neutron leakage when fresh fuel is located along the core periphery. In order to avoid this increased neutron leakage, it is anticipated that low-leakage fuel-management schemes will be employed in conjunction with extended discharge exposure. The reduction in neutron leakage that results when partially irradiated fuel is placed in peripheral locations, rather than fresh fuel, is estimated to result in a 4 to 6% reduction in uranium requirements, in addition to the previously mentioned 11% reduction from extending the discharge burnup and employing five-batch fuel management.

The use of a low-leakage fuel-management configuration also eliminates several minor problems associated with the higher power density of the peripheral fuel assemblies in the conventional fuel-management configuration, such as the resulting increase in neutron fluence at the vessel. When a low-leakage fuel management scheme is employed in conjunction with the extended discharge exposure, peripheral-fuel-assembly power densities are lower than those in present conventional three-batch fuel-management schemes, and hence the fluence at the reactor vesssel is reduced somewhat. However, even for conventional extended-burnup fuel-management schemes the increase in fluence at the reactor vessel is small, and is estimated to increase the nil-ductility transition temperature at end of life by less than 10°F. (For present-day fuel-management schemes, the nil-ductility transition temperature for the reference reactor increases from 40 to 178°F after 32 effective-full-power years; for the extended-burnup case with conventional fuel management, the end-of-life nil-ductility transition temperature would increase to an estimated 188°F, still well below the 200°F design criterion). Thus, the effect of the slight increase in neutron fluence at the reactor vessel that might occur if conventional fuel management is employed in conjunction with higher discharge exposures is not expected to significantly affect the brittle-fracture characteristics of the reactor-vessel material.

The batch-wise burnup results for the first four cycles are presented in Table 3-3 along with estimates of the transition burnup to the equilibrium cycle. These results show a transition to an equilibrium cycle length of 10,130 MWd/MT (approximately yearly cycles) for a five-batch fuel management with an equilibrium feed enrichment of 4.3 wt% U-235. The batch-wise discharge burnups approach near their equilibrium-cycle value of 50,600 MWd/MT after about seven or eight irradiation cycles.

In order to utilize five-batch fuel management in current operating reactors, it is necessary to undergo a series of transition cycles that allow the feed enrichments to increase from about 3 to 4.3 wt% and the batch fraction to decrease from one-third to one-fifth. This transition must be performed gradually in order to keep the radial power distribution within acceptable bounds. Fuel-management information is summarized in Table 3-4. The isotopic distributions of the fuel inventory at the beginning and at the end of the equilibrium cycle are listed in Tables 3-5 and 3-6, respectively. Reactor charge data for a 30-year lifetime are given in Table 3-7, and the discharge data are presented in Table 3-8. The material flow diagram is shown in Figure 3-1. The numerical identifiers in the fuel cycle steps are correlated with the fuel cycle descriptions of Volume VII.

Table 3-1. Reactor system design and performance data
(PWR once-through optimized discharge exposure 50,650 MWd/MT)

Reactor type	PWR
Reactor thermal power output (gross), MW	3,817
Electrical power output, MW[a]	
Gross	1,344
Net	1,270
Plant heat rate, Btu/kW-hr	10,212
Core design and performance parameters	
Core heat output, MW	3,800
Core volume, liters	40,050
Core loading, kg[b]	
Heavy metal	99,490
Fissile fuel	2,514
Conversion ratio[c]	0.56
Average discharge exposure, MWd/MTHM[d]	50,650
Peak discharge exposure, MWd/MTHM[d,e]	65,000
Fuel type	Oxide
Reactor inlet temperature, ^{o}F	565
Reactor outlet temperature, ^{o}F	621
End-of-cycle excess reactivity	0

[a]Depends on architect-engineer; these values assume mechanical-draft cooling.
[b]Initial uranium dioxide core.
[c]Integrated conversion equilibrium cycle.
[d]Heavy-metal charged.
[e]Rod average.

Table 3-2. Reactor design data specifications
(PWR once-through optimized discharge exposure 50,650 MWd/MT)

Geometric information	
Core height, cm	381.0
Number of core enrichment zones (nominal)[a]	5
Number of assemblies	241
Equivalent diameters, cm	365.8
Pins per assembly	236
Pin pitch-to-diameter ratio	1.325
Overall assembly length, cm	406.4
Lattice pitch, cm	1.288
Assembly material	Oxide fuel with Zircaloy-4 cladding
Cladding parameters	
Cladding outside diameter, mils	382.7
Cladding wall thickness, mils	25
Cladding material	Zircaloy-4
Fissile inventory at beginning of equilibrium cycle, kg	3,024
External fissile inventory, kg	NA
Fissile loss, kg/cycle	880
Specific power, kW/kg fissile[b]	1,256
Power density, kW/kg HM	38.3

[a]Five batches.
[b]Beginning of equilibrium cycle.

Table 3-3. Burnups accumulated by each batch during each cycle for a five-batch fuel-management scheme starting in cycle 1 to equilibrium[a]

Batch	Number of assemblies loaded	Number of fuel pins loaded	Enrichment (wt%)	Cycle 1	2	3	4	5	6	7	8	9	10	11	12	Batch discharge burnup (MWd/MT)
A	48	11,328	1.70	14,971												14,971
B	48	11,328	2.10	16,007	5,802											21,809
C	28	6,160	2.60	18,003	6,939	6,938										31,880
C*	20	4,400	2.60	15,457	6,720	7,246										29,423
C*	4	880	2.60	13,827	6,599	7,878	7,674									35,978
D	29	6,254	2.90	18,520	7,093	7,236	7,674									40,523
D*	16	3,456	2.90	16,561	7,517	7,707	7,674									39,489
E	40	9,440	3.40	10,501	8,140	8,417	9,308	8,426								44,792
E*	8	1,760	3.40	10,543	7,033	8,566	9,308	8,426								43,876
F	48	11,328	3.95		6,548	9,854	10,163	10,070	8,597							45,232
G	48	11,328	3.95			8,367	11,398	10,994	10,274	7,938						48,771
H	48	11,328	4.30				8,833	12,330	11,218	9,487	8,467					50,335
I	49	11,564	4.30					9,556	12,581	10,359	10,119	8,222				50,837
J	48	11,328	4.30						9,750	11,617	11,049	9,826	8,410			50,662
K	48	11,328	4.30							9,003	12,391	10,729	10,051	8,309		50,483
L	48	11,328	4.30								9,603	12,032	10,974	9,930	8,390	50,843
M	48	11,328	4.30									9,325	12,307	10,842	9,930	
N	49	11,564	4.30										9,538	12,160	10,842	
O	48	11,328	4.30											9,424	12,160	
Beginning-of-cycle exposure (MWd/MT)				0	12,057	14,431	16,538	18,160	19,581	21,064	20,995	21,294	21,065			
Cycle burnup (MWd/MT)				14,971	6,865	8,530	9,498	10,275	10,484	9,681	10,326	10,027	10,256			
End-of-cycle exposure (MWd/MT)				14,971	18,922	22,961	26,036	28,436	30,065	30,746	31,321	31,321	31,321			
Core-average enrichment				2.53	2.99	3.37	3.71	3.98	4.16	4.23	4.30	4.30	4.30			

[a]Burnup for cycles 1 through 4 is calculated; transition results for cycles 5 through 11 are estimated.

Table 3-4. Fuel-management information
(PWR once-through optimized discharge exposure 50,650 MWd/MT)

Average capacity factor, %	75
Approximate fraction of core replaced annually	One-fifth
Lag time assumed between fuel discharge and recycle reload, years	2
Fissile-material reprocessing loss fraction, %	1
Fissile-material fabrication loss fraction, %	1
Yellowcake and requirement, ST/GWe	
Initial core	408
Annual equilibrium reload	182.4
30-year cumulative requirement	5,196
Separative-work requirement, 10^3 SWU/GWe	
Initial core	222
Equilibrium reload	118
30-year cumulative requirement	3,488
Requirements for special fuel materials (fissile, plutonium, uranium-233, etc.), kg HM/GWe	
Initial load	0
Annual equilibrium charge, discharge	0
30-year cumulative requirement	0
Other data for proliferation-resistance assessment	
Fuel-element weight, kg	650
Fresh-fuel radiation level, air, mrem/hr	20
Discharge-fuel radiation level, air, R/hr at 90 days	77,000
Discharge-fuel radiation level, water, R/hr at 90 days	220
Discharge-fuel energy-generation rate after 90-day cooling, watts per element	17,000

Table 3-5. Fuel inventory at the beginning-of-equilibrium cycle (PWR once-through optimized discharge exposure 50,650 MWd/MT)

Isotope	Fuel inventory (kg)				
	Fresh fuel, Zone 1	First-burn fuel, Zone 2	Two-burn fuel, Zone 3	Three-burn fuel, Zone 4	Four-burn fuel, Zone 5
Th-232					
Pa-233					
U-232					
U-233					
U-234					
U-235	853.0	631.6	493.5	321.2	215.1
U-236		42.0	66.9	95.6	110.5
U-238	19,701.7	19,576.6	19,471.5	19,291.0	19,129.0
Pu-238		0.14	0.59	2.20	4.47
Pu-239		71.8	98.9	117.5	121.2
Pu-240		9.5	20.7	37.8	49.4
Pu-241		3.9	11.5	23.9	31.6
Pu-242		0.24	1.41	5.72	11.23
Fission products		215.9	382.4	643.3	855.5
Other isotopes					
Am-241		0.04	0.19	0.59	0.89
Cm-242		-	-	-	-
Np-237		1.86	4.30	8.99	13.02

Table 3-6. Fuel inventory at the end-of-equilibrium cycle (PWR once-through optimized discharge exposure 50,650 MWd/MT)

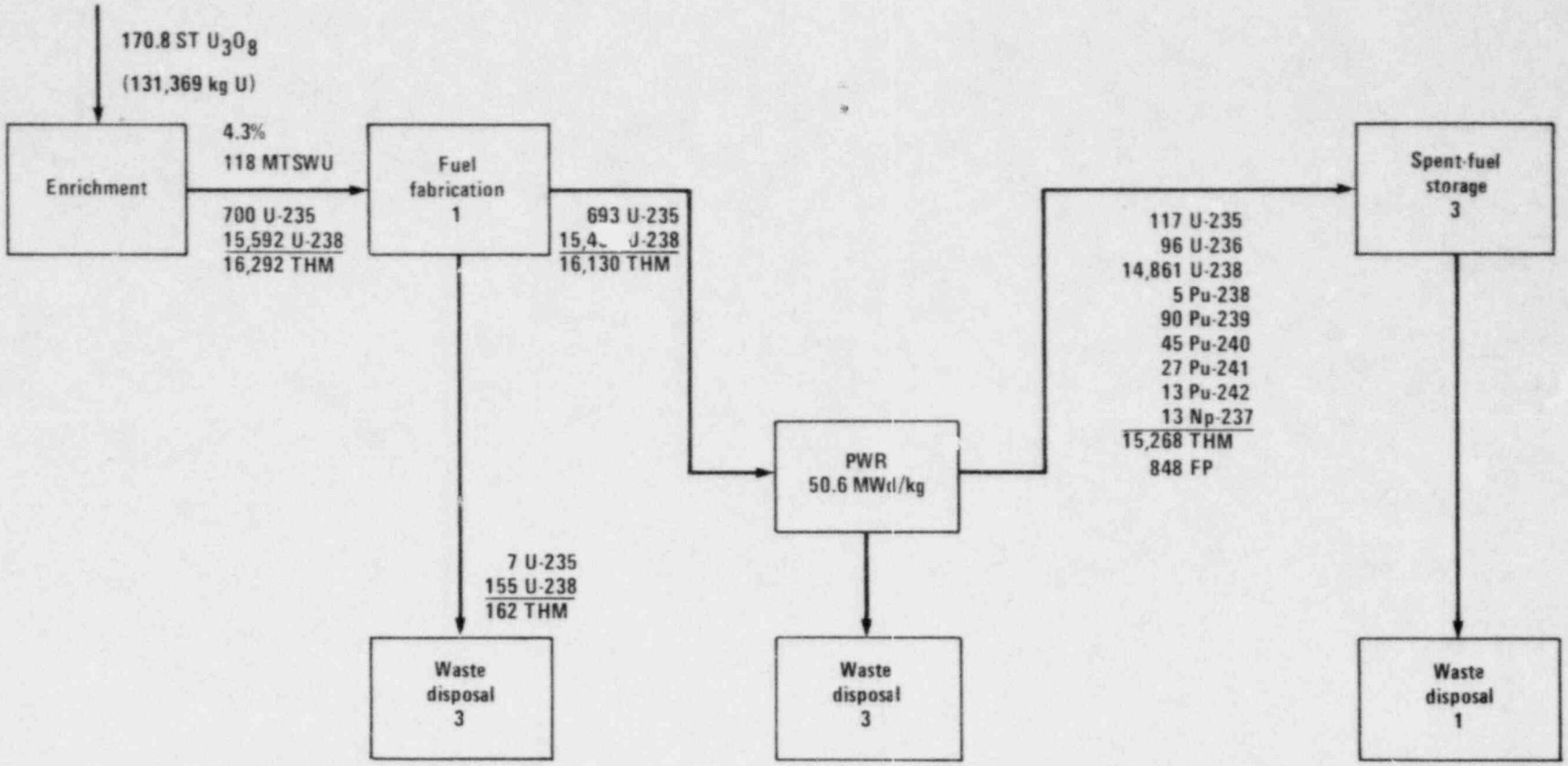
Isotope	Fuel inventory (kg)				
	First-burn fresh fuel, Zone 1	Second-burn fuel, Zone 2	Third-burn fuel, Zone 3	Fourth-burn fuel, Zone 4	Fifth-burn fuel, Zone 5
Th-232					
Pa-233					
U-232					
U-233					
U-234					
U-235	631.6	493.5	321.2	215.1	136.6
U-236	42.0	66.9	95.6	110.5	118.9
U-238	19,576.6	19,471.5	19,291.0	19,129.3	18,953.2
Pu-238	0.14	0.59	2.20	4.47	7.5
Pu-239	71.8	98.9	117.5	121.2	119.9
Pu-240	9.5	20.7	37.8	49.4	58.1
Pu-241	3.9	11.5	23.9	31.6	36.6
Pu-242	0.24	1.41	5.72	11.23	18.01
Fission products	215.9	382.4	643.2	855.5	1,095.3
Other isotopes					
Am-241	0.04	0.19	0.59	0.89	1.07
Cm-242	-	-	-	-	-
Np-237	1.86	4.30	8.99	13.02	16.80

Table 3-7. Reactor charge data for zones 1, 2, 3, etc. (PWR once-through optimized discharge exposure 50,650 MWd/MT)

Isotope	Charge (kg)	Charge per GWe (kg)
Th-232	0	
Pa-233	0	
U-232	0	
U-233	0	
U-234	0	
U-235	853.0	
U-236	0	
U-238	19,706.7	
Pu-238	0	
Pu-239	0	
Pu-240	0	
Pu-241	0	
Pu-242	0	
Np-237	0	
Total heavy metal	20,554.7	16,184.8
Fission products	(0)	(0)

Table 3-8. Discharge data for zones 1, 2, 3, etc.
(PWR once-through optimized discharge exposure
50,650 MWd/MT)

Isotope	Charge (kg)	Charge per GWe (kg)
Th-232	0	0
Pa-233	0	0
U-232	0	0
U-233	0	0
U-234	0	0
U-235	136.6	107.6
U-236	118.9	93.6
U-238	18,953.2	14,923.8
Pu-238	7.5	5.9
Pu-239	119.9	97.4
Pu-240	58.1	43.8
Pu-241	36.6	28.8
Pu-242	18.0	14.2
Np-237	16.8	13.2
Total heavy metal	19,465.5	15,327.2
Fission products	1,095.0	862.2



Notes:

1. Mass flows Q<P are in kilograms per 0.75 GWe-yr.
2. Abbreviations: FP, fission products; MTSWU, metric tons of separative-work units; ST, short ton; THM, total heavy metal.

Figure 3-1. Mass flows for the high-burnup once-through fuel cycle PWR LEU(5)-Mod OT.

3.2 SAFETY CONSIDERATIONS

The evaluation of selected safety parameters indicates that a reactor design with five-batch fuel management and with high burnup has a high probability of being licensable. Although no fundamental problems are expected, the safety analysis of a five-batch fuel-management design must be updated to insure that the results are within those of a three-batch design or within NRC acceptance criteria.

Preliminary results indicate that the following reactor characteristics may be more limiting than those of a three-batch reactor design:

1. The shutdown worth may be smaller.
2. Delayed-neutron fraction may be smaller.
3. The reactor may be less stable with respect to azimuthal xenon oscillations.
4. Reactivity insertion during a steam-line-break accident may be larger.
5. Offsite dose following a reactor accident may be larger.

Assurance is needed that there is adequate margin for each of these items. A more detailed discussion is presented in the sections that follow.

3.2.1 SHUTDOWN WORTH

Calculations of the shutdown worth at the end of an equilibrium fuel-management cycle indicate that the total shutdown worth (all control rods inserted) in five-batch fuel management is somewhat lower (about 10% lower) than that of a three-batch fuel management. This reduction in the worth of the control rods appears to result from a neutron-spectrum change due to the higher feed enrichments and the higher core-average exposures. The resulting higher ratio of fast flux to thermal flux reduces the effectiveness of the control rods, which are primarily thermal-neutron absorbers. The net rod worth, which considers the possibility that the most reactive rod would stick in the out-of-core position appears also to be smaller for five-batch fuel management.

The result of the lower control-rod worth is a tendency for reduced reactivity margins for certain accidents (in particular for the steam-line-break accident). The reduced margin may lead to some difficulty in licensing, depending on the amount of rod-worth margin that is available for a particular reactor design. It is noted that reactors such as the C-E System 80 are designed with extra rod worth to cover the possibility of recycling plutonium. In the absence of plutonium recycle, the extra rod worth is available to cover the expected reductions in rod worth resulting from the five-batch high-burnup design.

3.2.2 EJECTED-ROD WORTH

Calculations that consider the ejection of a rod at full power from one of the regulating control banks and the ejection of a rod at zero power from an initial condition with all regulating banks inserted indicate that the ejected-rod worth for a five-batch fuel-management scheme is no greater than that for a three-batch scheme. These calculations assume that the selection of control rods associated with the regulating control banks is the same for both a five-batch and a three-batch fuel-management pattern.

The specific requirements of the five-batch fuel-management design require the selection of different groups of regulating rods. However, since the ejected-rod worths used in the safety analysis are usually much larger than the calculated

values, any ejected-rod worth differences are not expected to affect the safety analysis results.

3.2.3 KINETICS PARAMETERS

Variations in the kinetics parameters due to the use of five-batch fuel management and high-burnup are expected to be small.

The delayed-neutron fraction at the beginning and end of an equilibrium cycle is expected to be somewhat smaller than the corresponding values for a three-batch fuel-management scheme because of higher core-average burnup.

Typical values of the delayed-neutron-fraction range from 0.0074 at the beginning of life of an all-uranium dioxide reactor compared to a value of 0.0065 at beginning of life for a reactor that uses recycled plutonium (Ref. 1). At the beginning of an equilibrium cycle, the delayed neutron fraction decreases to 0.0054 for the all-uranium dioxide reactor and to 0.0051 for a reactor that uses recycled plutonium (Ref. 1). The delayed-neutron fraction expected for a five-batch high-burnup design is expected to be between the values resulting from an all-uranium dioxide design and the values resulting from a plutonium recycle design. Therefore, the slightly smaller delayed-neutron fractions expected for five-batch fuel management are not expected to cause problems during the safety evaluation of this reactor design.

Variations in other kinetics parameters, such as precursor decay constants λ_i and the prompt-neutron lifetime, do not have an appreciable impact on safety analysis results, and hence the small variations that may result from a five-batch design are not expected to cause significant concern.

3.2.4 MODERATOR TEMPERATURE COEFFICIENT

Moderator temperature coefficients (MTC) in five-batch fuel management are expected to be somewhat more negative than those in three-batch fuel-management. Preliminary results indicate that the difference in the MTC may be on the order of $0.4 \times 10^{-4} \Delta\rho/^{\circ}F$ at end-of-cycle primarily because of the increased blackness of the core, which results from higher enrichments and larger fission-product inventories during an equilibrium cycle. Calculations indicate that at the beginning of the cycle the MTC difference may be larger (about 0.8×10^{-4} more negative), although this result is still preliminary.

This more negative MTC causes less limiting accident results for cores in which the negative power coefficient limits the severity of power excursions such as rod-ejection or rod-withdrawal incidents. In these cases, the more negative MTC results in a more negative power coefficient, which results in smaller power excursions.

For cold-water incidents, such as the steam-line-break incident or the idle-loop-startup incident, the more negative MTC increases the reactivity inserted by the cold water and results in a more limiting accident. For the steam-line-break accident, the results are further aggravated by the reduced shutdown worth discussed above.

The acceptability of the more negative moderator temperature coefficient requires analysis of an actual design with detailed design methods. The more negative MTC is not expected to lead to unacceptable results, but this expectation needs to be verified.

3.2.5 DECAY HEAT

The larger inventory of fission products due to the higher burnups of five-batch fuel management (compared with conventional three-batch fuel management) is not expected to significantly affect the decay heat during the first few days after shutdown. The relatively short-lived fission products, which contribute most of the decay heat during this time interval, reach a saturated concentration during the first year of irradiation. Increasing the irradiation time has therefore no significant effect on the decay heat during the first few days after shutdown. Safety analyses, such as those for LOCA events (small or large break), are therefore unaffected by decay-heat considerations. In addition, safety calculations are required to use the American Nuclear Society decay-heat standard based on infinite operation plus a +20% uncertainty, which is conservative for all core burnups.

In addition, the heat-removal requirements of the fuel storage pool are unaffected since the heating requirements are based on the early (larger) heating rates, which are unaffected by increased fuel exposure.

The long-term decay-heat rates (after several months), however, are dependent on the number of years of fuel irradiation, but this effect leads to only a small perturbation. For example, if the decay-heat rate is acceptable 90 days after fuel discharge after 3 years of irradiation, the same heating rate is obtained 105 days after shutdown for 5 years of irradiation. Such a small time increase does not have a significant impact on pool-storage requirements.

The long-term decay-heat rates (after several years) are not expected to have a significant impact on the design of a long-term waste-storage facility. Differences between fuel irradiated to 30 MWd/kg and fuel irradiated to 50 MWd/kg will be small in comparison to the margins that will have to be provided in long-term storage facilities.

3.2.6 OFFSITE DOSE

The offsite dose after a reactor accident may be larger because of the larger fission-product inventory as a result of a 5-year fuel-assembly irradiation time. No specific calculations have been performed here to determine the effect of this larger fission-product inventory on reactor operating limits. Such an analysis is recommended for future follow-on work.

3.3 ENVIRONMENTAL CONSIDERATIONS

This section addresses the environmental factors associated with the normal operation of the 50,000-MWd/MT (high-burnup) cycle. As discussed later in this section, the reactor core is assumed to be changed as required to accommodate the high-burnup cycle. The reactor-coolant system, reactor auxiliaries, balance of plant, and site are assumed to be unchanged from the description given in Section 2.1 for the 30,000-MWd/MT case (reference cycle). Therefore, the following sections refer back to the earlier sections where appropriate.

3.3.1 SUMMARY ASSESSMENT

This system is the same as the reference LWR except that advanced fuels capable of operation to 50,000 MWd/MT within current environmental technical specification limits would be used. The nonradiological impacts would therefore be the same as for the reference LWR because the systems are the same. The radiological impacts would be similar to those from the reference case (e.g., on presently operating reactors) because existing technical specifications would prohibit larger impacts. Furthermore, the results of ongoing fuel-development programs indicate that the fuel integrity to meet these requirements can be achieved.

3.3.2 REACTOR AND STEAM-ELECTRIC SYSTEM

The information given in Section 2.3.2 for the reference cycle applies also to the high-burnup cycle.

3.3.3 STATION LAND USE

The information given in Section 2.3.3 for the reference cycle applies also to the high-burnup cycle.

3.3.4 STATION WATER USE

The information given in Section 2.3.4 for the reference cycle applies also to the high-burnup cycle.

3.3.5 HEAT-DISSIPATION SYSTEM

The information given in Section 2.3.5 for the reference cycle applies also to the high-burnup cycle.

3.3.6 RADWASTE SYSTEMS AND SOURCE TERMS

Sources of radioactivity, release paths, and processing systems are the same as those of the reference cycle. The principal assumptions and plant parameters applicable to the reference cycle are given in Table 2-21. These data are also applicable to the high-burnup cycle.

Radioactivity originates from fission products, from the activation of core materials, and from the activation of coolant chemicals. Coolant chemistry and core structural materials are not significantly changed from the reference to the high-burnup cycle.

For a burnup of 50,000 MWd/MTU, the fission-product inventory of long-lived nuclides (principally cesium-134, cesium-137, iron-57, cobalt-60, tritium, and krypton-85) is increased by a factor of 1.67 to 2.0. There is a slight increase in the liquid radioactive source term (which includes additional sources), from 0.4 to 0.415 Ci/yr. The tritium source term (from ternary fissions) is more substantial (up to 583-700 Ci/yr from 350 Ci/yr). Since coolant chemistry is virtually unaffected, tritium from other sources is not affected. Short-lived nuclides are not increased because they become saturated.

Since the coolant activity limits currently imposed on LWRs would not be changed with the introduction of high-burnup fuels, the quantities of radionuclides available for release from normal operation would be unaltered.

The reference fuel is a late-generation design for which operational performance can only be estimated. Given the operating experience of PWR vendors as background, the fuel-failure fraction should be at the low end of the 1 in 10,000 to 10 in 10,000 range currently experienced in the nuclear industry. Experience with high-burnup test assemblies to date indicates that there are no new detrimental fuel-behavior phenomena.

3.3.7 CHEMICAL AND BIOCIDAL WASTES

The information presented in Section 2.3.7 is applicable to the high-burnup cycle.

3.3.8 EFFECTS OF OPERATION OF HEAT-DISSIPATION SYSTEM

The information presented in Section 2.3.8 is applicable to the high-burnup cycle.

3.3.9 RADIOLOGICAL IMPACT FROM ROUTINE OPERATIONS

The radiological impacts will be similar to those from the reference LWR because existing technical specifications (e.g., those for existing reactors) would prohibit higher impacts. The ratio of isotopes on normal releases would be somewhat different than for the reference LWR; high burnup of fuel would result in relatively more long-lived isotopes in the releases. This is shown as the percentage contribution by isotope to various dose components in Tables 3-9 through 3-12.

3.3.10 EFFECTS OF CHEMICAL AND BIOCIDAL DISCHARGES

The information presented in Section 2.3.10 is applicable to the high-burnup cycle.

3.3.11 OCCUPATIONAL EXPOSURES

Occupational exposures would not be significantly increased for the high-burnup cycle since the plant would be operated within the same technical specification limits on coolant activity as the reference (30-MWd/kg) cycle.

Table 3-9. Contributions to dose due to liquid effluents

Isotope	Percentage	
	Adult total body	Critical organ
Tritium	11	3
I-131	(a)	85
I-133	(a)	10
Cs-134	54	(a)
Cs-136	2	(a)
Cs-137	32	(a)
Others	1	2

[a]Less than 1%.

Table 3-10. Contribution of noble-gas emissions to total-body and skin doses

Isotope	Percentage	
	Total body	Skin
Kr-83m	(a)	(a)
Kr-85m	(a)	(a)
Kr-85	(a)	17
Kr-87	1	1
Kr-88	8	4
Kr-89	(a)	(a)
Xe-131m	(a)	(a)
Xe-133m	1	2
Xe-133	82	72
Xe-135m	(a)	(a)
Xe-135	4	3
Xe-137	(a)	(a)
Xe-138	(a)	(a)
Ar-41	3	1

[a]Less than 1%.

Table 3-11. Contribution of radioiodine and particulates to child thyroid doses

Isotope	Percentage
I-131	92
I-133	1
Co-60	1
Co-58	(a)
Fe-59	(a)
Mn-54	(a)
Cs-137	1
Cs-134	(a)
Sr-90	0
Sr-89	0
Tritium	2
C-14	2

[a]Less than 1%.

Table 3-12. Contribution of radioiodine and particulates to infant thyroid doses

Isotope	Percentage
I-131	98
I-133	2
Co-60	0
Co-58	0
Fe-59	0
Mn-54	0
Cs-137	0
Cs-134	0
Sr-90	0
Sr-89	0
Tritium	(a)
C-14	(a)

[a]Less than 1%.

3.4 LICENSING STATUS AND CONSIDERATIONS

Preliminary investigations by Combustion Engineering, Inc. (Ref. 1) indicate that the high-burnup design with a five-batch fuel-management scheme can be commercialized, after some modifications in the design of the fuel rod and core of the reference PWR as given in Reference 2. A specific design of fuel rod and core must be identified before any detailed safety evaluation and licensability assessment can be made. Although no fundamental problems are expected, the safety analysis must be updated to insure that the results are within those of the NRC acceptance criteria.

An experimental program is needed to confirm fuel-rod integrity for the high-burnup range that is required. A number of such programs are being proposed to DOE by various fuel vendors. The information required for design and licensing includes the following:

1. Percentage release of fission gases at higher burnups
2. Corrosion of the cladding surface resulting from increased residence time
3. Pellet-cladding interaction
4. Structural stability

The reactor characteristics pertinent to safety (which have been identified and discussed in Section 3.2) must be ascertained for the modified design. Data or evaluations are required in the following areas:

1. Power and temperature coefficients
2. Fuel-design limits and bases
3. ECCS performance, fuel thermal performance, core thermal-hydraulics
4. Accident spectrum for the safety evaluation of all aspects of the reactor system, including balance of plant
5. Fission-product inventory for activity source term in accident analyses

Also to be evaluated for a specific design are:

1. Xenon stability
2. Poison requirements for shutdown and refueling
3. Control requirements
4. Axial and radial peaking factors
5. Adequacy of engineered safeguard systems
6. Safety margins for normal operating maneuvers and for accidents
7. Calibration requirements of in-core and out-of-core instruments
8. Fuel-storage-pool licensability with respect to criticality and cooling requirements
9. Offsite dose commitments for normal operation and accident conditions

3.5 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

As discussed in Section 3.1, an increase in discharge burnup of PWR fuel is possible with modest changes in fuel and core designs and minor changes in fuel-management practices. The PWR NSSS and balance of plant are not different from those already deployed in the United States. Consequently, no basic reactor research and development is required.

Research and development, however, is required to develop a fuel-rod design and a core configuration capable of achieving the sizable increase in discharge burnup postulated in this report. Research and development is also required to demonstrate satisfactory performance of the fuel rods and fuel assemblies. For batch-average burnups of 50,000 MWd/MT, the average burnup of the highest duty fuel rods is in the range of 60,000 MWd/MT (solid pellets) to 75,000 MWd/MT (annular pellets), a range that is considerably higher than that for current PWRs (about 37,000 MWd/MT). The present knowledge of the irradiation behavior of uranium dioxide or thorium dioxide for exposures of this magnitude under PWR operating conditions is limited. Fission-gas release as a function of burnup and temperature, pellet-cladding interaction, dimensional stability, and corrosion behavior are some of the phenomena that are being investigated. The initial phase of this research and development program consists of few-assembly irradiations in which the peak discharge exposures anticipated in the optimized design are attained. The purpose of these few-assembly irradiations is to provide an early indication of the performance of fuel irradiated to high burnup and to provide information, such as fission-gas release, for the design of the high-burnup fuel. This part of the research and development program can most quickly be accomplished by reinserting spent PWR fuel for more irradiation.

The next phase of the research and development program consists of the design, construction, and irradiation of demonstration assemblies designed explicitly for high burnup. The design of this high-burnup fuel is similar to that of the present PWR fuel, but changes in design detail to accommodate the changed performance parameters identified in the initial phase of the research and development program are anticipated. These lead assemblies will be followed by an entire batch loading of high-burnup fuel.

The irradiation experiments will be followed up by postirradiation examinations, as well as theoretical efforts to correlate the experimental data and to develop analytical models for the design of fuel rods.

Establishing a testing and verification program for the modified design also involves some research and development. The program must address the possible changes in the ratio of departure from nucleate boiling, linear heat rate, peaking factors, critical heat flux, fuel densification (for annular pellets), and integrity of rods and assemblies.

REFERENCES FOR CHAPTER 3

1. Combustion Engineering, Inc. Improvements in Once-Through PWR Fuel Cycles, Interim Progress Report for Fiscal Year 1978.

2. Combustion Engineering, Inc., System 80 - Preliminary Safety Analysis Report (PSAR) - DESSAR, Standard PWR-NSSS, Docket No. STN 50-470.

3. NRC recommendations relating fission gas release are presented in R. O. Meyer et al., "Fission Gas Release from Fuel at High Burnup," NUREG-0418 (December 1977).

4. "Nuclear Power Plant Operating Experience - 1976," NUREG-0366 (December 1977).

Chapter 4

SELF-GENERATED PLUTONIUM SPIKED RECYCLE (PWR LEU(5) - PU-SPIKED RECYCLE)

4.1 DESCRIPTION

This reactor/fuel cycle combination is a pressurized-water reactor (PWR) using 3% low-enrichment uranium oxide pellet fuel and self-generated recycle fuel of partially partitioned uranium and plutonium which is spiked with cobalt-60. Fresh makeup fuel is low-enrichment uranium-235 (LEU(5)). Reprocessing wastes and recycle-fuel fabrication wastes will be sent to a geologic waste repository. Makeup-fuel fabrication wastes will be sent to a low-level shallow land disposal site. Low-enrichment uranium recovered from reprocessing makeup fuel will be sent to storage.

The fuel-cycle facilities associated with this reactor/fuel cycle combination, shown in the mass-flow diagram of Figure 4-1, are discussed in the following sections of Volume VII:

Enrichment	Section 3.0
Makeup-fuel fabrication 1	Section 4.1
Recycle-fuel fabrication 3	Section 4.2
Recycle-fuel reprocessing (Purex 5)	Section 5.3
Uranium-235 storage	Section 6.6
Waste disposal 2	Section 7.2
Waste disposal 3	Section 7.3

The primary motivation for uranium and plutonium recycle is the potential conservation of uranium resources. It is estimated that uranium-ore requirements can be reduced by 22% and that uranium-enrichment requirements can be reduced by 14% with the recycle option.

From 1957 through 1972, the U.S. Atomic Energy Commission (AEC) carried out extensive research to develop the technology for uranium and plutonium recycle. This resulted in the establishment of facilities for reprocessing spent fuel from light-water reactors (LWRs) and in U.S. Nuclear Regulatory Commission (NRC) licenses to operate the Big Rock Point, Quad Cities Unit 1, and Dresden Unit 1 reactors with mixed-oxide fuel. The demonstrated technical feasibility and the advantages of uranium and plutonium recycle in LWRs led the NRC and its predecessor, the AEC, to decide that wide-scale recovery and recycle of plutonium fuel in LWRs warranted analysis apart from that given for the licensing of any single recycle facility and that the adoption of rules governing such wide-scale use would constitute a major Federal action with a potential to affect significantly the quality of the human environment. Accordingly, pursuant to the National Environmental Policy Act of 1969 (NEPA), Section 102(2)(C), the NRC prepared a final Generic Environmental Statement on the Use of Recycle Plutonium Mixed Oxide Fuel in Light Water Cooled Reactors (GESMO) (Ref. 1).

The principal NRC staff findings based on evaluations of the health, safety, and environmental (but not safeguards) effects of wide-scale recycle of plutonium as fuel for LWRs are as follows:

1. The safety of reactors and fuel-cycle facilities is not affected significantly by the recycle of fissile materials.

2. Nonradiological environmental impacts resulting from the recycle of fissile materials from spent fuel are slightly smaller than those from a fuel cycle that does not reclaim residual fuel values.
3. Plutonium recycle extends uranium resources and reduces uranium-enrichment requirements, but it makes necessary the reprocessing and fabrication of plutonium-containing fuels.
4. While there are uncertainties, wide-scale recycle has a likely economic advantage versus a fuel cycle that does not reclaim residual fuel values.
5. Differences in health effects attributable to recycle provide no significant basis for the selection of a fuel-cycle option.
6. No waste-management considerations were identified that would bar the recycle of uranium and plutonium.

A self-generated plutonium recycle is chosen for providing the preliminary technical and economic data for the Nonproliferation Alternative Systems Assessment Program (NASAP), and its performance in the reference Combustion Engineering Inc., (C-E) System 80 PWR is assessed. (Similar performance could be achieved in other LWR designs.) A reference three-batch fuel-management scheme is employed. All the physical characteristics of the C-E System 80 (see Section 4.2.1 of Reference 2) have been retained in the core design, with the exception of the composition of the fuel pellets. The pellets are composed of a mixture of uranium and plutonium oxides. The fissionable material consists of both plutonium and uranium-235.

The generalized reactor performance characteristics are summarized in Table 4-1; the reactor-design data are summarized in Table 4-2. Additional data on fuel management are presented in Section 4.1.4.

4.1.1 FUEL MECHANICAL DESIGN

Except for the composition of the fuel pellets, the fuel mechanical design is the same as that of the reference PWR design, as given in Section 4.2.1 of Reference 2. The pellets are made of a mixture of plutonium dioxide and uranium dioxide powder. The powder undergoes comminution, compaction, and granulation to the desired consistency before pelletization. A great deal of experience has been gained in the design and performance evaluation of the mixed-oxide fuel from the following sources:

1. Plutonium utilization program (Ref. 3)
2. Plutonium-recycle experiment (Ref. 4)
3. Saxton plutonium project (Ref. 5)
4. Evaluation of mixed-oxide fuel in BWRs (Ref. 6)
5. Dresden plutonium-recycle demonstration program
6. Experience in the Big Rock Point reactor

Because of their ionic and crystalline similarities, uranium dioxide and plutonium dioxide form a complete solid solution. The physical and mechanical properties of the mixture are not drastically different from those of uranium dioxide. These properties are given in detail in Section 3.4.1 of Reference 1. The following characteristics of the mixed oxide are noteworthy:

1. Theoretical density is slightly higher
2. Melting point is lower
3. Thermal conductivity is lower
4. Thermal expansivity is the same
5. Enthalpy and specific heat are approximately the same

6. Brittle-fracture strength is somewhat lower
7. Plasticity is greater

Experiments have shown that the irradiation performance of the mixed-oxide fuel is also very similar to that of the uranium dioxide fuel. For example, the fission-gas release rate, swelling rate, and densification do not differ significantly. Performance characteristics are discussed in detail in Section 3.4.2 of Reference 1.

4.1.2 FUEL NUCLEAR DESIGN

A detailed nuclear design of the core has not been performed. However, the necessary nuclear-property data, design methods, and computer codes are available, although some improvements may be justifiable.

Plutonium fissioning in oxide fuels is not unique to recycled plutonium fuels. For example, near the end of an equilibrium cycle, a typical uranium dioxide core at a core-averaged exposure of 20,000 MWd/MTU will derive approximately 50% of its power from the fissioning of bred-in plutonium isotopes. Thus, in one sense, the use of plutonium as fuel in LWRs is not represented as a new situation.

The nuclear properties of mixed-oxide fuels differ in some extent from those of uranium dioxide, notably in the increased neutron cross section of the plutonium isotopes and the corresponding decrease in control-rod worth. The altered nuclear properties can be accommodated in most cases by using various rod-placement and enrichment schemes that make it feasible to design fuel assemblies that are interchangeable with the spent uranium dioxide assemblies they replace.

The following are some of the changes in the nuclear characteristics of a mixed-oxide core with respect to those of the reference PWR core (Ref. 2). A detailed discussion is available in Section 3.3 of Reference 1.

1. More negative moderator-temperature coefficient
2. More negative Doppler coefficient
3. Somewhat more severe local power peaking
4. Reduced control-rod worth
5. Improved xenon stability
6. Reduced soluble-boron worth
7. Reduced delayed-neutron fraction
8. Reduced prompt-neutron lifetime

These changes have some effect on the performance and safety characteristics of a mixed-oxide core. However, it is feasible to design the core so that the performance and safety characteristics will approach those of the reference core. To accomplish this, it may be necessary to limit the number of fuel rods that have plutonium as the major fissile material.

4.1.3 FUEL THERMAL-HYDRAULIC DESIGN

Available data indicate that the mixed-oxide fuel would have a lower melting point as well as a lower thermal conductivity. These facts should be accounted for in the fuel-rod thermal design, although no significant change from the reference design is expected. Detailed thermal-hydraulic design and its evaluation cannot be performed until a satisfactory nuclear design is established. However, the design methodology, data, and evaluation schemes are all available and no new computer codes

need be developed. Pertinent information on the reference PWR is available in Section 4.4 of Reference 2.

4.1.4 FUEL MANAGEMENT

Fuel-management information is summarized in Table 4-3. Fuel-cycle information presented in this table is based on a fuel-management scheme similar to that currently employed in PWRs, in which one-third of the core is replaced at annual refueling intervals. The yellowcake and separative work requirements tabulated in Table 4-3 assume that plutonium is fully recycled.

The isotopic distributions of the fuel inventory at the beginning and end of the equilibrium cycle are listed in Tables 4-4, and 4-5, respectively. The reactor-charge data for the makeup fuel and the recycle fuel are given in Tables 4-6 and 4-7, respectively; the reactor discharge data for the makeup fuel and the recycle fuel are given in Tables 4-8 and 4-9, respectively.

Table 4-1. Reactor system design and performance data (PWR with U/Pu recycle)

Reactor Type	PWR
Reactor thermal power output (gross), MW	3,817
Electrical power output,[a] MW	
Gross	1,344
Net	1,270
Plant heat rate (Btu/kW-hr)	10,212
Core design and performance parameters	
Core heat output, MW	3,800
Core volume, liters	40,050
Core loading, kg[b]	
Heavy metal	99,313
Fissile fuel	2,201
Conversion ratio[c]	0.60
Average discharge exposure, MW/MTHM[d]	30,390
Peak discharge exposure, MWd/MTHM[d,e]	38,900
Fuel type	Oxide
Reactor inlet temperature, ^{o}F	565
Reactor outlet temperature, ^{o}F	621
End-of-cycle excess reactivity	Zero(0)

[a]Depends on architect-engineer; these values assume mechanical-draft cooling.

[b]Initial UO_2 core.

[c]Integrated conversion equilibrium cycle.

[d]Heavy-metal charged.

[e]Rod average; max pellet 55,000 MWd/MTHM.

Table 4-2. Reactor design data specifications
(PWR with U/Pu recycle)

Geometric information	
Core height, cm	381.0
Number of core enrichment zones (nominal)	3
Number of assemblies	241
Equivalent diameters, cm	365.8
Pins per assembly	236
Pin pitch-to-diameter ratio	1.325
Overall assembly length, cm	406.4
Lattice pitch, cm	1.288
Assembly material	Oxide fuel with Zircaloy-4 cladding
Cladding parameters	
Cladding outside diameter, mils	382.7
Cladding wall thickness, mils	25
Cladding material	Zircaloy-4
Fissile inventory at beginning of equilibrium cycle, kg	3,150
External fissile inventory, kg	NA
Fissile loss, kg/cycle[a]	637
Specific power, kW/kg-fissile[b]	1,206
Power density, kW/kg HM	38.3

[a]Equilibrium cycle.
[b]Beginning of equilibrium cycle.

Table 4-3. Fuel management information
(PWR with U/Pu recycle)

Average capacity factor, %	75
Approximate fraction of core replaced annually	One-third
Lag time assumed between fuel discharge and recycle reload, yr	2-yr
Fissile material reprocessing loss fraction, %	1
Fissile material fabrication loss fraction, %	1
Yellowcake requirements, short tons/GWe	
Initial core	408
Annual equilibrium reload requirement	120
30-year cumulative requirements, 10^3 SWU/GWe	4,190
Separative-work requirements, 10^3 SWU/GWe	
Initial core	212
Equilibrium reload	83.7
30-year cumulative requirement	2,750
Requirements for special fuel materials (fissile Pu, U-233, etc.), kg HM/GWe	
Initial load	0
Annual equilibrium charge, discharge	0
30-year cumulative requirement	0
Other data for proliferation-resistance assessment	
Fuel element weight, kg	650
Fresh- and discharge-fuel radiation level at 1 meter, R/hr	Not calculated
Discharge-fuel energy-generation rate after 90-day cooling (watts per element)	Uranium dioxide assembly: 12,600 Mixed oxide assembly: 20,500
Spiking level at 1 meter at 6 months, R/hr	1,000

Table 4-4. Fuel inventory at the beginning-of-equilibrium cycle (PWR U/Pu recycle)

Isotope	Fuel inventory (kg)					
	Fresh makeup fuel, zone 1	Once-burnt makeup fuel, zone 2	Twice-burnt makeup fuel, zone 3	Fresh recycle fuel, zone 4	Once-burnt recycle fuel, zone 5	Twice-burnt recycle fuel, zone 6
Th-232						
Pa-233						
U-232						
U-233						
U-234						
U-235	719.4	462.4	300.3	67.2	57.86	49.74
U-236		11.7	73.7	--	2.45	4.3
U-238	23,261.49	23,082.0	22,902.0	9,397.8	9,351.5	9,288.6
Pu-238		0.24	1.21	--	0.29	1.40
Pu-239		87.7	113.0	338.5	292.6	257.0
Pu-240		16.8	35.9	252.3	241.2	229.5
Pu-241		7.3	20.2	127.2	124.5	120.8
Pu-242		0.75	4.5	96.8	95.9	95.6
Fission products		272.0	537.7		113.9	217.5
Other isotopes						
Am-241		0.08	0.37		4.23	6.4
Cm-242		--	--		--	--
Np-237		2.51	6.28		0.53	1.09

Table 4-5. Fuel inventory at the end-of-equilibrium cycle (PWR U/Pu recycle)

Isotope	Fuel inventory (kg)					
	Once-burnt makeup fuel, zone 1	Twice-burnt makeup fuel, zone 2	Thrice-burnt makeup fuel, zone 3	Once-burnt recycle fuel, zone 4	Twice-burnt recycle fuel, zone 5	Thrice-burnt recycle fuel, zone 6
Th-232						
Pa-233						
U-232						
U-233						
U-234						
U-235	462.3	300.3	191.7	57.9	49.8	42.5
U-236	11.7	73.7	89.2	2.5	4.3	5.8
U-238	23,081.5	22,901.5	22,713.5	9,337.5	9,304.5	9,204.9
Pu-238	0.24	1.21	3.02	0.30	1.40	3.11
Pu-239	87.7	112.9	119.9	293.2	257.5	231.1
Pu-240	16.8	35.9	50.3	241.6	229.9	216.1
Pu-241	7.3	20.2	30.2	124.7	121.0	117.4
Pu-242	0.75	4.5	10.7	96.1	95.8	95.1
Fission products	272.0	537.7	745.2	113.8	217.8	315.2
Other isotopes						
Am-241	0.08	0.37	0.69	4.24	6.40	7.34
Cm-242	--	--	--	--	--	--
Np-237	2.51	6.28	10.31	0.53	1.10	1.67

Table 4-6. Reactor charge data for zones 1, 2, 3, etc.
(PWR U/Pu recycle--makeup fuel)

	Charge (kg)													
Year	Th-232	Pa-233	U-232	U-233	U-234	U-235	U-236	U-238	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	Np-237	Total heavy metal
1						2,201		97,112		0	0	0	0		99,313
2						1,071		33,048		0	0	0	0		34,119
3						1,028		33,517		0	0	0	0		34,545
4						828		26,457		0	0	0	0		27,295
5						886		26,409		0	0	0	0		27,295
6						800		25,216		0	0	0	0		26,016
7						791		24,798		0	0	0			25,589
8						811		24,778		0	0	0			25,589
9						763		23,547		0	0	0			24,310
10						746		23,137		0	0	0			23,883
11						755		23,128		0	0	0			23,883
12						754		23,556		0	0	0			24,310
13						742		23,141		0	0	0			23,883
14						726		21,451		0	0	0			22,177
15						721		21,883		0	0	0			22,604
16						709		21,468		0	0	0			22,177
17						714		21,463		0	0	0			22,177
18						718		21,886		0	0	0			22,604
19						708		21,469		0	0	0			22,177
20						714		21,463		0	0	0			22,177
21						↓		↓		↓	↓	↓	↓		↓
22															
23															
24															
25															
26															
27															
28															
29															
30						714		21,463		0	0	0	0		22,177

Table 4-7. Reactor charge data for zones 1, 2, 3, etc.
(PWR U/Pu recycle--recycle fuel)

Year	Charge (kg)														Total heavy metal
	Th-232	Pa-233	U-232	U-233	U-234	U-235	U-236	U-238	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	Np-237	
1						0		0		0	0	0	0		0
2						0		0		0	0	0	0		0
3						0		0		0	0	0	0		0
4						47		6,583		132	40	18	4		6,824
5						47		6,545		140	56	27	9		6,824
6						59		8,196		160	66	35	14		8,530
7						59		8,174		176	68	38	14		8,529
8						59		8,174		176	69	38	14		8,530
9						70		9,786		202	100	54	23		10,235
10						70		9,761		212	107	58	27		10,205
11						70		9,742		215	114	62	33		10,236
12						70		9,732		220	117	64	33		10,236
13						70		9,731		221	117	64	33		10,236
14						82		11,376		233	135	74	42		11,942
15						81		11,363		237	140	76	45		11,942
16						81		11,349		241	144	78	49		11,942
17						81		11,337		246	147	80	50		11,941
18						81		11,341		244	147	80	49		11,942
19						81		11,320		246	153	83	58		11,941
20						81		11,306		250	157	86	61		11,941
21						↓		↓		↓	↓	↓	↓		↓
22															
23															
24															
25															
26															
27															
28															
29															
30						81		11,306		250	157	86	61		11,941

Table 4-8. Reactor discharge data for zones 1, 2, 3, etc.
(PWR U/Pu recycle--recycle fuel)

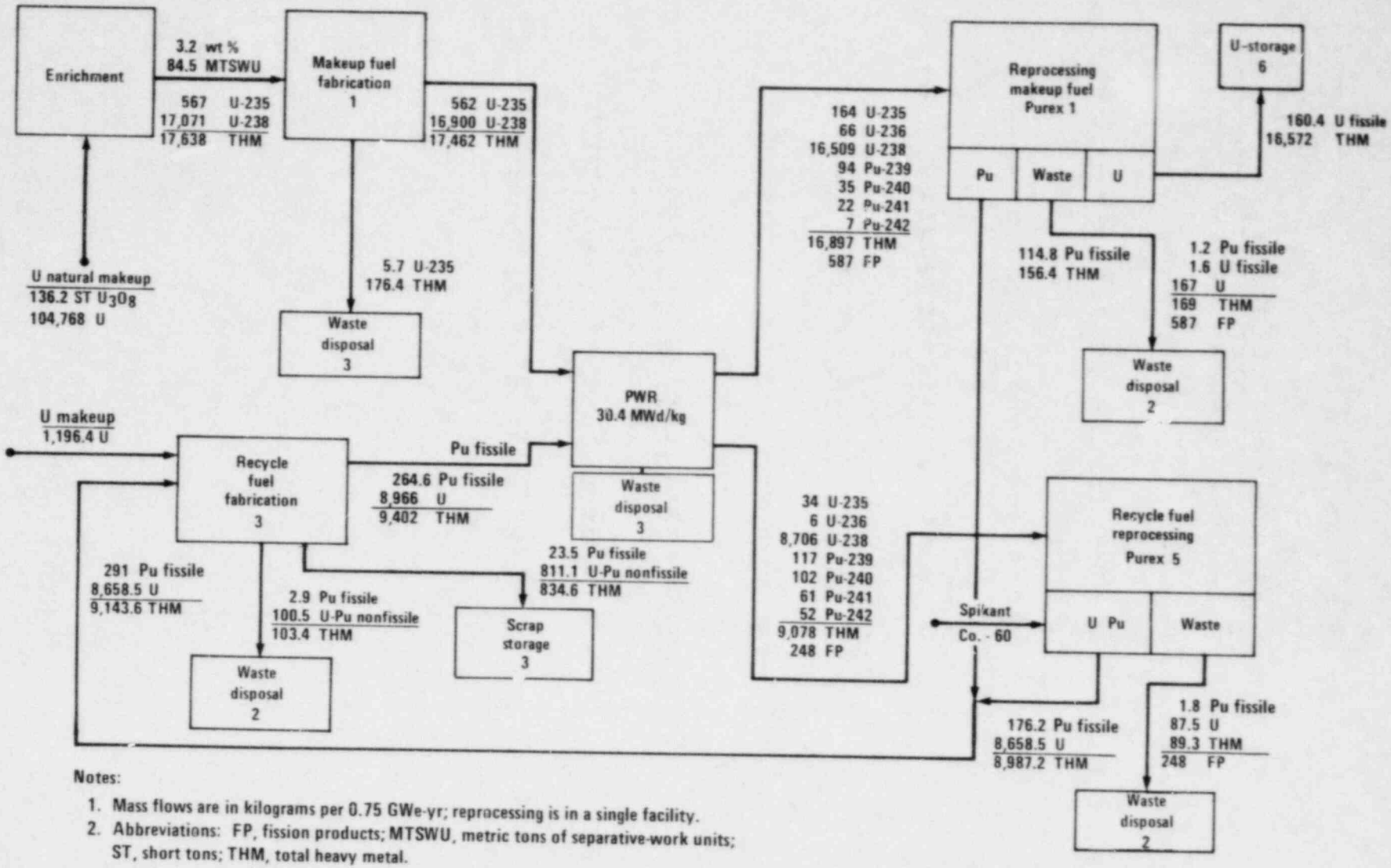
Year	Discharge (kg)															
	Th-232	Pa-233	U-232	U-233	U-234	U-235	U-236	U-238	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	Np-237	Total heavy metal	Fission products
1						0	0	0		0	0	0	0		0	0
2						0	0	0		0	0	0	0		0	0
3						0	0	0		0	0	0	0		0	0
4						0	0	0		0	0	0	0		0	0
5						0	0	0		0	0	0	0		0	0
6						19	5	6,427		64	47	27	12		6,601	180
7						21	5	6,395		72	56	32	17		6,598	180
8						26	6	8,006		84	64	38	23		8,248	225
9						27	6	7,990		92	69	41	23		8,248	225
10						27	6	7,989		92	69	41	23		8,247	225
11						33	7	9,564		112	90	54	33		9,893	270
12						34	7	9,543		118	96	57	36		9,891	270
13						35	7	9,526		121	100	60	41		9,890	270
14						36	7	9,518		125	103	61	41		9,891	270
15						36	7	9,517		125	103	61	41		9,890	270
16						40	8	11,121		134	114	68	51		11,536	316
17						41	8	11,109		138	117	70	54		11,537	316
18						41	8	11,097		140	120	72	57		11,535	316
19						42	8	11,086		144	123	74	57		11,534	316
20						42	8	11,088		143	123	73	57		11,534	316
21						42	8	11,070		145	126	75	64		11,530	315
22						43	8	11,057		148	130	77	66		11,529	315
23						↓	↓	↓		↓	↓	↓	↓		↓	↓
24																
25																
26																
27																
28																
29																
30						43	8	11,057		148	130	77	66		11,529	315

NOTE: Discharge exposure is 30,400 MWd/MT.

Table 4-9. Reactor discharge data for zones 1, 2, 3, etc.
(PWR U/Pu recycle--makeup fuel)

Year	Discharge (kg)															
	Th-232	Pa-233	U-232	U-233	U-234	U-235	U-236	U-238	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	Np-237	Total heavy metal	Fission products
1						235	55	33,167		137	42	21	4		33,661	380
2						219	81	30,933		135	54	29	9		31,460	730
3						248	110	31,291		160	66	39	14		31,928	1,108
4						301	127	32,271		183	70	44	14		33,010	1,120
5						282	123	32,729		182	71	44	1[illegible]		33,446	1,123
6						226	99	25,838		145	57	35			26,412	917
7						260	104	25,800		147	55	35	11		26,412	917
8						222	95	24,621		139	54	33	11		25,175	874
9						220	94	24,214		137	53	33	11		24,762	860
10						232	96	24,200		137	52	33	11		24,761	860
11						216	90	22,996		130	50	31	10		23,523	817
12						210	88	22,594		128	49	31	10		23,110	803
13						216	89	22,588		128	49	31	10		23,111	803
14						211	90	23,002		130	50	31	10		23,524	817
15						208	88	22,597		128	49	31	10		23,111	803
16						215	85	20,958		120	45	28	9		21,460	745
17						208	85	21,374		122	46	29	9		21,873	760
18						205	84	20,969		119	45	28	9		21,459	745
19						208	84	20,966		119	45	28	9		21,459	745
20						206	85	21,376		121	46	29	9		21,872	760
21						204	84	20,970		119	45	28	9		21,459	745
22						208	84	20,966		119	45	28	9		21,459	745
23						↓	↓	↓		↓	↓	↓	↓		↓	↓
24																
25																
26																
27																
28																
29																
30						208	84	20,966		119	45	28	9		21,459	745

NOTE: Equilibrium discharge exposure is 30,400 MWd/MT.



Notes:

1. Mass flows are in kilograms per 0.75 GWe-yr; reprocessing is in a single facility.
2. Abbreviations: FP, fission products; MTSWU, metric tons of separative-work units; ST, short tons; THM, total heavy metal.

Figure 4-1. Mass flows for the uranium-plutonium spiked recycle, PWR LEU(5)-Pu.

4.2 SAFETY CONSIDERATIONS

Extensive safety and licensing reviews have been performed for mixed-oxide fuels at the Big Rock Point, Quad Cities Unit 1, and Dresden Unit 1 reactors, and no fundamental safety-related problems have been identified for plutonium recycling in LWRs. Since the reactor plant employed in this study is the reference C-E System-80 PWR, it can be readily concluded that the system is fundamentally licensable for plutonium recycling. All the physical properties data, design methodology, and computer codes are available, although some improvements may be justifiable.

Because there are differences (though not large) in the neutronic and physical characteristics of the mixed-oxide and uranium dioxide cores (as discussed in Section 4.1), the transient as well as accident behavior of the mixed-oxide core will be somewhat different from those of the reference PWR core. Detailed analysis for plutonium recycling has not been performed; however, a qualitative discussion is given below for events of moderate and low frequency.

Events of moderate frequency that produce anticipated operational transients fall into three general categories:

1. Those that cause an increase in power
2. Those that cause an increase in coolant temperature
3. Those that cause a decrease in coolant temperature

The more negative Doppler, moderator-temperature, and void-reactivity coefficients in a mixed-oxide-fueled reactor will make the first type of transient, such as uncontrolled rod-bank withdrawal, less severe. The smaller delayed-neutron fraction and shorter prompt-neutron lifetime potentially make the first type of transients more severe for the mixed-oxide reactors, but the more negative coefficients are controlling. The boron-dilution transient would be less severe with mixed-oxide fuels because the soluble-boron worth is less. Plutonium segregation could occur during sustained overpower operation that caused centerline temperatures to rise, but the consequences of such segregation are judged to be unimportant.

An example of the second type of transient is the loss of turbine load. In this case, the more negative moderator-temperature coefficient of a mixed-oxide core would make the temperature and pressure transients less severe in a PWR but potentially more severe in a BWR because of the reactivity increase resulting from rapid void collapse.

In the third type of transient, such as startup of an inactive coolant loop, the more negative moderator and void coefficients of a mixed-oxide core would tend to be somewhat detrimental. Because sufficient shutdown margin is always maintained, however, the consequences of this transient are not serious.

The more serious design-basis accidents that are postulated for LWRs have been analyzed by several fuel vendors by comparing plutonium-recycle cores to uranium dioxide cores. The more negative Doppler, moderator-temperature, and void-reactivity coefficients compensate for the lower delayed-neutron fraction and the shorter prompt-neutron lifetime, and the consequences of the accidents are comparable for mixed-oxide and uranium oxide cores except for the accidents discussed below.

PWR Steam-Line Break

The PWR steam-line-break accident results in a rapid cooling of the core and a potential return to criticality because of the negative moderator-temperature coefficient. To prevent this, more control rods or a higher boron-injection rate may be required. More restrictive fuel management will be required to minimize such changes.

Rod Ejection

The postulated rod-ejection accident for a mixed-oxide-fueled reactor may be more or less severe, depending on the core design. More negative reactivity coefficients and lower ejected-rod worths are advantageous, while the lower delayed-neutron fraction, the shorter prompt-neutron lifetime, and delayed Doppler feedback, when large plutonium dioxide agglomerates are present, are detrimental.

Loss of Coolant

The consequences of a LOCA event are not appreciably different for uranium dioxide and mixed-oxide cores. Several factors, however, tend to make the accident less severe with mixed oxides. Additional flux depression will compensate for the lower thermal conductivity of mixed-oxide fuel so that the stored energy will be somewhat less. Expecially where annular pellets are used, the stored energy of a mixed-oxide fuel might be significantly reduced. The lower delayed-neutron fraction and shorter prompt-neutron lifetime make the decay of neutron fissioning after the accident more rapid, resulting in less residual fission power. After 100 seconds, the fission-product decay heat is several percent less for plutonium-239 than for uranium-235 fissions because of the different fission-product yields. The net energy per fission for plutonium-239 is 2 to 3% higher than that for uranium-235 fissions, thus requiring fewer fissions for the same energy output. The result is a somewhat lower short-term decay heat for mixed-oxide rods for a given power rating. At the end of a cycle, when over 50% of the fissions in a uranium dioxide core are from plutonium, the difference between mixed-oxide fuel and uranium dioxide fuel decay heat would be reduced.

4.3 ENVIRONMENTAL CONSIDERATIONS

This section addresses the environmental factors associated with normal operation of the uranium/plutonium spiked recycle fuel cycle. The reactor-coolant system, reactor auxiliaries, balance of plant, and site are assumed to be as described in Section 2.1 for the 30,000-MWd/MTU case (reference cycle).

4.3.1 SUMMARY ASSESSMENT

The spikant does not result in a significant increase in the estimated cobalt-60 content of the coolant (over that in the reference LWR) because the cobalt-60 contribution from the spikant is much less than that from the activation of corrosion/erosion products. The impacts, including the radiological impacts, are therefore estimated to be not significantly different from those of the reference LWR.

4.3.2 REACTOR AND STEAM-ELECTRIC SYSTEM

The information given in Section 2.3.2 for the reference cycle applies also to the recycle case.

4.3.3 STATION LAND USE

The information given in Section 2.3.3 for the reference cycle applies also to the recycle case.

4.3.4 STATION WATER USE

The information given in Section 2.3.4 for the reference cycle also applies to the recycle case.

4.3.5 HEAT-DISSIPATION SYSTEM

The information given in Section 2.3.5 for the reference cycle also applies to the recycle case.

4.3.6 RADWASTE SYSTEMS AND SOURCE TERMS

Sources of radioactivity, release paths, and processing systems are the same as those for the reference cycle. The principal assumptions and plant parameters applicable to the reference cycle are given in Table 2-21. These data are also applicable to the recycle case, with the possible exception of the operating-power fission-product source term (0.25%). This parameter is discussed below.

4.3.6.1 Source Term

The descriptive material in Section 2.3.6.1 is applicable to the recycle case. This section describes the sources of radioactivity and the plant systems that determine the source term. The radioactivity originates from fission products, from the activation of core materials, and from the activation of coolant chemicals. Coolant chemistry and core structural materials are not significantly changed from the reference cycle.

The environmental consequences of using the $(Pu,U)O_2$ fuel relative to LWR UO_2 fuel are discussed in the Generic Environmental Statement on Mixed Oxide Fuels

(Ref. 1), pages IV C-43-108, and source terms for using both fuel types in a 1,000-MWe PWR with U-tube generators are given in Tables IV C-18 and IV C-19. These data are not used directly, but they are valuable for comparative analyses. The general conclusion is that the use of fuel containing up to 5% plutonium in the UO_2 fuel matrix has relatively little effect in changing liquid and gaseous activity levels in comparison with the equivalent reference LWR UO_2 fuel data. The presence of the cobalt-60 spikant in the fuel matrix, however, increases the concentration of a nuclide that is already present as a fission product. The magnitude of the increase may be estimated from GESMO data. According to Table IV C-13, which shows the nuclide inventory before refueling, the core under study already has approximately 427 grams (3.4 ppm) and 362 grams (2.9 ppm) of cobalt-60 for the uranium-only fuel and for the mixed-oxide fuel, respectively. The addition of 6 ppm of cobalt-60 in the mixed-oxide fuel at fabrication would increase the average cobalt-60 inventory throughout operation to 7.5 ppm, which is about 510% of the original cycle average cobalt-60 inventory level. Similar concentration levels are anticipated in the uranium/plutonium spiked recycle fuel. In the event of a fuel failure, it may be assumed that the primary coolant activity from cobalt-60 originating inside the fuel would be increased by a factor of up to 5.1, as compared with the reference case. However, this source of cobalt-60 coolant activity is small in comparison with the activated-corrosion-product source (Refs. 4 and 7), and there would be only a small increase in the total cobalt-60 in the coolant.

Fuel Design and Operational Effects

It is anticipated that the $(Pu,U)O_2$ fuel with the 6-ppm cobalt-60 spikant will use the same design and be operated in the same manner as the fuel cited for the reference case. Consequently, additional fuel failures as a result of the design and operation are not anticipated.

The increased neutron cross section of the plutonium isotopes, the addition of the cobalt-60 spikant, and the corresponding decrease in control-rod worth can be accommodated in most cases by using various rod placement and enrichment schemes. This makes it feasible to design fuel assemblies that are interchangeable with the UO_2 assemblies they replace.

Recent LWR mixed-oxide fuel experience in the United States has demonstrated the performance of this type of fuel. Thousands of $(Pu,U)O_2$ fuel rods have been burned in BWRs, including Dresden 1, Big Rock Point, and Quad Cities 1, over the past decade and have shown no significant adverse effects. Moreover, large numbers of fuel rods have been irradiated in experimental test facilities such as the Plutonium Recycle Test Reactor (PRTR) constructed for this purpose; a comparison facility, the Plutonium Recycle Critical Facility (PRCF); and the Experimental Boiling Water Reactor (EBWR). In addition, related experience with mixed-oxide fuels has come from the liquid-metal fast-breeder reactor (LMFBR) program.

A typical uranium dioxide core near the end of its equilibrium cycle will derive as much as 50% of its power from the fission of bred-in plutonium isotopes. Thus, in one sense, the use of plutonium fuel is not a new situation. Although a number of $(Pu,U)O_2$ fuel rods have been experimentally irradiated and burned in power-production units with no substantial problems recorded, it is not possible to state with high confidence the effects of the cobalt-60 spikant on the fuel. Therefore, a detailed study of the effects of the cobalt-60 spikant on long-term fuel performance would be desirable in future research and development programs.

4.3.6.2 Liquid-Radwaste System

The equipment descriptions and flow diagrams in Section 2.3.6.2 are also applicable to the recycle case. No significant changes in radwaste amounts or activity levels are anticipated for the recycle case as compared with the reference case.

4.3.6.3 Gaseous-Radwaste System

The equipment descriptions and flow diagrams in Section 2.3.6.3 are applicable to the recycle case as well. No significant changes in gaseous radwaste-releases are anticipated for the recycle case as compared with the reference case.

4.3.6.4 Solid Radwastes

The data in Section 2.3.6.4 are applicable also to the recycle case.

4.3.6.5 Comparison with Predicted Releases from Other Studies

On the basis of the foregoing and assuming no change in the fuel-failure fraction from the reference cycle, the release of important nuclides would not increase significantly. Important nuclides would be within the ranges defined in Tables 2-24 and 2-25.

4.3.7 CHEMICAL AND BIOCIDAL WASTES

The information presented in Section 2.3.7 is applicable also to the high-burnup recycle case.

4.3.8 EFFECTS OF OPERATION OF HEAT-DISSIPATION SYSTEM

The information presented in Section 2.3.8 is applicable also to the recycle case.

4.3.9 RADIOLOGICAL IMPACT FROM ROUTINE OPERATION

The radiological impact from routine operation will be the same as for the reference reactor because the amount of cobalt-60 in the coolant that comes from the cobalt-60 in the fuel is small compared to the amount from activated corrosion and wear product.

4.3.10 EFFECTS OF CHEMICAL AND BIOCIDAL DISCHARGES

The information presented in Section 2.3.10 is applicable also to the recycle case.

4.3.11 OCCUPATIONAL EXPOSURES

Occupational exposures would be increased slightly for the recycle case. Doses from operation and from radioactive waste handling would not be affected. Exposures chargeable to refueling would be increased since fresh fuel would arrive in a shielded shipping cask. Additional man-hours and exposure would be incurred in handling the cask, removing the fuel, decontaminating the cask, and so on. The increase in occupational exposure would be only a very small percentage of the total annual occupational exposure.

The addition of spiking material to the fuel should have a negligible effect on in-plant exposures since recycle fuel would be handled remotely during fuel receiving

operations. The contribution of the spiking material to primary system activity should also be negligible compared to that from activation products and fission products, and hence there should be no significant effect on exposure incurred during maintenance.

4.4 LICENSING STATUS AND CONSIDERATIONS

Three major power reactors--Big Rock Point, Dresden Unit 1, and Quad Cities Unit 1--are already licensed for operation with mixed-oxide cores. Moreover, except for operation with mixed-oxide fuel, the reference C-E System 80 PWR has undergone extensive licensing reviews. No significant change in the performance of the reactor during normal operation or accident conditions is expected for plutonium recycling. In the light of the above observations, it is reasonable to conclude that the system is readily licensable.

4.5 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

Extensive research was sponsored by the AEC between 1957 and 1972 to develop the technology for plutonium recycle in LWRs. As a result, detailed data base, analysis techniques, and computer codes (for analysis ar ' design) have been developed. Extensive fuel irradiation experience has also been gained. Consequently, no more major research and development efforts are necessary. However, in view of the fact that experimental data for mixed-oxide cores are not as extensive as for uranium dioxide cores, several observations are in order.

Critical experiments have been small in size, and the larger neutron-leakage effects introduce additional uncertainty in the data. Furthermore, the limited experimental data on localized quantities, such as fuel-cell-neutron-reaction rates for the various isotopes, make it more difficult to determine whether calculations are in agreement or not. In the energy region below 3 electron volts, the complicated cross-section structure makes it difficult to insure that compensating effects are not obscuring errors in the analysis of experimental data. For these reasons, there must be more conservatism in mixed-oxide core-design calculations than in uranium dioxide core-design calculations.

In order to calculate quantities such as the moderator temperature coefficient of reactivity accurately, there is need for a neutron-thermalization computer techni e that adequately treats the complicated resonance-cross-section region below 3 electron volts. It is desirable to have the thermal cutoff--between the fast and thermal calculations--well above the 1.05-electron-volt resonance of plutonium-240. Commonly used codes such as THERMOS have a weakness in that the number of groups available (~35) does not give sufficient resolution to treat resonances properly. In theory, it is possible to generate libraries for use in integral-transport-theory codes with any number of thermal groups. This is not normally done in the industry for the heterogeneous lattices. Another technique is to perform Monte Carlo calculations in the range 0 to 3 electron volts and then to compute correlation factors for use with codes that have a thermal cutoff of 0.625 electron volt.

In determining effective fast-group cross sections, a calculational method that explicitly determines the self-shielding and Doppler broadening in the plutonium-240 and plutonium-242 resonances is needed. A typical method is that of Nordheim (Refs. 8 and 9), which has been incorporated into several fast-neutron-spectrum codes such as GAM-II (Ref. 10).

Particle self-shielding effects in mixed-oxide fuels are probably unimportant from a nuclear standpoint because most vendors are considering fuels in which all but a few volume percent of the plutonium dioxide particles are smaller than 20 to 50 micrometers in diameter.

Uncertainties in the calculation of safety-related quantities such as reactivity coefficients, control-rod worths, and power distributions can be accommodated in the design. Some increase in design margins may be necessary to allow for a possible increase in the uncertainty of core parameters in a mixed-oxide core and may involve economic penalties. Therefore, continued improvement in the data base and calculational techniques is well justified.

A comprehensive review of the status of experimental work on plutonium, both in operating reactors and in critical experiments, has been presented by Uotinen et al. (Ref. 11). The paper, with its 130 references, also discusses problem areas in calculational techniques.

REFERENCES FOR CHAPTER 4

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Chapter 5

DENATURED URANIUM-233/THORIUM CYCLE (PWR DU(3)-Th RECYCLE DU(3))

5.1 DESCRIPTION

This reactor/fuel cycle combination is a pressurized-water reactor (PWR) using 12% uranium-233 denatured with uranium-238 and mixed with thorium oxide to fabricate pellet recycle fuel. The spent fuel is reprocessed to recover the uranium-233/uranium-238 which is blended with additional uranium-233 from a secure storage center to 12% fissile assay. Recovered plutonium is spiked and sold to a secure storage center. Reprocessing wastes and recycle fuel fabrication wastes will be sent to a geologic waste repository. Thorium is placed in interim storage for 10 years.

The fuel-cycle facilities associated with this reactor/fuel cycle combination shown in the mass-flow diagram (Figure 5-1) are discussed in the following sections of Volume VII.

Recycle-fuel fabrication 3	Section 4.1
Reprocessing (Thorex 3)	Section 5.5
Waste disposal 2	Section 7.2
Waste disposal 3	Section 7.3

To provide preliminary technical and economic data for the Nonproliferation Alternative Systems Assessment Program (NASAP), a specific denatured uranium-233/thorium cycle has been chosen, and its performance when incorporated into the reference Combustion Engineering, Inc., (C-E) System 80 PWR is assessed. (Similar performance could be achieved in other light-water reactor (LWR) designs.) A reference three-batch fuel-management scheme is employed. All physical characteristics of the C-E System 80 (see Section 4.2.1 of Reference 1) have been retained in the core design with the exception of the composition of the fuel pellets.

The generalized reactor-performance characteristics are summarized in Table 5-1, and the reactor-design data are summarized in Table 5-2. Additional data on fuel management are presented in Section 5.1.4.

The concept of uranium denatured with thorium has received only limited attention so far. Some experience has been gained from the designs of the Shippingport, Indian Point, and Elk River reactors in terms of the use of uranium dioxide-thorium dioxide in fuel pellets. Introduction of this mixed-oxide form of fuel in commercial PWRs would have to be preceded by significant research and development efforts to generate the necessary data base for nuclear design and licensing assessment. This is discussed further in Section 5.5. However, before any serious research and development program is launched, some scoping studies are required to quantitatively assess the potential benefits of the denatured-uranium/thorium fuel cycle. A feasibility study on the use of thorium fuel cycles in PWRs has been performed by Combustion Engineering under the sponsorship of the Electric Power Research Institute (EPRI). The study is reported in detail in Reference 2. This study examined the potential resource savings, technical feasibility, and economic motivation for employing thorium-based fuel cycles in present PWRs. The program was divided essentially into four major tasks. The initial phase of the program involved the development and evaluation of the calculational methods required to analyze thorium fuel cycles with a certainty in the results comparable

to that for uranium cycles. These methods were then employed to survey fuel cycles considered feasible for use in unmodified, current-design PWRs. The fuel cycles with the greatest potential for improving fuel utilization were analyzed in greater detail to better define overall resource requirements and to determine the effect of their use on PWR core operating characteristics. The final phase of the study considered the effect on fuel utilization of using thorium-based fuel cycles in modified PWRs. The modifications investigated ranged from simple lattice changes, such as coolant-to-fuel volume ratios, to more advanced PWR-based designs involving changes in both core parameters and operating concepts.

No unique or fundamental constraint is imposed on the deployment of the thorium fuel cycle by reprocessing or fabrication since these technologies appear to be relatively well developed and/or similar to those required in the uranium cycle; however, no commercial facilities for thorium reprocessing or fabrication currently exist or are planned. Irradiation performance of thorium-bearing fuels has been satisfactory and indicates no problems that might preclude thorium fueling.

5.1.1 FUEL MECHANICAL DESIGN

Except for the composition of the fuel pellets, the fuel mechanical design is the same as that of the reference PWR design, as given in Section 4.2.1 of Reference 1. The pellets are made of vibratory compacted thorium dioxide-uranium dioxide powder. Insufficient data on the physical properties and irradiation behavior of the fuel are available to evaluate the mechanical fuel design.

5.1.2 FUEL NUCLEAR DESIGN

A detailed nuclear design of the core has not been performed. Nuclear-property data for denatured-uranium/thorium (such as the resonance absorption cross section) are not well established. When data become available, the presently available computer codes can be employed (perhaps with some minor modifications) for the nuclear design of the core.

5.1.3 FUEL THERMAL-HYDRAULIC DESIGN

Available data indicate that the high-temperature thermal conductivity of thorium dioxide-uranium dioxide mixtures would be lower than that of the uranium dioxide used in the reference design. The possible influence of this property on the fuel thermal design must be evaluated carefully, although the general thermal behavior is not expected to change significantly. Detailed thermal-hydraulic design and its evaluation cannot be performed until a satisfactory nuclear design is established. No new computer code development is deemed necessary.

5.1.4 FUEL MANAGEMENT

Fuel-management information is summarized in Table 5-3. The fuel-cycle information presented in this table is based on a fuel-management scheme similar to that currently employed in PWRs, in which one-third of the core is replaced at annual refueling intervals. The yellowcake and separative-work requirements tabulated in Table 5-3 assume that plutonium has been stored for future use--that is, no credit has been taken for plutonium production.

The isotopic distribution of the fuel inventory is listed in Tables 5-4 and 5-5 for the beginning and end of the equilibrium cycle, respectively. The reactor charge data for a 30-year lifetime are given in Table 5-6 and the discharge data in Table 5-7.

The material-flow diagram for the high-burnup case is shown in Figure 5-1. The numerical identifiers in the fuel-cycle steps are correlated with the fuel-cycle descriptions of Volume VII.

The makeup uranium-233 is not found in nature; it must be created by irradiating thorium-bearing fuels. It can be created, for example, by using thorium fuel enriched with plutonium in converter reactors (light-water reactors, etc.) or by thorium-blanketed liquid-metal fast-breeder reactors. The design of reactors to produce uranium-233 is not discussed in this report; rather, it is assumed that uranium-233 is obtained from a stockpile produced by one of the previously mentioned options.

A "secure" fuel cycle facility is assumed to be an International Atomic Energy Agency-safeguarded facility with a maximum level of security by design and with guard force to prevent diversion of materials that are directly weapons-usable with comparatively little effort (e.g., highly enriched uranium and plutonium).

Table 5-1. Reactor system design and performance data (PWR with denatured uranium-233/thorium fuel)

Reactor type	PWR
Reactor thermal power output (gross), MW	3,817
Electrical power output, MW[a]	
Gross	1,344
Net	1,270
Plant heat rate, Btu/kW-hr	10,212
Core design and performance parameters	
Core heat output, MW	3,800
Core volume, liters	40,050
Core loading, kg[b]	
Heavy metal	93,550
Fissile fuel	2,430
Conversion ratio[c]	0.76
Average discharge exposure, MWd/MTHM[c,d]	33,390
Peak discharge exposure, MWd/MTHM[c,d]	42,750
Fuel type	Oxide
Reactor inlet temperature, °F	565
Reactor outlet temperature, °F	621
End-of-cycle excess reactivity	Zero (0)

[a]Depends on architect-engineer; these values assume mechanical-draft cooling.

[b]Initial denatured uranium-233/thorium core.

[c]Integrated conversion equilibrium cycle.

[d]Heavy-metal charged.

Table 5-2. Reactor design data specifications
(PWR with denatured uranium-233/thorium fuel)

Geometric information	
Core height, cm	381.0
Number of core enrichment zones (nominal)	3
Number of assemblies	241
Equivalent diameter, cm	365.8
Pins per assembly	236
Pin pitch-to-diameter ratio	1,325
Overall assembly length, cm	406.4
Lattice pitch, cm	1.288
Assembly material	Oxide fuel with Zircaloy-4 cladding
Cladding parameters	
Cladding outside diameter, mils	382.7
Cladding wall thickness, mils	25
Cladding material	Zircaloy-4
Fissile inventory at beginning of equilibrium cycle, kg	2,743
External fissile inventory, kg	NA
Fissile loss, kg/cycle[a]	386
Specific power, kW/kg fissile[b]	1,385
Power density, kW/kg HM	40.6

[a]Uranium-233 value given; in addition, 89 kg plutonium fissile sold.
[b]Beginning of equilibrium cycle.

Table 5-3. Fuel-management information
(PWR with denatured uranium-233/thorium fuel)

Average capacity factor, %	75	
Approximate fraction of core replaced annually	One-third	
Lag time assumed between fuel discharge and recycle reload, years	2	
Fissile material reprocessing loss fraction, %	1	
Fissile material fabrication loss fraction, %	1	
U_3O_8 and ThO_2 requirements, short tons/GWe	U_3O_8	ThO_2
Initial core	0	57.96
Annual equilibrium reload requirement	0	20.0
30-year cumulative requirement	0	573.32
Separative-work requirements, 10^3 SWU/GWe		
Initial core	0	
Equilibrium reload	0	
30-year cumulative requirement	0	
Requirements for special fuel materials (fissile U-233), kg HM/GWe		
Initial load	1,885	
Annual equilibrium charge/discharge	772/468	
30-year cumulative requirement	10,488 (net)	
Other data for proliferation-resistance assessment		
Fuel-element weight, kg	594	
Fresh and discharge fuel radiation level, R/hr at 1 meter	Not calculated	
Discharge fuel energy-generation rate after 90-day cooling (W-hr/element)	Not calculated	

Table 5-4. Fuel inventory at the beginning-of-equilibrium cycle

Isotope	Fuel inventory (kg)		
	Fresh fuel, Zone 1	Once-burnt fuel, Zone 2	Twice-burnt fuel, Zone 3
Th-232	22,044.76	21,856.33	21,670.17
Pa-233		24.37	26.23
U-232		10.95	11.30
U-233	980.01	771.17	648.02
U-234	268.41	282.63	289.22
U-235	69.78	71.26	73.20
U-236	32.65	35.66	38.56
U-238	7,784.04	7,691.39	7,599.11
Pu-238		0.25	0.90
Pu-239		49.46	64.61
Pu-240		8.48	17.44
Pu-241		3.82	12.01
Pu-242		0.32	2.24
Fission products		381.15	729.83
Other isotopes			
Am-241		0.040	0.221
Cm-242		--	--
Np-237		2.35	4.39

Table 5-5. Fuel inventory at the end-of-equilibrium cycle

Isotope	Fuel Inventory (kg)		
	Fresh fuel, Zone 1	Once-burnt fuel, Zone 2	Twice-burnt fuel, Zone 3
Th-232	21,856.33	21,670.16	21,480.55
Pa-233	24.37	26.23	28.03
U-232	10.95	11.30	11.60
U-233	771.17	648.02	565.69
U-234	282.63	289.22	291.41
U-235	71.28	73.20	75.33
U-236	35.66	38.56	41.39
U-238	7,691.39	7,599.11	7,503.77
Pu-238	.25	0.90	1.90
Pu-239	49.46	64.61	71.37
Pu-240	8.48	17.44	22.84
Pu-241	3.82	12.01	19.06
Pu-242	0.32	2.24	5.64
Fission products	381.15	729.83	1,058.44
Other isotopes			
Am-241	0.040	0.221	0.451
Cm-242	--	--	--
Np-237	2.35	4.39	6.26

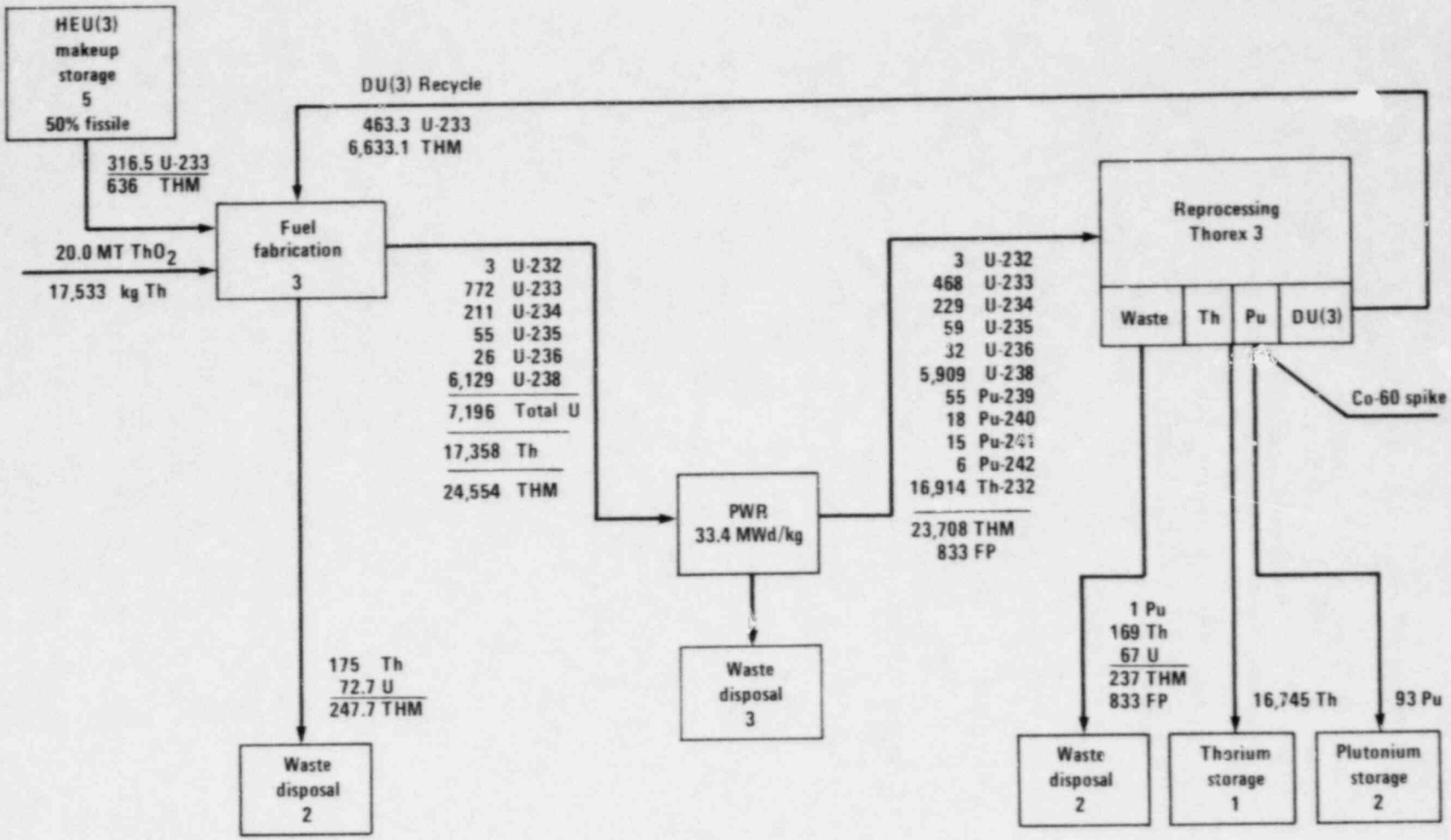
Table 5-6. Reactor charge data (kilograms) for zones 1, 2, 3, etc.

Year	Th-232	Pa-233	U-232	U-233	U-234	U-235	U-236	U-238	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	Np-237	Total heavy metal
1	73,602		3.7	2,394	0	35	0	17,527							93,558
2	23,214		1.7	957	0	14	0	7,007							31,186
3	23,214		1.7	957	0	14	0	7,007							31,186
4	22,891		2.7	973	96	21	3	7,185							31,186
5	22,891		2.7	973	96	21	3	7,185							31,186
6	22,891		2.7	973	96	21	3	7,185							31,186
7	22,891		2.7	973	96	21	3	7,185							31,186
8	22,891		2.7	973	96	21	3	7,185							31,186
9	22,745		3.2	975	161	37	9	7,255							31,186
10	22,745		3.2	975	161	37	9	7,255							31,186
11	22,745		3.2	975	161	37	9	7,255							31,186
12	22,745		3.2	975	161	37	9	7,255							31,186
13	22,745		3.2	975	161	37	9	7,255							31,186
14	22,604		3.5	978	208	51	15	7,323							31,186
15	22,604		3.5	978	208	51	15	7,323							31,186
16	22,604		3.5	978	208	51	15	7,323							31,186
17	22,604		3.5	978	208	51	15	7,323							31,186
18	22,604		3.5	978	208	51	15	7,323							31,186
19	22,505		3.6	977	242	61	24	7,346							31,186
20	22,505		3.6	977	242	61	24	7,346							31,186
21	22,505		3.6	977	242	61	24	7,346							31,186
22	22,505		3.6	977	242	61	24	7,346							31,186
23	22,505		3.6	977	242	61	24	7,346							31,186
24	22,045		3.7	980	268	70	33	7,784							31,186
25	22,045		3.7	980	268	70	33	7,784							31,186
26	22,045		3.7	980	268	70	33	7,784							31,186
27	22,045		3.7	980	268	70	33	7,784							31,186
28	22,045		3.7	980	268	70	33	7,784							31,186
29	22,045		3.7	980	268	70	33	7,784							31,186
30	22,045		3.7	980	268	70	33	7,784							31,186

Table 5-7. Reactor discharge data (kilograms) for zones 1, 2, 3, etc.[a]

Year	Th-232	Pa-233	U-232	U-233[b]	U-234	U-235	U-236	U-238	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	Np-237	Total heavy metal	Fission products
1	25,705		0.9	537	52	9	1	7,375		34	11	6	1		93,558	328
2	23,908		2.0	517	60	14	2	5,900		49	14	9	2		30,448	746
3	22,592		2.2	558	95	19	3	6,733		61	21	18	6		30,109	1,058
4	22,592		2.2	558	95	19	3	6,733		61	21	18	6		30,109	1,058
5	22,592		2.2	558	95	19	3	6,733		61	21	18	6		30,109	1,058
6	22,292		2.7	571	164	37	9	6,911		64	22	18	6		30,100	1,058
7	22,292		2.7	571	164	37	9	6,911		64	22	18	6		30,100	1,058
8	22,292		2.7	571	164	37	9	6,911		64	22	18	6		30,100	1,058
9	22,292		2.7	571	164	37	9	6,911		64	22	18	6		30,100	1,058
10	22,292		2.7	571	164	37	9	6,911		64	22	18	6		30,100	1,058
11	22,158		2.9	579	211	56	16	6,981		65	22	18	6		30,120	1,058
12	22,158		2.9	579	211	56	16	6,981		65	22	18	6		30,120	1,058
13	22,158		2.9	579	211	56	16	6,981		65	22	18	6		30,120	1,058
14	22,158		2.9	579	211	56	16	6,981		65	22	18	6		30,120	1,058
15	22,158		2.9	579	211	56	16	6,981		65	22	18	6		30,120	1,058
16	22,019		3.0	592	246	61	25	7,051		66	22	18	6		30,116	1,058
17	22,019		3.0	592	246	61	25	7,051		66	22	18	6		30,116	1,058
18	22,019		3.0	592	246	61	25	7,051		66	22	18	6		30,116	1,058
19	22,019		3.0	592	246	61	25	7,051		66	22	18	6		30,116	1,058
20	22,019		3.0	592	246	61	25	7,051		66	22	18	6		30,116	1,058
21	21,930		3.1	591	272	67	33	7,073		67	22	18	6		30,092	1,058
22	21,930		3.1	591	272	67	33	7,073		67	22	18	6		30,092	1,058
23	21,930		3.1	591	272	67	33	7,073		67	22	18	6		30,092	1,058
24	21,930		3.1	591	272	67	33	7,073		67	22	18	6		30,092	1,058
25	21,930		3.1	591	272	67	33	7,073		67	22	18	6		30,092	1,058
26	21,481		3.2	594	291	75	41	7,504		70	23	19	7		30,097	1,058
27	21,481		3.2	594	291	75	41	7,504		70	23	19	7		30,097	1,058
28	21,481		3.2	594	291	75	41	7,504		70	23	19	7		30,097	1,058
29	21,481		3.2	594	291	75	41	7,504		70	23	19	7		30,097	1,058
30	21,481		3.2	594	291	75	41	7,504		70	23	19	7		30,097	1,058

[a]Equilibrium discharge exposure is 33,400 MWd/MT.
[b]Pu-238 included.



Notes:

1. Mass flows are in kilograms per 0.75 GWe-yr.
2. Abbreviations: HEU, high-enrichment uranium-233; FP, fission products; DU(3), denatured uranium-233; THM, total heavy metal.

Figure 5–1. Mass flows for denatured uranium-thorium cycle, PWR DU(3)-Th, recycle DU(3).

5.2 SAFETY CONSIDERATIONS

Since the reactor plant employed in this evaluation of the denatured-uranium/thorium fuel cycle is the reference C-E System 80, it can be readily concluded that the concept is fundamentally licensable from a reactor-safety viewpoint. However, although preliminary evaluations of the characteristics of thorium-bearing cores (as in Reference 2) indicate that the response during postulated accidents is satisfactory, 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 uranium dioxide-fueled cores, and hence a thorough reevaluation of the performance of the thorium-fueled LWR for anticipated operational occurrence and other postulated accidents will be necessary. Consequently, it will be necessary to reanalyze the range of events typically reported in Chapter 15 of Reference 1 to demonstrate that the safety performance of the LWR fueled with denatured uranium/thorium falls within the criteria established for uranium dioxide-fueled operation. A qualitative evaluation of some of the key events that are postulated for safety analysis is given below.

In case of a loss-of-coolant accident (small or large LOCA), the consequences are not expected to be significantly different for uranium dioxide-based cores and thorium dioxide-based cores. System responses are primarily determined by the short-term decay-heat-removal requirements of the core. The physical properties of thorium dioxide are very similar to those of uranium dioxide, with the thermal conductivity being somewhat smaller than that for uranium dioxide cores but possibly less than that in plutonium-recycle cores, which have substantially higher heavy-metal inventories (i.e., americium and curium).

The steam-line-break accident can result in a rapid cooldown of the reactor-coolant system, which, as a result of the negative moderator-temperature coefficient, can cause reactor power to increase. The increase in reactor power results in an automatic trip of the control-element assemblies (CEAs) and the shutdown of the reactor. This accident is potentially slightly more severe in thorium cores than in equilibrium-cycle uranium dioxide cores because of the lower CEA worth throughout life.

The consequences of a CEA ejection accident for thorium dioxide cores are expected to be comparable to those for uranium dioxide cores. The ejected CEA worth and local power peaking in thorium cores and the equilibrium cycle core are comparable. For the thorium cores, the more negative Doppler coefficient and the longer prompt-neutron lifetime are advantageous, while the smaller delayed-neutron fraction is detrimental to the consequences of this accident. Although the severity of this accident cannot be quantitatively assessed without a more detailed analysis, the competing core characteristics appear to indicate comparable consequences.

The power mismatch and peaking factors resulting from the inadvertent loading of a fuel assembly in an improper position depend on the local fuel burnup and the fuel-management scheme. It is therefore difficult to make general conclusions on the severity of this incident as a result of the use of thorium fuel. Nevertheless, thorium-based fuels generally have lower early-in-life reactivity than do uranium-based fuels, and hence incorrect placement of thorium dioxide fuel may have less severe consequences than a similar incident involving uranium dioxide fuel.

The consequences of a steam-generator-tube rupture are not substantially different for thorium cores than for uranium cores. Radiological release as a result of the penetration of the barrier between the reactor-coolant system and the main steam system will

differ slightly because of the different fission-product yields for the various fissionable isotopes.

The consequences of a fuel-handling accident during refueling will not be significantly different for thorium cores and for uranium cores since it is assumed in the analysis that the dropped assembly is an irradiated assembly with appreciable fission-product buildup. The difference in fission-product buildup between the two fuel types should have only a minor effect on the severity of this accident.

Events in which CEAs malfunction include misaligned, stuck, or dropped CEAs. The limiting cases for this incident are those that result in the maximum reactivity addition to the core and/or the highest local power peaking. As for the CEA-withdrawal incident, the consequences of CEA malfunction would be comparable to those of the uranium dioxide core.

Uncontrolled boron dilution is defined as a decrease in reactor-coolant-system boron concentration due to the inadvertent addition of unborated water. The reactivity associated with this deboration incident depends on the initial soluble-boron concentration, the dilution rate, and the soluble-boron worth. Since the soluble-boron worth is comparable to that for the equilibrium uranium dioxide core, the reactivity addition rate to the core will be approximately the same for a given initial concentration and dilution rate, making the severity of the accident about the same as that for the uranium dioxide core.

For incidents that result in an increase in the reactor-coolant temperature (e.g., loss of load, loss of normal feedwater, or loss of AC power), thereby reducing the margin to departure from nucleate boiling, the primary concern is the rate of temperature increase in the primary system. The type of fuel in the core does not materially influence the system response. Thus, changes in the reactivity coefficients resulting from the use of thorium fuel do not change the consequences of these incidents to any significant degree. Furthermore, the slightly positive or small negative moderator temperature coefficient (expected when core spatial effects are included) for thorium cores at the beginning of the cycle is well within the conservative initial conditions of current safety analyses.

For incidents that result in a decrease in coolant temperature, specifically idle loop startup and excess load, a positive reactivity addition occurs when the moderator temperature coefficient is negative. Since the thorium cores appear to have somewhat less negative moderator-temperature coefficients, the consequences of these incidents will probably be less severe for the thorium cores than for the uranium cores.

The above qualitative discussion indicates that the consequences of postulated accidents for thorium cores are comparable to those typical of PWRs presently operating on the conventional uranium dioxide cycle.

5.3 ENVIRONMENTAL CONSIDERATIONS

This section addresses the environmental factors associated with normal operation of the denatured uranium-233/thorium cycle (hereafter referred to as the PWR thorium cycle). As discussed later in this section, the reactor core is assumed to be changed as required to accommodate this cycle. The reactor-coolant system, reactor auxiliaries, balance of plant, and site are assumed unchanged from the description given in Section 2.1 for the 30,000-MWd/MT case (reference cycle). Therefore, the following sections refer to the earlier sections where appropriate.

5.3.1 SUMMARY ASSESSMENT

The source terms for this system are estimated to be similar to those from the reference cycle. Therefore, the radiological impacts would also be similar. This conclusion is predicated on achieving fuel integrity similar to that of the reference case.

5.3.2 REACTOR AND STEAM-ELECTRIC SYSTEM

The information given in Section 2.3.2 for the reference cycle applies also to this cycle.

5.3.3 STATION LAND USE

The information given in Section 2.3.3 for the reference cycle applies also to this cycle.

5.3.4 STATION WATER USE

The information given in Section 2.3.4 for the reference cycle applies also to this cycle.

5.3.5 HEAT-DISSIPATION SYSTEM

The information given in Section 2.3.5 for the reference cycle applies also to this cycle.

5.3.6 RADWASTE SYSTEMS AND SOURCE TERMS

Sources of radioactivity, release paths, and processing systems are the same as for the reference cycle. The principal assumptions and plant parameters applicable to the reference cycle were given in Table 2-21. These data are also applicable to the PWR thorium cycle, with the possible exception of the operating-power fission-product source term (0.25%). This parameter is discussed below.

5.3.6.1 Source Term

The descriptive material in Section 2.3.6.1 is generally applicable to this cycle. That section describes the sources of radioactivity and the plant systems that determine the source term. As discussed therein, radioactivity released to the environment and contributing to occupational exposure originates from fission products, from the activation of core materials, and from the activation of coolant chemicals. Core structural materials are not significantly changed from the reference cycle. Coolant

chemistry is similar except that the cycle-average boron concentration in the coolant is expected to be several percent higher for the PWR thorium cycle than for the reference cycle.

The major question in connection with this cycle is the long-term performance of the thorium fuels in terms of fission-product release. As discussed earlier, thorium oxide has properties that are similar to those of uranium oxide. Thorium fuel was used in some early reactor cores, most notably the initial core of the Indian Point Unit 1 PWR. The major center of thorium-fuel development in recent years has been the light-water breeder reactor (LWBR) program. In this program, fuel-design methods and computer codes have been developed and small-scale thorium fuel irradiations conducted. The Shippingport reactor is now operating with the uranium-233/thorium LWBR core.

The introduction of uranium-233 and thorium into LWR plants would not be expected to result in a measurable change in the environmental impact of nuclear reactors. This was the stated conclusion in the final environmental impact statement for the LWBR program, which used these fuels. Without more detailed data that could be compared for the reference LWR and the uranium-233/thorium fuel cycles, the information from the LWBR program is cited as follows:

> . . . While there are slight differences in yields of the various fission product isotopes between uranium-233 and uranium-235, these differences are not large. No isotope is introduced by fissioning uranium-233 that is not produced by fissioning in a uranium-235 fueled reactor. The material properties and performance of uranium-233 and thorium fuel are similar to those of regular LWR fuels. Substantially all of the fission products would be retained within the fuel rod whether the fuel was uranium-233 or uranium-235.
>
> With fuel element defects in conventional LWR cores, some fission products would be released into the primary coolant. A portion of the gaseous fission products released to the coolant would ultimately be released to the atmosphere under controlled conditions. The situation would be expected to be normal for LWBR cores. The same primary coolant radioactivity upper limits which dictate changes in reactor plant operation would also apply to the uranium-233/thorium fuels. Neither the probability of accidents nor the effects would be significantly different as a result of using LWBR fuels. Therefore, the hazards to the public would be no worse for normal and accident conditions by the substitutions of LWBR fuel assemblies for conventional LWR fuel assemblies. Based on the detailed analysis (re uranium toxicity), it is concluded that under normal and accident conditions, the use of LWBR-type fuel would not have a more severe radiological impact on the environment than current LWR fuels.

For the PWR-thorium cycle, the release of fission products from the core and, therefore, to the environment, would be expected to be similar to that for the reference cycle, assuming equal fuel-failure rates. The krypton-85 yield is somewhat less for the thorium cycle, while the yields of other important isotopes (xenon-133, iodine-135, etc.) are about equal. However, because of important differences between core parameters for the LWBR core and the PWR core, fuel performance cannot be predicted with certainty. (The parameters that are significantly different include the average heat rate, coolant temperature, and burnup.)

The performance of uranium-233/thorium fuels will be determined only through extensive testing and plant operation in the future. Consequently, future research and development programs aimed at characterizing the behavior of uranium-233/thorium fuels under irradiation are required for the PWR thorium cycle as an extension of the research and development efforts already available from the LWBR program.

In summary, the source term from the PWR-thorium fuel would be similar to that for the reference core if fuel-failure rates are similar.

5.3.6.2 Liquid-Radwaste System

The equipment descriptions and flow diagrams in Section 2.3.6.2 are applicable also to the PWR thorium cycle. Because the sources of radioactivity would be similar, liquid releases would also be similar to those for the reference cycle (Section 2.3.6.2).

5.3.6.3 Gaseous-Radwaste Systems

The equipment descriptions and flow diagrams in Section 2.3.6.3 are applicable also to the PWR thorium cycle. Because the source term is similar, the gaseous releases would also be similar to those for the reference cycle.

5.3.6.4 Solid Radwastes

The data in Section 2.3.6.4 are applicable to the PWR thorium cycle.

5.3.6.5 Comparison with Predicted Releases from Other Studies

On the basis of the foregoing and assuming no change in the fuel-failure fraction from the reference cycle, the release of important nuclides would not change markedly. The major isotope contributing to whole-body and skin doses is xenon-133, and its release would be relatively unchanged as would the release of iodine-131, the major contributor to the thyroid dose. These nuclides would then be within the ranges defined in Tables 2-24 and 2-25.

5.3.7 CHEMICAL AND BIOCIDAL WASTES

The information presented in Section 2.3.7 is applicable to the PWR thorium cycle.

5.3.8 EFFECTS OF OPERATION OF HEAT-DISSIPATION SYSTEM

The information presented in Section 2.3.8 is applicable to the PWR thorium cycle.

5.3.9 RADIOLOGICAL IMPACT FROM ROUTINE OPERATION

Since the rates of release of radioactivity are estimated to be similar to those from the reference cycle, the radiological impact from routine operation would also be similar.

5.3.10 EFFECTS OF CHEMICAL AND BIOCIDAL DISCHARGES

The information presented in Section 2.3.10 is applicable to this cycle.

5.3.11 OCCUPATIONAL EXPOSURES

Occupational exposures would be increased somewhat for the thorium recycle case as compared to the reference cycle. Doses from operation and from radioactive waste handling would not be affected. Bred uranium-233 is accompanied by uranium-232 and its daughter products which emit penetrating gamma rays. Therefore, exposures chargeable to refueling would be increasd since fresh fuel would arrive in a shielded shipping cask. Additional man-hours and exposure would be incurred in handling the cask, removing the fuel, decontaminating the cask, and so on. The increase in occupational exposure would be only a very small percentage of the total annual occupational exposure.

5.4 LICENSING STATUS AND CONSIDERATIONS

Except for the composition of the fuel pellets, the reactor plant employed is the standard C-E System 80, which has undergone extensive licensing reviews. Although the nuclear characteristics and irradiation behavior of the fuel could be somewhat different from those of the reference uranium dioxide fuel, no significant change in the reactor performance during normal operation or accident conditions is expected. The outlook for licensing is therefore favorable. A thorough analysis of design-basis events would, however, have to be performed. Moreover, appropriate safety criteria, such as acceptable fuel-design limits and limits 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. For example, changes in fuel and moderator temperature reactivity coefficients, boron worth, control-rod worth, prompt-neutron lifetime, and the delayed-neutron fraction must be addressed since they can have a large impact on NSSS performance and safety. The effects of alternative fuel cycles on the dynamic responses of the system should be determined for all transients required by NRC Regulatory Guide 1.70. It will also be necessary to determine the implications of denatured-fuel cycles on plant operation and load-change performance, to determine whether the response of plant control and protection systems are altered.

Design of mechanisms and procedures for handling fuel assemblies must take into account the increased activity associated with uranium-232, which is invariably present in uranium-233.

5.5 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

As discussed in Section 5.1, the PWR nuclear steam supply system (NSSS) and balance of plant are identical with those already in commercial operation with uranium fuel in the United States and in many other countries. Consequently no basic reactor-development research and development are required.

However, technology for the utilization of thorium-bearing fuels and for the recycle of uranium-233 is much less well developed than is the technology of the uranium fuel cycle. The use of the denatured-uranium/thorium fuel cycle will therefore necessitate significant R&D effort in the areas of fuel fabrication, reprocessing, and for reactor-related data base and verification-type development.

The fabrication of uranium-233-bearing fuels is significantly different from the fabrication of uranium- or plutonium-bearing fuels because uranium-233 contains trace amounts of uranium-232, which is produced along with the fissile material uranium-233 from thorium fuels. Since the decay of uranium-232 leads to daughter products that emit highly energetic gamma rays, the fabrication of uranium-233-bearing fuels necessitates remote operations and shielded facilities. Although such remote and shielded facilities are easy to visualize conceptually, the fabrication process is complex and such facilities have yet to be engineered for reactor-grade uranium-233. One significant problem that must be addressed is the maintenance of such remote equipment, which must be quickly maintainable to avoid long downtimes for repair, as this would compromise the economics of the fabrication process. Because of the complexity of the pelletization process, it may be desirable to fabricate uranium-233-bearing fuels using VIPAC or SPHEREPAC technologies--technologies that appear more amenable to remote operations. The use of VIPAC or SPHEREPAC fabrication would, of course, necessitate additional research and development for process development and also for in-reactor performance qualifications, since neither of these alternative fabrication technologies is employed for the manufacture of commercial-grade fuels.

The denatured-uranium/thorium fuel cycle also introduces significant new requirements for fuel reprocessing and waste-treatment research and development. Reprocessing of thorium-based fuels is based on the Thorex process. Although this process has been demonstrated for lower radiation exposure fuel, it is much less developed than the Purex process utilized for reprocessing uranium-based fuels. Since spent denatured-thorium fuels will contain significant quantities of plutonium, as well as uranium and thorium, a modified version of the Thorex process will have to be developed and tested. Reprocessing of the thorium-based fuels is also complicated by the fact that, unlike uranium dioxide, thorium dioxide dissolves very slowly in nitric acid unless the fluoride ion is present. The introduction of fluoride complicates the treatment of waste from the fuel-dissolving process and will necessitate additional research and development in this area. The fluoride ion also complexes with the zirconium cladding so that thorium dioxide dissolution is severely retarded unless excess fluoride is added (which would severely increase equipment corrosion). A more acceptable approach may be to remove the cladding before dissolving the thorium dioxide by some chemical or mechanical means. Here again, additional research and development will be required both to develop the dissolution process itself and for the treatment of waste produced in this process.

Although there has been some experience with the irradiation of thorium-based fuels in LWRs, additional research and development will also be required, especially in the areas of data-base development and fuel-performance qualification. This effort

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

The fuel-cycle-related reactor research and development that must be performed for the thorium fuel cycles include data-base development, reactor-component development, and reactor/fuel-cycle demonstration. Such data-based development, particularly, consists of physics verification and the establishment of safety-related fuel-performance characteristics.

Another aspect of fuel-cycle-related research and development is the reactor/fuel-cycle demonstration. This demonstration includes the initial core-physics design and safety analysis, which identifies the changes in reactor design necessitated by the denatured-uranium/thorium fuel cycle and any resulting changes in design-basis events.

In a physics verification program, the first aspect requiring attention is the development of improved cross sections for thorium and for isotopes in the thorium depletion chains, such as uranium-233, uranium-234, and protactinium-233. Cross-sectional information for such elements has been largely neglected in the past and is believed to be much more uncertain than the corresponding cross sections of isotopes present in uranium-based fuel cycles. Resonance integral measurements must be performed for denatured fuels at room temperature and also at elevated temperatures. These experiments are very important in accurately calculating safety-related physics characteristics and also in establishing the quantities of plutonium produced during irradiation. The second aspect of the physics verification program consists of a series of critical experiments. Experiments should be performed for the fuel type under consideration (i.e., denatured uranium-233) and should preferably be performed both at room temperature and at elevated (moderator) temperatures. These experiments serve as a basis for demonstrating the ability of analytical models to predict such safety-related parameters as reactivity, power distributions, moderator temperature reactivity coefficients, boron worth, and control-rod worth.

Another major area of data-base development consists of the establishment of safety-related fuel-performance information such as transient fuel-damage limits, thermal performance for both normal operation and with respect to LOCA margins in stored heat, dimensional stability (densification and swelling), gas absorption and release behavior, and fuel-cladding interaction. Transient fuel-damage experiments are needed to provide information on the performance of denatured-thorium fuels under the more rapid transients possible during anticipated operational occurrences and other postulated accidents.

REFERENCES FOR CHAPTER 5

1. Combustion Engineering Inc., System 80--Preliminary Safety Analysis Report (PSAR)--CESSAR, Standard PWR-NSSS, Docket No. STN 50-470.

2. Electric Power Research Institute, Assessment of Thorium Fuel Cycles in Pressurized Water Reactors, EPRI NP-359, February 1977.

Chapter 6

PLUTONIUM/THORIUM CYCLE (Pu/ThO_2 BURNER), SPIKED RECYCLE

6.1 DESCRIPTION

The reactor/fuel-cycle combination described here is a pressurized water reactor (PWR) using pellet-type fuel. The spent fuel is reprocessed to recover and separate the uranium-233, plutonium, and thorium. The recovered uranium-233 is denatured to a 12% fissile content with depleted uranium and sent to a storage facility. The recovered plutonium, which is spiked with Co-60, is recycled to fuel fabrication, where it is mixed with makeup plutonium from a secure storage facility and with new thorium oxide. The thorium recovered during reprocessing is sent to an interim thorium storage facility, where it is allowed to decay for at least 10 years. Wastes from fuel fabrication and reprocessing are sent to a geologic waste repository.

An alternative method of fuel fabrication would utilize the Sphere-Pac technology. The status of the development of Sphere-Pac technology is not as advanced as pellet-fuel technology, and if this reactor/fuel-cycle combination were to be employed on a near-term basis, pellet-type fuel would probably be used. The Sphere-Pac process is introduced, however, because of potential gains in fabrication costs over the pellet process and the expectation that it will reach an adequate stage of development by the time this reactor/fuel-cycle combination is introduced.

The fuel-cycle facilities associated with this reactor/fuel-cycle combination are shown in the mass-flow diagram of Figure 6-1 and are discussed in the following sections of Volume VII:

Fuel fabrication 3	Chapter 4
Reprocessing (Thorex 3)	Section 5.5
Waste disposal 2	Section 7.2
Waste disposal 3	Section 7.3

To provide preliminary technical and economic data for the Nonproliferation Alternative Systems Assessment Program (NASAP), a specific plutonium/thorium cycle has been chosen, and its performance when incorporated into the reference Combustion Engineering, Inc. (C-E), System 80 PWR is assessed. (Similar performance could be obtained with other light-water reactor (LWR) designs.) A reference three-batch fuel-management scheme is used. All physical characteristics of the C-E System 80 (see Section 4.2.1 of Reference 1) have been retained in the core design with the exception of the composition of the fuel.

The generalized reactor-performance characteristics are summarized in Table 6-1, and the reactor-design data are summarized in Table 6-2. Additional data on fuel management are presented in Section 6.1.3.

The concept of burning plutonium with thorium has received only limited attention so far. Introduction of this mixed-oxide form of fuel in commercial PWRs would have to be preceded by significant research and development efforts to generate the necessary data base for nuclear design and licensing assessment. This is discussed further in Section 6.5. However, before any serious research and development program is launched,

some scoping studies are required to quantitatively assess the potential benefits of the plutonium/thorium fuel cycle.

A preliminary evaluation of the fuel cycle has been performed by Combustion Engineering under the sponsorship of the U.S. Department of Energy. The study is reported in References 2 and 3. It is shown that the characteristics of the plutonium/thorium burner are quite similar to those of the uranium/thorium reactors and that the former is a feasible alternative for deployment at secure nuclear energy centers.

No unique or fundamental constraint on the deployment of the plutonium/thorium fuel cycle is imposed by reprocessing or fabrication since these technologies appear to be relatively well developed and/or similar to those required in the uranium cycle; however, no commercial facilities for thorium reprocessing or fabrication currently exist or are planned. Irradiation performance of thorium-bearing fuels has been satisfactory and indicates no problems that might preclude thorium fueling.

6.1.1 FUEL MECHANICAL DESIGN

Except for the composition of the fuel, the fuel mechanical design is similar to that of the reference PWR design, as given in Section 4.2.1 of Reference 1. The fuel rods are made of thorium dioxide-plutonium dioxide dried gel microspheres packed by vibratory compaction. Insufficient data on the physical properties and irradiation behavior of the fuel are available to evaluate the mechanical fuel design.

6.1.2 FUEL NUCLEAR DESIGN

A detailed nuclear design of the core has not been performed. Nuclear-property data for plutonium/thorium are fairly well known, with the exception of interference effects due to overlap of the resources of the various isotopes, but these should not be large effects. When all data become available, the presently available computer codes can be employed (perhaps with some minor modifications) for the nuclear design of the core.

6.1.3 FUEL THERMAL-HYDRAULIC DESIGN

Available data indicate that the high-temperature thermal conductivity of thorium dioxide-plutonium dioxide mixtures would be lower than that of the uranium dioxide used in the reference design. The possible influence of this property on the fuel thermal design must be evaluated carefully, although the general thermal behavior is not expected to change significantly. Detailed thermal-hydraulic design and its evaluation cannot be performed until a satisfactory nuclear design is established. No new computer-code development is deemed necessary.

6.1.4 FUEL MANAGEMENT

Fuel-management information is summarized in Table 6-3. The fuel-cycle information presented in this table is based on a fuel-management scheme similar to that currently employed in PWRs, in which one-third of the core is replaced at annual refueling intervals

The isotopic distribution of the fuel inventory is listed in Tables 6-4 and 6-5 for the beginning and end, respectively, of the equilibrium cycle. The reactor charge data for a 30-year lifetime are given in Table 6-6; the discharge data are given in Table 6-7.

The mass-flow diagram is shown in Figure 6-1. The numerical identifiers in the fuel-cycle steps are correlated with the fuel-cycle descriptions of Volume VII.

The fuel-management characteristics (Ref. 2) of the initial three cycles for the plutonium/thorium and plutonium/uranium burners are compared in Table 6-8. The higher fissile loading requirements of the thorium burner are readily apparent. This is a consequence of the larger thermal absorption cross section of thorium-232 as compared with uranium-238 and the resulting lower reactivity of the thorium-based fuel for a given fissile enrichment. Since the first-cycle lifetime is much shorter than that of the uranium burner, a second cycle longer than that of the uranium burner results. A more satisfactory strategy is to employ a shorter first cycle so that the total energy produced in the first few cycles is comparable to that of the uranium burner.

Reference 2 also provides typical fuel-loading patterns and power distributions for the first three cycles of the uranium and thorium burners. The assembly-averaged peaking factors for the thorium cores are comparable to those for the uranium cores, indicating that fuel management will be no more limiting for the thorium burner than for the uranium burner. In fact, the smaller reactivity differences between the fresh and the burned thorium fuel make flux gradients less severe so that acceptable fuel-loading patterns are more easily obtained.

Table 6-1. Reactor design and performance data: pressurized-water reactor fueled with plutonium and thorium

Reactor gross thermal power output, MWt	3,817
Electrical power output, MWe[a]	
Gross	1,344
Net	1,270
Plant heat rate, Btu/kW-hr	10,212
Core design and performance parameters	
Core heat output, MWt	3,800
Core volume, liters	40,050
Core loading, kg[b]	
Heavy metal	93,519
Fissile fuel	3,129
Conversion ratio[c]	0.61
Average discharge burnup, MWd/MTHM[d,e]	33,400
Peak discharge burnup, MWd/MTHM[d,e]	42,763
Fuel type	Oxide
Reactor inlet temperature, °F	565
Reactor outlet temperature, °F	621
End-of-cycle excess reactivity	0

[a]Depends on architect-engineer; these values assume the use of mechanical draft cooling.
[b]Initial plutonium-thorium core, assuming no shims.
[c]Integrated conversion equilibrium cycle.
[d]Heavy metal charged.
[e]Equilibrium cycle.

Table 6-2. Reactor design specifications: pressurized-water reactor with plutonium-thorium core

Geometric information	
Core height, cm	381.0
Number of core enrichment zones (nominal)	3
Number of assemblies	241
Equivalent diameter, cm	365.8
Number of rods per assembly	236
Rod pitch-to-diameter ratio	1.325
Overall assembly length, cm	406.4
Lattice pitch, cm	1.288
Assembly material	Oxide fuel, Zircaloy-4 cladding
Cladding parameters	
Cladding outside diameter, mils	382.7
Cladding-wall thickness, mils	25
Cladding material	Zircaloy-4
Fissile inventory at beginning of equilibrium cycle, kg	5,726
Fissile loss, kg/cycle[a]	623
Specific power, kW/kg fissile	663.6
Power density, kW/kg HM	40.6

[a]Includes 1.5% losses on back-end material and 10% loss of plutonium-241 due to 2-year decay.

Table 6-3. Fuel-management information: pressurized water reactor fueled with plutonium and thorium

Average capacity factor, %	75	
Approximate fraction of core replaced annually	1/3	
Lag time between fuel discharge and recycle reload, yr	2	
Fissile material reprocessing loss fraction, %	1	
Fissile material fabrication loss fraction, %	1	
Thorium dioxide requirements, ST/GWe		
Initial core	70.15	
Annual equilibrium reload requirement	21.43	
30-year cumulative requirement	705.32	
Separative-work requirements, 10^3 SWU/GWe		
Initial core	0	
Equilibrium reload	0	
30-year cumulative requirement	0	
Requirements for special fuel materials, kg HM/GWe	Fissile Pu	U-233
Initial load	2,464	
Annual equilibrium charge/discharge	1,613/913	0/268
30-year cumulative requirement	23,720	8170

Table 6-4. Fuel inventory at beginning-of-equilibrium cycle

Isotope	Inventory, kg		
	Fresh fuel: Zone 1	Once-burnt fuel: Zone 2	Twice-burnt fuel: Zone 3
Th-232	30,312.70	27,052.14	26,894.62
Pa-233		20.40	21.37
U-232		0.12	0.48
U-233		121.97	230.37
U-234		4.82	12.69
U-235		0.27	1.31
U-236		0.003	0.029
Pu-238		1.17	5.80
Pu-239	1,548.40	1,064.32	807.83
Pu-240	1,543.59	1,312.11	1,229.79
Pu-241	734.45	625.99	591.08
Pu-242	589.33	521.93	515.54
Fission products		375.36	719.83
Other isotopes			
Am-241		21.88	33.24
Np-327		0.0005	0.0010

Table 6-5. Fuel inventory at end-of-equilibrium cycle

Isotope	Inventory, kg		
	Once-burnt fuel: Zone 1	Twice-burnt fuel: Zone 2	Thrice-burnt fuel: Zone 3
Th-232	27,052.14	26,894.62	26,736.32
Pa-233	20.40	21.37	22.67
U-232	0.12	0.48	0.95
U-233	121.97	230.37	317.80
U-234	4.82	12.69	22.49
U-235	0.27	1.31	3.26
U-236	0.003	0.029	0.108
Pu-238	1.17	5.80	13.06
Pu-239	1,064.32	807.83	603.42
Pu-240	1,312.11	1,229.79	1,136.12
Pu-241	625.99	591.08	556.57
Pu-242	521.93	515.54	508.46
Fission products	375.36	719.83	1,045.87
Other isotopes			
Am-241	21.88	33.24	38.37
Np-327	0.00005	0.0010	0.0053

Table 6-6. Reactor charge data

Year	Quantity (kg)					
	Th-232	Pu-239	Pu-240	Pu-241	Pu-242	Total
1	89,091	2,552	1,071	577	226	93,519
2	29,065	1,141	479	258	102	31,045
3	28,948	1,209	507	274	108	31,046
4	28,468	1,227	821	444	223	31,183
5	28,468	1,227	821	444	223	31,183
6	28,468	1,227	821	444	223	31,183
7	28,468	1,227	821	444	223	31,183
8	28,468	1,227	821	444	223	31,183
9	28,030	1,265	1,022	536	324	31,177
10	38,030	1,265	1,022	536	324	31,177
11	28,030	1,265	1,022	536	324	31,177
12	38,030	1,265	1,022	536	324	31,177
13	38,030	1,265	1,022	536	324	31,177
14	27,697	1,313	1,167	594	406	31,177
15	27,697	1,313	1,167	594	406	31,177
16	27,697	1,313	1,167	594	406	31,177
17	27,697	1,313	1,167	594	406	31,177
18	27,697	1,313	1,167	594	406	31,177
19	27,433	1,356	1,285	630	472	31,176
20	27,433	1,356	1,285	630	472	31,176
21	27,433	1,356	1,285	630	472	31,176
22	27,433	1,356	1,285	630	472	31,176
23	27,433	1,356	1,285	630	472	31,176
24	27,217	1,390	1,386	659	529	31,181
25	27,217	1,390	1,386	659	529	31,181
26	27,217	1,390	1,386	659	529	31,181
27	27,217	1,390	1,386	659	529	31,181
28	27,217	1,390	1,386	659	529	31,181
29	27,217	1,390	1,386	659	529	31,181
30	27,217	1,390	1,386	659	529	31,181

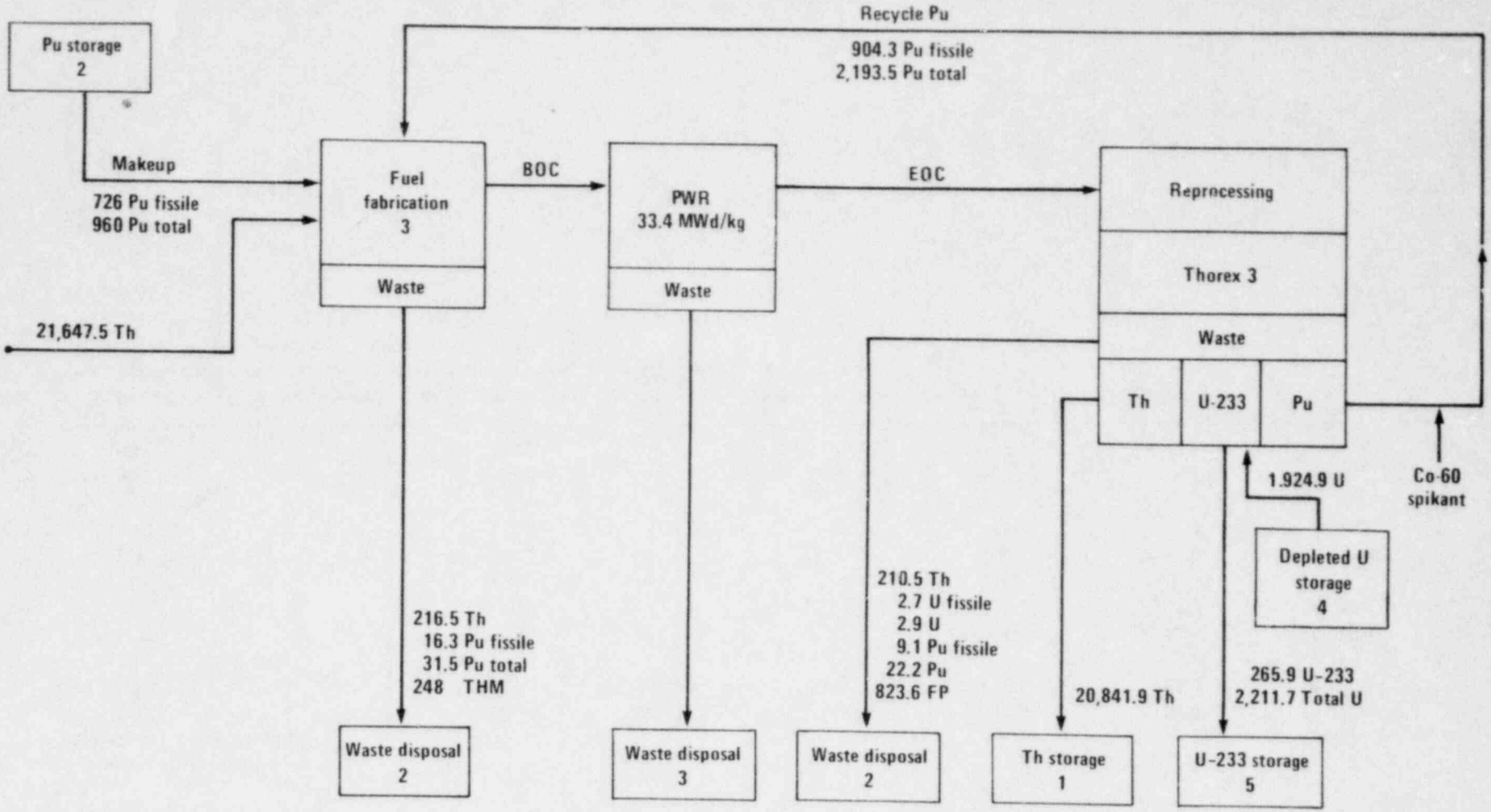
Table 6-7. Reactor discharge data

Year	Quantity (kg)												
	Th-232	Pa-233	U-232	U-233	U-234	U-235	Pu-238	Pu-239	Pu-240	Pu-241	Pu-242	Total	Fission products
1	30,134.0	0.0	0.3	197.0	9.8	0.7	0.0	274.0	247.3	146.9	67.2	31.076.8	328
2	29,252.5	0.0	0.9	360.4	20.4	2.3	0.0	218.4	295.3	176.8	94.9	30,422.0	740
3	28,507.2	0.0	1.6	361.5	28.8	4.3	0.0	278.4	391.3	241.7	132.4	29,947.2	1,046
4	28,497.2	0.0	1.6	370.0	29.9	4.5	0.0	289.6	428.9	258.2	140.1	30,020.0	1,046
5	28,497.2	0.0	1.6	370.0	29.9	4.5	0.0	289.6	428.9	258.2	140.1	30,020.0	1,046
6	27,938.0	0.0	1.6	359.2	26.8	4.0	0.0	358.3	650.6	373.1	244.2	29,955.8	1,046
7	27,938.0	0.0	1.6	359.2	26.8	4.0	0.0	358.3	650.6	373.1	244.2	29,955.8	1,046
8	27,938.0	0.0	1.6	359.2	26.8	4.0	0.0	358.3	650.6	373.1	244.2	29,955.8	1,046
9	27,938.0	0.0	1.6	359.2	26.8	4.0	0.0	358.3	650.6	373.1	244.2	29,955.8	1,046
10	27,938.0	0.0	1.6	359.2	26.8	4.0	0.0	358.3	650.6	373.1	244.2	29,955.8	1,046
11	27,517.6	0.0	1.6	353.1	25.2	3.7	0.0	456.9	813.3	447.4	332.4	29,951.2	1,046
12	27,517.6	0.0	1.6	353.1	25.2	3.7	0.0	456.9	813.3	447.4	332.4	29,951.2	1,046
13	27,517.6	0.0	1.6	353.1	25.2	3.7	0.0	456.9	813.3	447.4	332.4	29,951.2	1,046
14	27,517.6	0.0	1.6	353.1	25.2	3.7	0.0	456.9	813.3	447.4	332.4	29,951.2	1,046
15	27,517.6	0.0	1.6	353.1	25.2	3.7	0.0	456.9	813.3	447.4	332.4	29,951.2	1,046
16	27,200.3	0.0	1.6	347.2	23.9	3.5	0.0	516.3	940.6	494.1	402.2	29,929.7	1,046
17	27,200.3	0.0	1.6	347.2	23.9	3.5	0.0	516.3	940.6	494.1	402.2	29,929.7	1,046
18	27,200.3	0.0	1.6	347.2	23.9	3.5	0.0	516.3	940.6	494.1	402.2	29,929.7	1,046
19	27,200.3	0.0	1.6	347.2	23.9	3.5	0.0	516.3	940.6	494.1	402.2	29,929.7	1,046
20	27,200.3	0.0	1.6	347.2	23.9	3.5	0.0	516.3	940.6	494.1	402.2	29,929.7	1,046
21	26,946.4	0.0	1.6	343.0	23.0	3.3	0.0	564.1	1,045.7	528.1	458.7	29,913.9	1,046
22	26,946.4	0.0	1.6	343.0	23.0	3.3	0.0	564.1	1,045.7	528.1	458.7	29,913.9	1,046
23	26,946.4	0.0	1.6	343.0	23.0	3.3	0.0	564.1	1,045.7	528.1	458.7	29,913.9	1,046
24	26,946.4	0.0	1.6	343.0	23.0	3.3	0.0	564.1	1,045.7	528.1	458.7	29,913.9	1,046
25	26,946.4	0.0	1.6	343.0	23.0	3.3	0.0	564.1	1,045.7	528.1	458.7	29,913.9	1,046
26	26,736.3	0.0	1.6	340.5	22.5	3.3	0.0	603.4	1,145.3	556.6	508.4	29,917.9	1,046
27	26,736.3	0.0	1.6	340.5	22.5	3.3	0.0	603.4	1,145.3	556.6	508.4	29,917.9	1,046
28	26,736.3	0.0	1.6	340.5	22.5	3.3	0.0	603.4	1,145.3	556.6	508.4	29,917.9	1,046
29	26,736.3	0.0	1.6	340.5	22.5	3.3	0.0	603.4	1,145.3	556.6	508.4	29,917.9	1,046
30	26,736.3	0.0	1.6	340.5	22.5	3.3	0.0	603.4	1,145.3	556.6	508.4	29,917.9	1,046
30	26,894.6	21.4	0.5	230.4	12.7	1.4	5.8	807.8	1,229.8	519.1	516.1	30,311.0	
30	27,052.1	20.4	0.1	122.0	4.8	0.3	1.2	1,064.3	1,312.1	626.0	522.1	30,725.3	

Table 6-8. Fuel-management characteristics of initial burner cycles for a 1,300-MWe PWR[a]

Parameter		First cycle		Second cycle		Third cycle	
		UO_2	ThO_2	UO_2	ThO_2	UO_2	ThO_2
Cycle length (full-power days)		371	267	213	228	270	290
Average makeup plutonium fissile enrichment, wt% Pu fissile	Zone 1	1.25	2.36				
	Zone 2	1.75	3.14	3.03	4.49	3.25	4.79
	Zone 3	2.85	4.60				
Core inventory, kg[b]							
Total plutonium, kg							
BOC		2,884	4,429	3,484	4,696	4,074	5,220
EOC		2,582	3,444	3,243	3,910	3,660	4,181
Fissile plutonium							
BOC[c]		2,038	3,130	2,331	3,098	2,734	3,422
EOC		1,650	2,116	2,064	2,337	2,372	2,435
Uranium-233							
BOC[d]		0	0	0	294	0	337
EOC		0	425	0	576	0	684
Uranium-235							
BOC		711	0	572	1	587	2
EOC		480	2	472	4	475	6

[a]Results from coarse-mesh (16 nodes per assembly), two-dimensional PDQ calculations.
[b]Abbreviations: BOC, beginning of cycle; EOC, end of cycle.
[c]Includes decay of Np-239 during shutdown.
[d]Includes decay of Pa-233 during 30-day shutdown.



Notes:

1. Mass flows in kg per 0.75 GWe-yr.
2. Data base: Addendum to NASAP PSEID Vol. 1 by Argonne National Laboratory, March 8, 1979. Data normalized from a 1,270-MWe reactor; beginning of cycle = year 25; end of cycle = year 30.
3. Abbreviations: BOC, beginning of cycle; EOC, end of cycle; FP, fission products; THM, total heavy metal.

	BOC	EOC
Thorium	21,431	21,052.4
U-233	–	268.1
Total U	–	289.7
Pu fissile	1,614	913.4
Pu total	3,122	2,215.7
THM	24,553	23,557.8
FP	–	823.6

Figure 6-1. Mass flows for the LWR plutonium/thorium, plutonium spiked recycle, fuel cycle.

6.2 SAFETY CONSIDERATIONS

Since the reactor plant employed in this evaluation of the plutonium/thorium fuel cycle is the reference C-E System 80, it can be readily concluded that the concept is fundamentally licensable from a reactor-safety viewpoint. Although preliminary evaluations of the characteristics of thorium-bearing cores (as in Refs. 2 and 4) indicate a satisfactory response during postulated accidents, 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 uranium dioxide-fueled cores. A thorough reevaluation of the performance of the thorium-fueled LWR for anticipated operational occurrences and other postulated accidents will therefore be necessary. Consequently, it will be necessary to reanalyze the range of events typically reported in Chapter 15 of Reference 1 to demonstrate that the safety performance of the LWR fueled with plutonium/thorium meets the criteria established for uranium dioxide-fueled operation.

The core-physics parameters examined in Reference 2 include control-rod and soluble-boron worths, moderator and fuel temperature coefficients, and delayed-neutron fractions and prompt-neutron lifetimes. These parameters form the basis for the safety evaluation of various postulated accidents and plant transients. A comparison of the physics parameters for a thorium/plutonium burner and for a uranium/plutonium burner is given in Table 6-9.

In general, the core-physics parameters for the two reactors compared in Table 6-9 are quite similar, indicating comparable behavior in postulated accidents and plant transients. Nevertheless, the following differences are noted: the effective delayed-neutron fraction (effective beta value) and the prompt-neutron lifetime are smaller for the thorium burner. These are the controlling parameters in the reactor's response to short-term (on the order of seconds) power transients. However, the most limiting accident for this type of transient is usually the rod-ejection accident, and, since the control-rod worth is lower for the thorium burner, the consequences of the smaller values of these kinetics parameters are largely mitigated.

The moderator and fuel temperature coefficients are parameters that affect the inherent safety of the core. In the power operating range, the combined responses of these reactivity feedback mechanisms to an increase in reactor thermal power must be a decrease in core reactivity. Since both coefficients are negative, this requirement is easily satisfied. The fuel temperature coefficient is about 25% more negative for the thorium burner, and the moderator temperature coefficient is approximately 20% less negative. Since these differences largely equalize each other, the consequences of accidents that involve a core temperature transient would be comparable. For some accidents, however, individual temperature coefficients are the controlling parameters, and for these cases the consequences must be evaluated on a case-by-case basis.

Control-rod and soluble-boron worths are strongly dependent on the thermal-neutron diffusion length. Because of the larger thermal absorption cross section of thorium-232 and the higher plutonium loadings of the thorium burner, the diffusion length and, consequently, the control-rod and soluble-boron worths are smaller. Of primary concern is the maintenance of an adequate shutdown margin to compensate for the reactivity defects during postulated accidents--for example, for the reactivity increase associated with moderator cooldown in the steam-line-break accident. Analyses of individual accidents of this type are beyond the scope of this report but would have

to be performed to assess fully the consequences of the 10% reduction in control-rod worth at the beginning of the cycle.

The consequences of a fuel-handling accident during refueling will not differ significantly for thorium cores and for uranium cores since it is assumed in the analysis that the dropped assembly is irradiated, with appreciable fission-product buildup. The difference in fission-product buildup between the two fuel types should have only a minor effect on the severity of this accident.

Events in which control element assemblies (CEAs) malfunction include misaligned, stuck, or dropped CEAs. The limiting cases for this incident are those that result in the maximum reactivity addition to the core and/or the highest local power peaking. As for the CEA-withdrawal incident, the consequences of CEA malfunction would be comparable to those of the uranium dioxide core.

The above qualitative discussion indicates that the consequences of postulated accidents for the thorium burner are comparable to those of the uranium burner. Furthermore, this comparison indicates that other than the possibility of requiring additional control rods, a thorium-based plutonium burner is feasible and no modifications are required to a PWR already designed to accommodate an all-plutonium core.

Table 6-9. Safety-related core-physics parameters (third cycle)

Parameter[a]	Core type	
	Uranium/plutonium	Thorium/plutonium
Effective delayed-neutron fraction		
BOC	0.00430	0.00344
EOC	0.00438	0.00367
Prompt-neutron lifetime, 10^{-6} sec		
BOC	10.54	9.03
EOC	12.53	11.30
Inverse soluble boron worth, ppm/%$\Delta\rho$		
BOC	221	270
EOC	180	217
Fuel temperature coefficient, $10^{-5}\,\Delta\rho/^{o}F$		
BOC	-1.13	-1.40
EOC	-1.15	-1.42
Moderator temperature coefficient, $10^{-4}\,\Delta\rho/^{o}F$		
BOC	-1.65	-1.31
EOC	-3.32	-2.60
Control-rod worth, % of UO_2 burner		
BOC	--	90
EOC	--	96

[a]Abbreviations: BOC, beginning of cycle; EOC, end of cycle.

6.3 ENVIRONMENTAL CONSIDERATIONS

The nonradiological environmental effects of the plutonium/thorium burner--such as land use, water use, and chemical and biocidal discharges--would be similar to those of the reference LWR since all the characteristics of the plant would be the same except the composition and shape of the fuel pellets in the core.

The radiological environmental effects would also be similar. The radioactivity release paths and radioactive-waste-processing systems would be identical with those of the reference LWR. Slight variations, however, may exist between the quantities of specific radioactive isotopes released from the reference LWR and the quantities that would be released from the conceptual plutonium/thorium burner plant. This is mainly due to differences in fuel composition and a probable corresponding slight shift in the amounts of different fission products as a function of mass number.

As pointed out in Sections 6.1.1 and 6.1.2, insufficient data exist on the mechanical, nuclear, and thermal-hydraulic behavior of the fuel; thus, a proper evaluation of the radioactive release rates cannot be made at this time. It is assumed, therefore, that the fission-product release rates would be comparable to those of current LWRs.

Radiation exposure to plant workers is related to plant design and is not greatly affected by the installed core. Most of the exposure would be incurred in maintenance, repair, reactor operation, waste processing, and refueling operations, which would be almost identical with those of the reference LWR. Therefore, occupational exposure in the conceptual plutonium/thorium LWR would be about the same as in the reference LWR.

6.4 LICENSING STATUS AND CONSIDERATIONS

Except for the composition of the fuel, the reactor plant employed is the standard C-E System 80, which has undergone extensive licensing reviews. Although the nuclear characteristics and irradiation behavior of the fuel could be somewhat different from those of the reference uranium dioxide fuel, no significant change in the reactor performance during normal operation or accident conditions is expected. The outlook for licensing is therefore favorable. However, thorough analysis of design-basis events would have to be performed. Moreover, appropriate safety criteria, such as acceptable fuel-design limits and limits 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. For example, changes in fuel and moderator temperature reactivity coefficients, boron worth, control-rod worth, prompt-neutron lifetime, and delayed-neutron fraction must be addressed since they can have a large impact on the performance and safety of the nuclear steam supply system. The effects of alternative fuel cycles on the dynamic responses of the system should be determined for all transients required by NRC Regulatory Guide 1.70. It will be necessary to determine the implications of the new fuel cycle on plant operation and load-change performance, to establish whether the response of plant control and protection systems is altered.

In addition, the licensing acceptability of the alternative Sphere-Pac fuel for full-scale application has to be established. Likely concerns about Sphere-Pac fuel are (1) fuel-rod failure statistics compared to pellet fuel and (2) fission-gas release compared to pellet fuel. There are virtually no data available now.

6.5 RESEARCH, DEVELOPMENT, AND DEMONSTRATION

As discussed in Section 6.1.1, the nuclear steam supply system and balance of plant for this type of reactor are identical with those already in commercial operation with uranium fuel in the United States and in many other countries. Consequently no basic reactor-development research and development is required.

Technology for the utilization of thorium-bearing fuels, however, is much less well developed than is the technology of the uranium fuel cycle. The use of the plutonium/thorium fuel cycle would therefore necessitate significant research and development efforts in the areas of fuel fabrication, reprocessing, development of a reactor-related data base, and verification-type development.

The use of Sphere-Pac fabrication would require additional research and development for process development and also for in-reactor performance qualifications, since this alternative fabrication technology is not currently employed for the manufacture of commercial-grade fuels.

The plutonium/thorium fuel cycle also introduces significant new requirements for fuel reprocessing and waste-treatment research and development. Reprocessing of thorium-based fuels is based on the Thorex process. Although this process has been demonstrated for lower radiation exposure fuel, it is much less developed than the Purex process used for reprocessing uranium-based fuels. Since spent thorium fuels will contain significant quantities of plutonium, as well as uranium and thorium, a modified version of the Thorex process will have to be developed and tested. Reprocessing of the thorium-based fuels is also complicated by the fact that, unlike uranium dioxide, thorium dioxide dissolves very slowly in nitric acid unless the fluoride ion is present. The introduction of fluoride complicates the treatment of waste from the fuel-dissolving process and will necessitate additional research and development in this area. Furthermore, the fluoride ion complexes with the zirconium from the cladding; thorium dioxide dissolution is therefore severely retarded unless excess fluoride is added (which would greatly increase equipment corrosion). A more acceptable approach may be to remove the cladding, by some chemical or mechanical means, before dissolving the thorium dioxide. Here again, additional research and development will be required both to develop the dissolution process itself and for the treatment of waste produced in this process.

Although there has been some experience with the irradiation of thorium-based fuels in LWRs, additional research and development will also be required, especially in the areas of data-base development and fuel-performance qualification. This effort is necessary to insure that fuel performance meets licensing requirements and to develop the information required for licensing thorium-fueled cores. Such information as in-reactor densification and swelling behavior, fission-gas release, thermal conductivity of the fuel, and coefficients of reactivity must be established. Also the performance of Sphere-Pac fuel, compared to pellet fuel, has to be demonstrated in terms of fuel-rod failure statistics, fission-gas release, possible hydriding failure, fuel-clad interaction behavior, possible fuel relocation at moderate to high burnup, post-failure release of fission products, and behavior of the fuel containing varying concentrations of gadolinium oxide as a burnable poison. Subsequent research and development would consist of in-reactor irradiation demonstrations where significant quantities of thorium-based fuels, fabricated with processes and equipment representative of commercial fabrication technology, would be irradiated to provide a demonstration of in-reactor fuel performance.

The fuel-cycle-related reactor research and development that must be performed for the thorium fuel cycles include data-base development, reactor-component development, and reactor/fuel-cycle demonstration. Such data-based development consists of physics verification and the establishment of safety-related fuel-performance characteristics.

Another aspect of fuel-cycle-related research and development is the reactor/fuel-cycle demonstration. This demonstration includes the initial core-physics design and safety analysis, which identifies the changes in reactor design necessitated by the plutonium/thorium fuel cycle and any resulting changes in design-basis events.

In a physics verification program, the first aspect requiring attention is the development of improved cross sections for thorium and for isotopes in the thorium-depletion chains, such as uranium-233, uranium-234, and protactinium-233. Cross-sectional information for such elements has been largely neglected in the past and is believed to be much more uncertain than the corresponding cross sections of isotopes present in uranium-based fuel cycles. Resonance integral measurements must be performed at room temperature and also at elevated temperatures. These experiments are very important for the accurate calculation of safety-related physics characteristics and also in establishing the quantities of plutonium consumed during irradiation.

The second aspect of the physics verification program consists of a series of critical experiments. Experiments should be performed for the fuel type under consideration and preferably should be performed both at room temperature and at elevated (moderator) temperatures. These experiments serve as a basis for demonstrating the ability of analytical models to predict such safety-related parameters as reactivity, power distributions, moderator temperature reactivity coefficients, boron worth, and control-rod worth.

Another major area of data-base development consists of the establishment of safety-related fuel-performance information such as transient fuel-damage limits, thermal performance for both normal operation and with respect to loss-of-coolant accident margins in stored heat, dimensional stability (densification and swelling), gas absorption and release behavior, and fuel-cladding interaction. Transient fuel-damage experiments are needed to provide information on the performance of plutonium/thorium fuels under the more rapid transients possible during anticipated operational occurrences and other postulated accidents.

REFERENCES FOR CHAPTER 6

1. Combustion Engineering, Inc., System 80--Preliminary Safety Analysis Report (PSAR)--CESSAR, Standard PWR-NSSS, NRC Docket No. STN 50-470.

2. R. A. Matzie et al., An Evaluation of Denatured Thorium Fuel Cycles in Pressurized Water Reactors, TIS-5161, Combustion Engineering, Inc.

3. Assessment of PWR Plutonium Burners for Nuclear Energy Centers, COO-2786-1, UC-78, June 1976.

4. Electric Power Research Institute, Assessment of Thorium Fuel Cycles in Pressurized Water Reactors, EPRI NP-359, February 1977.

APPENDIX A

U.S. Nuclear Regulatory Commission Review of Safeguards Systems for the Nonproliferation Alternative Systems Assessment Program Alternative Fuel-Cycle Materials

BACKGROUND

The procedures and criteria for the issuance of domestic licenses for possession, use, transport, import, and export of special nuclear material are defined in 10 CFR 70, which also includes requirements for nuclear material control and accounting. Requirements for the physical protection of plants and special nuclear materials are described in 10 CFR 73, including protection at domestic fixed sites and in transit against attack, acts of sabotage, and theft. The U.S. Nuclear Regulatory Commission (NRC) has considered whether strengthened physical protection may be required as a matter of prudence (Ref. 1). Proposed upgraded regulatory requirements to 10 CFR 73 have been published for comment in the Federal Register (43 FR 35321). A reference system described in the proposed upgraded rules is considered as but one representative approach for meeting upgraded regulatory requirements. Other systems might be designed to meet safeguards performance criteria for a particular site.

NONPROLIFERATION ALTERNATIVE SYSTEMS ASSESSMENT PROGRAM SAFEGUARDS BASIS

The desired basis for the NRC review of safeguards systems for the Nonproliferation Alternative Systems Assessment Program (NASAP) alternative fuel-cycle materials containing significant quantities of strategic special nuclear material (SSNM),[a] greater than 5 formula kilograms,[b] during domestic use, transport, import, and export to the port of entry of a foreign country is the reference system described in the current regulations and the proposed revisions cited above. The final version of the proposed physical protection upgrade rule for Category I[c] material is scheduled for Commission review and consideration in mid-April. This proposed rule is close to being published in effective form and, together with existing regulations, will provide a sound basis for identification of possible licensing issues associated with NASAP alternative fuel cycles. This regulatory base should be applied to evaluate the relative effectiveness of a spectrum of safeguards approaches (added physical protection, improved material control and accounting, etc.) to enhance safeguards for fuel material types ranging from unadulterated to those to which radioactivity has been added.

To maintain safeguards protection beyond the port of entry into a country whose safeguards system is not subject to U.S. authority, and where diversion by national or subnational forces may occur, proposals have been made to increase radioactivity of strategic special nuclear materials (SSNMs) that are employed in NASAP alternative fuel cycles. Sufficient radioactivity would be added to the fresh-fuel material to require that, during the period after export from the United States and loading into the foreign reactor, remote reprocessing through the decontamination step would be necessary to recover low-radioactivity SSNM from diverted fuel. It is believed that with sufficient radioactivity to require remote reprocessing, the difficulty and time required in obtaining material for weapons purposes by a foreign country would be essentially the same as for spent fuel. In addition, the institutional requirements imposed by the Nuclear Non-Proliferation Act of 1978 include application of International Atomic Energy Authority (IAEA) material accountability

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[a] ≥20% U-235 in uranium, ≥12% U-233 in uranium, or plutonium.

[b] Formula grams = (grams contained U-235) + 2.5 (grams U-233 + grams plutonium); Ref. 10 CFR 73.30.

[c] IAEA definitions of highly enriched uranium (>20%).

requirements to nuclear-related exports. A proposed additional institutional requirement would be that verification of fuel loading into a reactor would be necessary by the IAEA prior to approval of a subsequent fuel export containing SSNM.

Another proposed alternative that could be used to provide additional safeguards protection against diversion of shipments of SSNM by subnational groups would be to mechanically attach and lock in place a highly radioactive sleeve over the SSNM container or fuel assembly.

NRC REVIEW

It is requested that NRC perform an evaluation of a spectrum of safeguards measures and deterrents that could be utilized to protect the candidate alternative fuel cycles. For the fuel cycles under review, consideration should be given to both unadulterated fuel materials and those to which added radioactive material purposely has been added. The relative effectiveness of various safeguards approaches (such as upgraded physical protection, improved material control and accountancy, dilution of SSNM, decreased transportation requirements, few sites handling SSNM, and increased material-handling requirements as applied to each fuel material type) should be assessed. The evaluation should consider, but not be limited to, such issues as the degree to which added radioactive contaminants provide protection against theft for bomb-making purposes; the relative impacts on domestic and on international safeguards; the impact of radioactive contaminants on detection for material control and accountability, measurement, and accuracy; the availability and process requirements of such contaminants; the vulnerability of radioactive sleeves to tampering or breaching; the increased public exposure to health and safety risk from acts of sabotage; and the increased radiation exposure to plant and transport personnel. Finally, in conducting these assessments, the NRC must consider the export and import of SSNM as well as its domestic use.

As part of this evaluation, we request that the NRC assess the differences in the licensing requirements for the domestic facilities, transportation systems to the port of entry of the importer, and other export regulations for those unadulterated and adulterated fuel-cycle materials having associated radioactivity as compared to SSNM that does not have added radioactivity. The potential impacts of added radioactivity on U.S. domestic safeguards, and on the international and national safeguards systems of typical importers for protecting exported sensitive fuel cycle materials from diversion should be specifically addressed. Aspects which could adversely affect safeguards, such as more limited access for inspection and degraded material accountability, as well as the potential advantages in detection or deterrence should be described in detail. The potential role, if any, that added radioactivity could or should play should be clearly identified, particularly with regard to its cost effectiveness in comparison with other available techniques, and with consideration of the view that the radioactivity in spent fuel is an important barrier to its acquisition by foreign countries for weapons purposes. Licensability issues that must be addressed by research, development, and demonstration programs also should be identified.

Table A-1 presents a listing of unadulterated fuel materials and a candidate set of associated radiation levels for each that should be evaluated in terms of domestic use, import, and export:

Table A-1. Minimum radiation levels for various fuel material types

Fuel Material Type	Minimum radiation level during 2-year period, rem/hr at 1 meter (Ref. 6)	
	Mixed[a]	Mechanically attached[b]
PuO_2, $HEUO_2$ powder or pellets[c]	1,000/kgHM	10,000/kgHM
PUO_2-UO_2 and $HEUO_2$-ThO_2 powder or pellets[c]	100/kgHM	10,000/kgHM
LWR, LWBR, or HTGR recycle fuel assembly (including type b fuels)	10/assembly	1,000/assembly
LMFBR or GCFR fuel assembly (including type b fuels)	10/assembly	1,000/assembly

[a]Radioactivity intimately mixed in the fuel powder or in each fuel pellet.

[b]Mechanically attached sleeve containing Co-60 is fitted over the material container or fuel element and locked in place (hardened steel collar and several locks).

[c]HEU is defined as containing 20% or more U-235 in uranium, 12% or more of U-233 in uranium, or mixtures of U-235 and U-233 in uranium of equivalent concentrations.

The methods selected for incorporating necessary radioactivity into the fuel material will depend on the radioactivity level and duration, as well as other factors such as cost. Candidate methods and radiation levels are indicated in the following table and references.

Table A-2. Candidate methods and radiation levels for spiking fuel materials

Fuel material type	Minimum 2-year radiation level, (rem/hr at 1 m)	Process	Minimum initial radiation level, (rem/hr at 1 m)	References
PuO_2, $HEUO_2$ powder or pellets	1,000/kgHM	Co-60 addition	1,300/kgHM	2, 3, 5, 6
PuO_2-UO_2 and $HEUO_2$-ThO_2 powder or pellets	100/kgHM	Co-60 addition	130/kgHM	2, 3, 5, 6
		Fission product addition (Ru-106)	400/kgHM	2, 3, 5, 6
LWR, LWBR, or HTGR recycle fuel assembly	10/assembly	Co-60 addition	13/assembly	2, 3, 5, 6
		Fission-product addition (Ru-106)	40/assembly	2, 3, 5, 6
		Pre-irradiation (40 MWd/MT)	1,000 (30 day)/ assembly	4
LMFBR or GCFR fuel assembly	10/assembly	Co-60 addition	13/assembly	2, 3, 5, 6
		Fission-product addition (Ru-106)	40/assembly	2, 3, 5, 6
		Pre-irradiation (40 MWd/MT)	1,000 (30 day)/ assembly	4

REFERENCES FOR APPENDIX A

1. Safeguarding a Domestic Mixed Oxide Industry Against a Hypothetical Subnational Threat, NUREG-0414, U.S. Nuclear Regulatory Commission, May 1978.

2. J. E. Selle, Chemical and Physical Considerations of the Use of Nuclear Fuel Spikants for Deterrence, ORNL TM-6412, Oak Ridge National Laboratory, October 1978.

3. J. E. Selle, P. Angelini, R. H. Rainey, J. I. Federer, and A. R. Olsen, Practical Considerations of Nuclear Fuel Spiking for Proliferation Deterrence, ORNL TM-6483, Oak Ridge National Laboratory, October 1978.

4. G. F. Pflasterer and N. A. Deane, Pre-Irradiation Concept Description and Cost Assessment, GEFR 00402.

5. Modification of Strategic Special Nuclear Materials to Deter Their Theft or Unauthorized Use, IRT-378-R, (The Spiking of Special Nuclear Materials as a Safeguards Measure, Vol. 2, BNL File No. 5.9.1 for Vol. 1).

6. E. A. Straker, Material Radiation Criteria and Nonproliferation, SAI-01379-50765, Science Application, Inc., January 8, 1979.

APPENDIX B

Responses to Comments by the U.S. Nuclear Regulatory Commission
PSEID, Volume I, Light-Water Reactors

Preface

This appendix contains comments and responses resulting from the U.S. Nuclear Regulatory Commission (NRC) review of the preliminary safety and environmental submittal of August 1978. It should be noted that the NRC comments are the result of reviews by individual staff members and do not necessarily reflect the position of the Commission as a whole.

A. RESPONSES TO GENERAL COMMENTS

1. Regarding the U.S. Nuclear Regulatory Commission (NRC) request to reduce the number of reactor concepts and fuel-cycle variations, the Nonproliferation Alternative Systems Assessment Program (NASAP) set out to look at a wide variety of reactor concepts and fuel cycles with potential nonproliferation advantages. These various concepts have differing performance characteristics in other important respects, such as economics, resource efficiency, commercial potential, and safety and environmental features. The relative importance of these other characteristics and trade-offs has been determined and the findings are incorporated in the NASAP final report.

2. Regarding the comment on the need to address safeguards concepts and issues, some concepts for providing protection by increasing the level of radioactivity for weapons-usable materials have been described in Appendix A to each preliminary safety and environmental information document (PSEID). Appendix A has been revised to reflect NRC comments.

   An overall assessment of nonproliferation issues and alternatives for increasing proliferation resistance is provided in Volume II of the NASAP final report and reference classified contractor reports.

B. QUESTIONS ON EXTENDED BURNUP

Question 1:

When there is a request for a license to permit extended burnup to 50,000 MWd/MT, the applicant will, of course, have to satisfy the criteria established in the Standard Review Plan, in particular, for Fuel System Design. A considerable portion of the Standard Review Plan is concerned with the analyses and assessment of transients and accidents. In Volume I of the light-water reactor preliminary safety and environmental information document (LWR PSEID) and supporting documentation, we see little evidence of a comprehensive and systematic progam to consider these areas. As we understand it, the bulk of the experimental effort in the area of extended burnup is in the area of "normal operation" irradiation of lead assemblies to 50,000 MWd/MT, while little, if any, is in the area of transient behavior. To what extent does the U.S. Department of Energy (DOE) research and development (R&D) plan for extended burnup include transient testing of high burnup fuel pins? Include in your discussion the type of testing planned, the schedule, and the facilities to be used (e.g., a power burst facility).

Response:

The DOE has recently initiated a research program whose objective is to perform a licensing assessment of the use of extended burnup fuel in light-water reactors. This assessment is directed toward identifying the additional information needs which may be required to support the licensing of extended burnup fuel, but which will not be available from existing R&D programs. One output of this licensing assessment program will be recommendations on how additional information can be obtained as well as what, if any, future R&D programs should be initiated. A specific response to Question No. 1 will be available upon completion of this licensing assessment program. The program is scheduled for completion during the second quarter of FY-1980.

Question 2:

The PSEID, Vol. I, presented little specific information on the various design changes necessary to accommodate the increased fission gas inventory for the high burnup option (to 50,000 MWd/MT). At the meeting with Combustion Engineering, Inc. (C-E) on November 7, 1978, various possibilities were presented including change in fuel rod length, and/or change in fuel column length for solid, hollow, and duplex pellets. Has the DOE been able to narrow down these possibilities and arrive at a best option for accommodating the fission gas pressure problem?

Response:

A single design has not come forth as yet. A number of different designs are presently undergoing extensive evaluation as part of the extended burnup demonstration program. Because of the early stage of their development, a multitude of designs is still being carried along with the anticipation that as more data and analyses become available, preferred designs will emerge. Because each of the fuel manufacturers is working independently at developing extended burnup fuel, a number of different approaches may be adopted to accommodate the effects of going to extended

burnup. Only in the long term, when extensive fuel performance data becomes available, could a single design be selected as the preferred design.

Question 3:

In addition to the Nonproliferation Alternative Systems Assessment Program (NASAP) program at Combustion Engineering, there are other reactor manufacturers who have extended burnup studies in place (e.g., Babcock & Wilcox, Westinghouse, Exxon, and General Electric). How do these programs complement, if at all, the C-E program directed toward better fuel utilization? Are there unique features of any of these programs that should be taken into account in an overall licensing/safety evaluation of extended burnup cores?

Response:

In addition to the NASAP program at Combustion Engineering, there are several other DOE programs directed toward achieving better fuel utilization in the once-through (O-T) LWR. Some of these progams have been initiated by NASAP and involve studies of a variety of options for reducing uranium requirements in LWRs. Other DOE development programs (some in progress and some in procurement) involve commercial reactor demonstrations of extended burnup fuel and other uranium utilization improvement options and test-reactor irradiations followed by postirradiation examination.

The NASAP program is scheduled for completion by the end of December 1979. The NASAP assessment has included studies of potential retrofittable modifications in LWR design and/or operational procedure that provide improved uranium utilization. The specific option of extended burnup is being considered as a part of NASAP studies at General Electric, Westinghouse, and Exxon. Table B-1 summarizes the objectives of the NASAP studies at General Electric, Exxon, and Westinghouse. For the extended burnup option, these studies are focused both on determining the uranium utilization and economic aspects and on providing recommended development programs for commercializing and licensing extended burnup cores. The results of these studies will be evaluated as input to the DOE development and demonstration programs for improved uranium utilization. Results from the DOE extended burnup development and demonstration programs will be available for the NRC licensing/safety evaluations of extended burnup cores.

Each of the five LWR fuel suppliers in the United States has proposed DOE high burnup demonstrations in an operating reactor (with utility participation).

In the Oconee plant of the Duke Power Co., a demonstration is proceeding in which five Babcock & Wilcox Mark B fuel assemblies which were due to be discharged are being examined and reloaded to burn an extra cycle following an NRC licensing review. These assemblies will be examined again upon completion of the extra cycle. The objective of this project is an assembly average discharge burnup of 38,000 MWd/MT, but four of the assemblies are expected to attain 40,900 MWd/MT. In a DOE program with Arkansas Power and Light, fuel capable of achieving a discharge batch average burnup of 45,000 MWd/MT will be designed and developed. After irradiation of two sets of lead test assemblies, this demonstration is planned to involve an entire reload batch of approximately 59 fuel assemblies. Other DOE high-burnup demonstration projects are in the proposal or procurement phase.

Table B-1. Ongoing NASAP studies of potential once-through LWR fuel-utilization-improvement options

Contractor	Objective
Westinghouse	Evaluate a variety of potential O-T improvement options. Select more promising options and estimate improvement in uranium utilization from base case and effect on fuel and power costs. Determine difficulty of retrofit and estimate potential commercialization date based on R&D needs, costs, and schedules.
Exxon	Determine uranium utilization, fuel and power costs as a function of burnup, batch size, and cycle length for pressurized-water reactors (PWRs) and boiling-water reactors (BWRs). Identify materials concerns for high-burnup fuels and recommend development and demonstration programs to collect data required for commercial acceptance of high-burnup fuels, including safety and licensing aspects.
General Electric	Estimate the potential U_3O_8 savings from the most promising BWR O-T cycle improvements both individually and in combination. Determine the major practical difficulties in the implementation and retrofit of the most promising options and determine the research, development, and demonstration requirements to prove technical feasibility, gain NRC approval, and provide preliminary scoping cost estimates and schedules. Determine the extent of impact of O-T improvements on fuel utilization in the recycle mode.

The DOE is also sponsoring fuel-cladding interaction remedy programs with the objective of demonstrating advanced fuel designs that are resistant to this failure mechanism at both current burnup levels and higher burnup levels. In phase 1 of a program with Commonwealth Research Corp. and General Electric Corporation, four lead test assemblies (LTAs) containing barrier fuels begin irradiation in Quad Cities-1 in February 1979, following the NRC licensing review. Continued irradiation and evaluation of the LTAs is part of the phase 2 activities. In a program with Consumers Power Co./Exxon Nuclear Co. and Battelle Pacific Northwest Laboratory, selected fuel-cladding interaction remedy fuels were licensed for irradiation testing in the Big Rock Point Reactor, and are also being ramp tested in the Halden Reactor.

A program has been initiated with the Tennessee Valley Authority (TVA). The first phase of the study will concentrate on eight options for improving uranium utilization in BWRs, including the extended burnup option. The TVA will manage the project and will involve the General Electric Corporation as a subcontractor. Priority is being given to alternatives which can be implemented in the near term.

The DOE also plans to participate in an international cooperative program, currently being formulated, to obtain irradiation data on high burnup effects. The technical management will be performed by Battelle Pacific Northwest Laboratory.

In summary, information relative to increased fuel burnup and other uranium utilization improvement techniques is being obtained in a variety of on-going experimental and power reactor demonstration programs. Demonstrations in progress have gone through the NRC licensing procedures. Additionally, other demonstration programs are in either the proposal or procurement stage as are assessments to determine additional information needs that may be required to support licensing full-reload batches of extended burnup fuel. As the development and demonstration of extended exposure LWR fuels proceeds through current and future DOE programs, additional information will be developed for consideration in NRC licensing/safety evaluation.

Question 4:

Provide a complete list of the nonsaturating fission products produced for a 50,000 MWd/MT cycle. Also provide a decay heat curve for extended burnup.

Response:

Table B-2 is a complete list of the nonsaturating fission products obtained from ORIGEN calculations for the 30,000- and 50,000-MWd/MT cycles. Their respective activities in curies/assembly are shown at time of discharge. For ease of comparison, the percentage increase in activity resulting from going to the higher burnup is also shown.

Figure B-1 shows the decay heat from both the 30,000- and 50,000-MWd/MT cycles as a function of time. These curves show that in the long term (e.g., at 10 years), the decay heat of the extended burnup cycle is about twice that of the conventional burnup cycle. In the short term, the decay heats of the two cycles are comparable since they are set by the saturated fission products.

Table B-2. Fuel assembly activities,[a] nonsaturated isotopes only

Nuclide	Activity (Ci)/assembly 30,000 MWd/t	Activity (Ci)/assembly 50,000 MWd/t	Percent[b]
Kr-85	3.71^{+3}	5.31^{+3}	43.1
Sr-90	3.05^{+4}	4.49^{+4}	47.2
Y-90	3.20^{+4}	4.73^{+4}	47.8
Mo-103	5.86^{+5}	6.09^{+5}	3.9
Tc-103	5.98^{+5}	6.21^{+5}	3.9
Ru-103	6.06^{+5}	6.37^{+5}	5.1
Tc-106	2.59^{+5}	2.80^{+5}	8.1
Ru-106	2.11^{+5}	2.65^{+5}	25.6
Sn-129	3.90^{+4}	4.06^{+4}	4.1
Sb-129	1.21^{+5}	1.26^{+5}	4.1
Te-129m	3.15^{+4}	3.30^{+4}	4.8
Te-129	1.15^{+5}	1.20^{+5}	4.4
I-127[c]	.1379	.2121	53.8
I-129	1.28^{-2}	2.04^{-2}	59.4
Cs-134	9.42^{+4}	1.83^{+5}	94.2
Sb-134	3.32^{+4}	3.39^{+4}	2.1
Xe-135	1.25^{+4}	1.35^{+5}	8.0
Cs-135	1.31^{-1}	2.39^{-1}	82.4
Cs-136	2.22^{+4}	3.82^{+4}	72.1
Cs-137	4.26^{+4}	6.45^{+4}	51.4
Ba-137m	4.02^{+4}	6.13^{+4}	52.5
Xe-143	6.68^{+3}	6.80^{+3}	1.8
Xe-144	1.52^{+3}	1.61^{+3}	5.9
Ce-144	4.45^{+5}	4.61^{+5}	3.6

[a] $Y.YY^{+x}$ represents $Y.YY \times 10^{+x}$.

[b] $$\text{Percent} = \frac{\text{50,000 MWd/t Activity} - \text{30,000 MWd/t Activity}}{\text{30,000 MWd/t Activity}}$$

[c] I-127 is a stable isotope--units are gram-atoms/assembly.

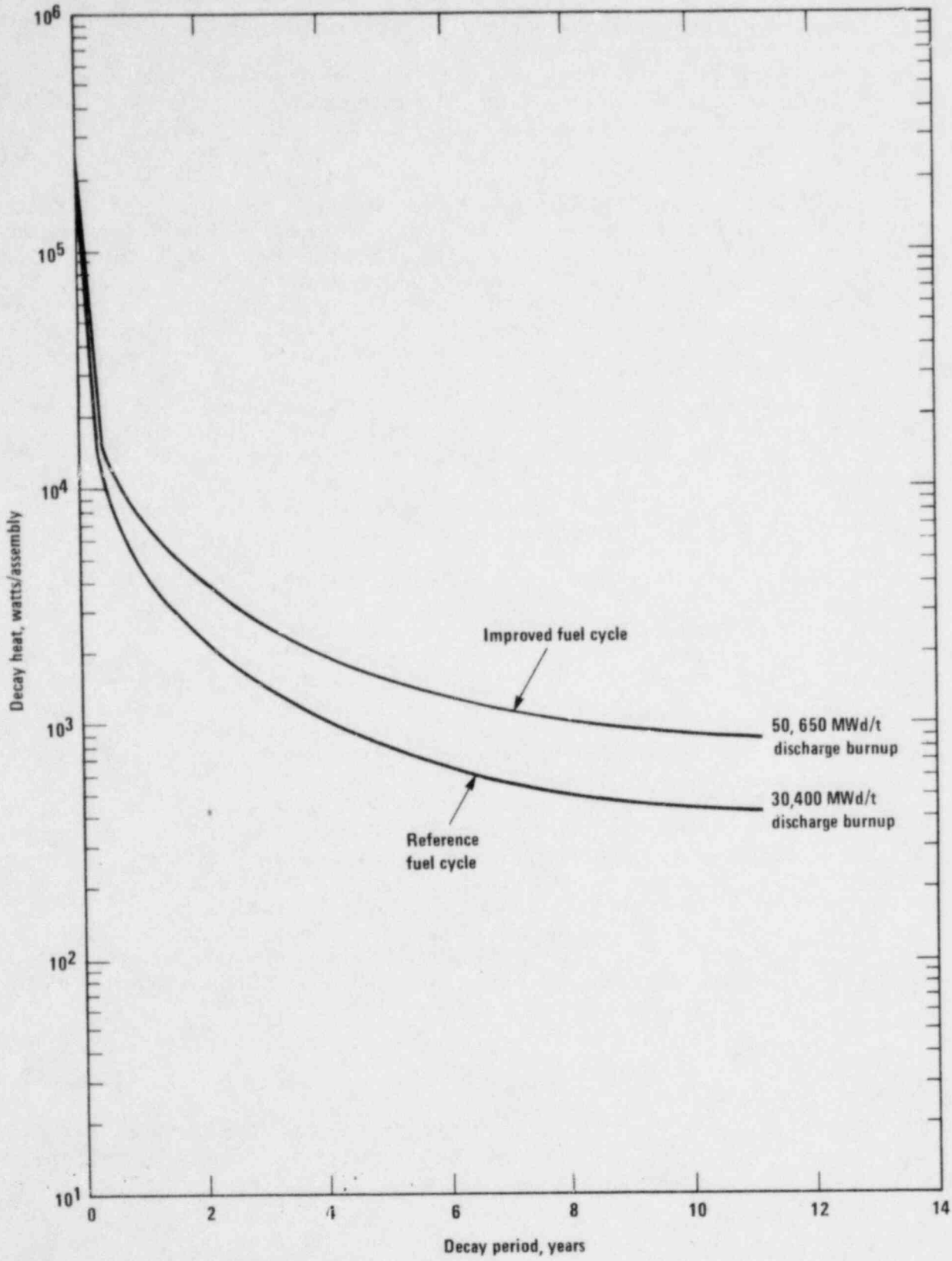

Figure B-1. Decay heat generation vs. cooling time per assembly.

Question 5:

Provide any analyses or assessments of power peaking due to the increasd U-235 enrichment necessary for the LWR extended burnup option.

Response:

The limited analyses performed thus far for the LWR extended burnup option have indicated that higher power peaking than the conventional fuel cycle may occur for standard fuel management schemes (e.g., three-batch, OUT-IN fuel shuffling). This higher peaking appears to be due to the larger differences in the infinite multiplication factors between the fresh and partially burned fuel batches in the extended burnup fuel mangement scheme compared with those in the conventional fuel management scheme. This larger "ripple" in infinite multiplication factors can cause larger reactivity and flux gradients which, in turn, yield a larger "ripple" in the power distribution. The extended burnup cycle potentially has peaking factors, therefore, that are 5 to 9% higher than in the conventional cycle. This conclusion is based on comparisons of a "nonoptimized" extended burnup fuel management scheme against a "near-optimized" conventional fuel management, and thus tends to be pessimistic. More extensive experience with these extended burnup fuel management schemes may result in peaking factors comparable to those of the conventional fuel cycle.

Figures B-2 and B-3 present two-dimensional (x-y) power distributions for the extended-burnup (five-batch) core and the conventional core. These power distributions were derived from coarse mesh (16 nodes/assembly) PDQ calculations with the pin powers estimated from spline fits of the nodal powers. Although the assembly (box) powers are fairly accurate, the pin powers are underpredicted by about 5% because peaking around the water holes is not taken into account. Regardless of the exact value of the pin-to-box peaking, these figures appear to indicate that power peaking will be somewhat higher in the extended burnup cycle than in the conventional cycle.

Analyses have not as yet been performed for extended burnup fuel cycles employing advanced fuel management schemes (e.g., low leakage). The use of burnable poison shims in these fuel management schemes has the potential of reducing the above-noted "ripples" in reactivity; moreover, B_4C or borosilicate are used in present shim rods. The use of gadolinium mixed homogeneously with the UO_2 may facilitate obtaining even more uniform power distributions than those obtained with current burnable poison shims because restrictions on the number and locations of these shim rods would be eliminated if gadolinium were used.

Question 6:

The present enrichment limit for fuel handling and storage at PWR plants is 4%. What approach does the DOE intend to take in these areas in light of the increased enrichment (4.3%) for extended burnup cores?

Response:

Licensing of fuel handling and storage will have to be reviewed by the reactor owners as part of their reload submittal to the NRC. In the very near term, this will occur for a few extended burnup demonstration assemblies. Similar requirements for a full core will not occur before about 1990. By this time, it is anticipated that

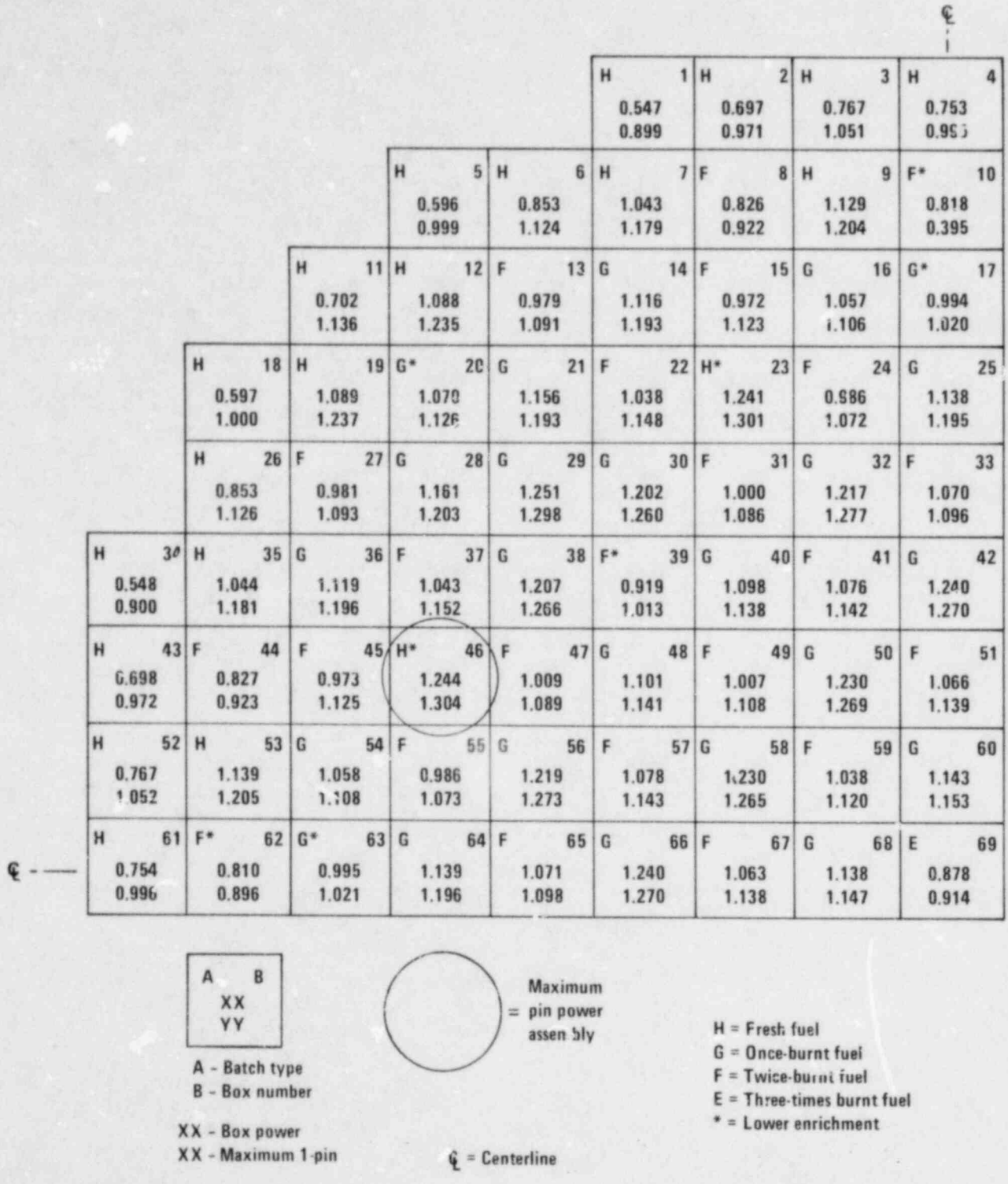

Figure B-2. Core power distribution early in a near-equilibrium cycle (30,000 MWd/t, 3-batch cycle).

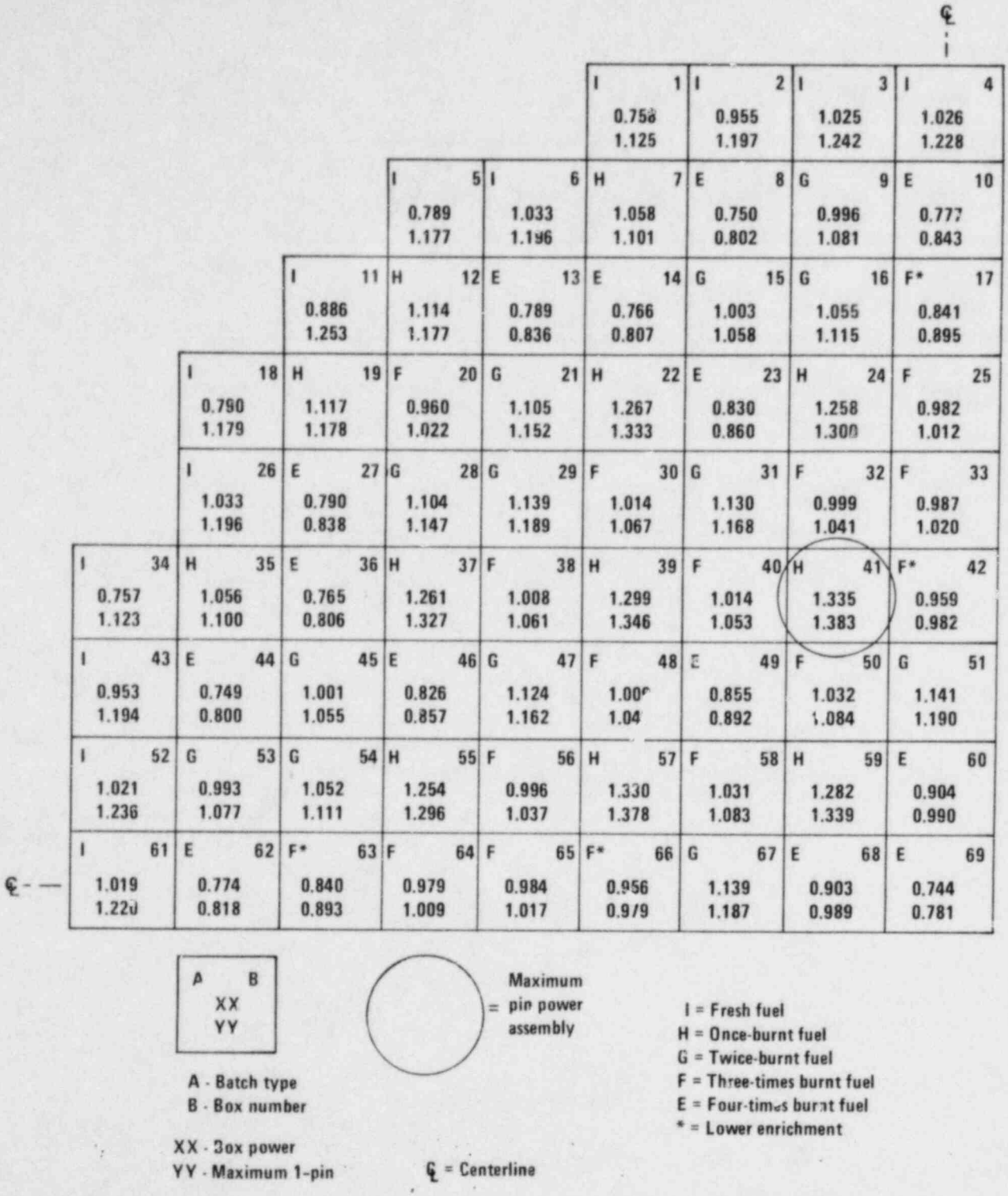

Figure B-3. Core power distribution early in a near-equilibrium cycle (50,000 MWd/t, 5-batch cycle).

away-from-reactor (AFR) storage will also be available to alleviate some of the restrictions in the storage area. In the meantime, fuel manufacturers are investigating the impact of the higher enrichments of extended burnup fuel cycles on fuel handling, storage, and transportation in support of their participation in the DOE uranium utilization improvement (extended burnup fuel demonstration) program. This effort will be aimed primarily at providing the backup data needed to license the few assemblies in the early demonstration programs. Further effort will undoubtedly be required to support the full-batch and full-core demonstrations in the later stages of these DOE programs. Finally, the recently initiated DOE program on the licensing assessment of extended burnup fuel will address the issues of fuel manufacture, handling, storage, and transportation as one of its subtasks; the results of this program will be available in FY-1980.

Question 7:

Provide any analyses or assessments of the change in shutdown margin in going from 30,000 to 50,000 MWd/MT.

Response:

The shutdown margin requirements of a PWR are typically set by the steamline-break (SLB) accident. Since the moderator temperature coefficient of reactivity is more negative for the 50,000 MWd/MT cycle than for the 30,000 MWd/MT cycle, and since the net rod worth (with the most reactive rod stuck out) is smaller, the shutdown margin in the event of an SLB accident will be less. Figure B-4 is presented as an example of the difference in core reactivities that might occur as a result of the cooldown during an SLB accident. At 400°F, approximately 0.38% $\Delta\rho$ more reactivity will be inserted by cooldown in the 50,000 MWd/MT cycle than in the 30,000 MWd/MT cycle. Similarly, at normal operating temperatures, the net control rod worth of the 50,000 MWd/MT cycle is about 1.13% $\Delta\rho$ less than that of the 30,000 MWd/MT cycle. These two effects result in a smaller shutdown margin (by about 1.4% $\Delta\rho$) in the event of an SLB accident.

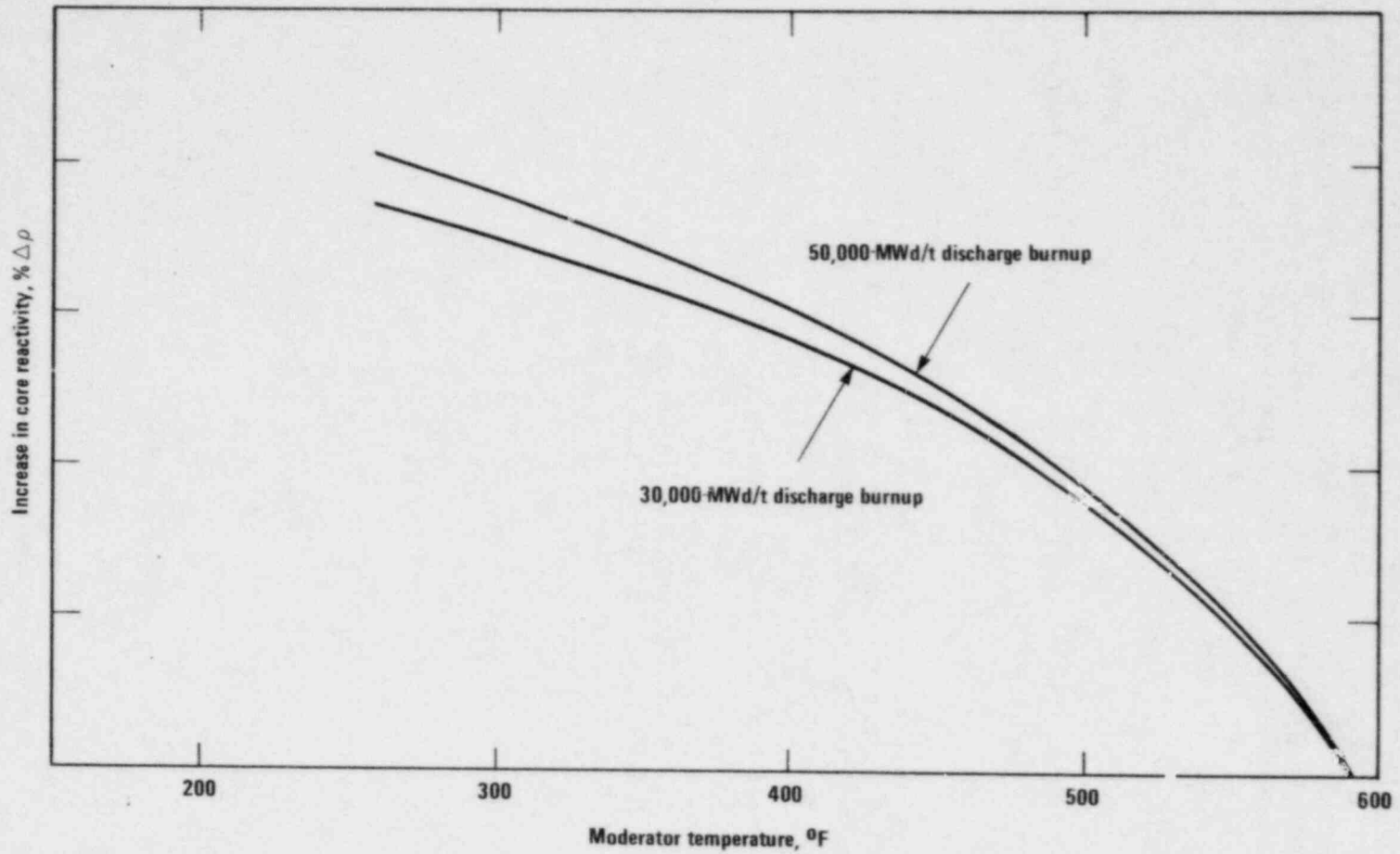

Figure B-4. Steam line break cooldown (worst rod stuck out). Increase in core reactivity relative to 592°F T_{mod} vs. moderator temperature (equilibrium cycle PWR).

C. QUESTIONS ON SPIKED RECYCLE

The flow sheet for the fuel cycle of a PWR using 3 to 5% low-enrichment uranium (LEU) with Pu recycle and Co-60 spiking shows two Purex reprocessing operations. In this arrangement, fresh uranium fuel is reprocessed in Purex 1 and mixed-oxide fuel is reprocessed in Purex 2. This plan is difficult to understand and leads to the following questions:

1. From a proliferation standpoint, why is it acceptable to recover about 40% of the plutonium as pure plutonium, while the remainder is recovered as coprocessed 2% Pu in uranium?

2. What is the purpose or intent of the two Purex operations? Are they designed to optimize recycle of uranium? Do the two Purex operations represent the same solvent extraction line, with fuel being campaigned, or are the operations carried out in physically distinct equipment?

   Some discussion of the purpose of these two Purex lines is required, together with an indication of the incremental reprocessing costs relative to a single Purex operation.

   Does the flow sheet apply for both PWRs and BWRs? If not, what is the plan for BWR units?

3. The use of Co-60 represents the addition of a spikant to the presently conceived recycle flow sheets. In developing a generic environmental assessment of a fuel cycle, major impacts of processing and disposal of all the fuel cycle material should be included. In the case of the Co-60 feed material, the assessment should include all of the operations involved in producing the radioactive cobalt, including the reactors and cobalt-processing facilities.

   A Co-60 balance across each of the fuel cycle operations should be given (i.e., input, amount to waste, amount release). In addition, the behavior of the soluble cobalt in the recycle fuel fabrication operations (preparation of oxide, sintering) should be given.

   Further, the use of Co-60 should be analyzed in light of its effects on operations such as fuel transportation, fuel fabrication, reactor operations, reprocessing, etc.

   In addition, the level of occupational exposure in the overall handling and use of recycle fuel in the cycle should be assessed and the potential effects on population exposures should be considered.

Responses to Questions 1-3:

The recovery of pure plutonium within a secure reprocessing facility is not expected to cause proliferation problems since the product plutonium nitrate stream is diluted with the coprocessing product and the cobalt-60 spike before shipment to the recycle fuel fabricator. The intent of two Purex operations is to recover cobalt-free uranium for recycle to enrichment as fuel. The reprocessing is assumed to be conducted in a single solvent extraction line with the fuel campaigned. Make-up and recycle fuel rods are assumed to be assembled into bundles in a hot-cell facility (due to presence of Co-60) and disassembled at the reprocessing plant to segregate the

rods containing Co-60. The flow sheet should apply to both PWRs and BWRs. The incremental reprocessing cost would be for the disassembly, front end operations, and the product-blending operations.

At the present time, details of the spiking process and necessary supporting facilities have not been developed.

Question 4:

In overview, the benefits obtained by this concept should balance with the economic, environmental, social, etc. costs that are incurred and it should be compared with similar cost/benefit analyses for other alternative fuel cycles.

Response:

Comparative cost/benefit analyses of alternative fuel cycles have not yet been performed. It is to be noted that the PSEIDs addressed the various design options together with a preliminary assessment of the safety and environmental implications of the various options. In-depth design and fabrication data are not available for the alternative fuel cycles.

Specifically, the concept of spiking has been selected as one of the possible approaches for increasing proliferation resistance. Decisions such as the choice of a specific spikant material, its relative concentration, etc. have not been worked out in depth. Consequently, a detailed assessment of the impact of spiking on fuel-cycle cost, safety, licensing, improvement in proliferation resistance, and improvement in resource utilization has not been performed yet.

Some aspects of these issues are covered in the NASAP final report.

D. QUESTIONS ON DENATURED URANIUM RECYCLE

Question 1:

Chapter 5 outlines the concept of a PWR using uranium fuel enriched with 12% U-233 and mixed with thorium oxide. The flow sheet for this denatured U-233/thorium cycle (PWR DU(3)-Th recycle DU(3)) does not appear to be a self-standing or independent one. The flow sheet and reactor charge data show that U-233 is required for sustained operations. From the standpoint of the fuel-cycle diagram, the source of the U-233 supply is not mentioned or described.

Response:

The flow sheet was not intended to be self-standing or independent. U-233 is required for sustained operations. Since U-233 does not appear in nature, it must be supplied by an exogenous source. Typically, it has been envisioned that U-233 would be produced in a plutonium/thorium burner reactor (e.g. an LWR deployed in a secure Nuclear Energy Center) or in the thorium blanket of a fast-breeder reactor. The appropriateness of the U-233 source was not addressed by Combustion Engineering, Inc., in any of its studies for the NASAP.

NRC Comment:

The fuel cycle shown requires at least two "Secure" centers: one for 50% U-233, which is denatured to 12% U-233 during fuel fabrication, and another for storage of spiked plutonium, which is recovered from this fuel cycle. Substantial additional information on the flow sheet is required for its assessment, such as:

Question 2:

a. What is the source of U-233 supply? Data must be supplied on the reactor cycle that produces U-233 so that environmental, safety, and safeguards impacts of that production can be given.

Response:

From the standpoint of the fuel-cycle diagram, the source of U-233 is immaterial. It could be from an LMFBR with thorium blankets, from government production reactors, or from other sources, such as the thorium blanket used in an LWR. System descriptions and safety/environmental assessments of possible sources of U-233 (LMFBRs, HTGRs, and GCFRs) are given in PSEID Volumes III, IV, V, and VI.

b. What are the definitions of a "secure" storage center for U-233 and a "secure" storage center for plutonium? What are the fuel-fabrication and reprocessing facilities considered not to require "secure" status?

Response:

A "secure" storage center is assumed to be a government-operated facility with adequate security by design, and with a guard force, to prevent diversion of material. Fuel-fabrication and reprocessing facilities are assumed to be commercial facilities whose material in process and as product is sufficiently radioactive to prevent

diversion, or is of a chemical nature, i.e. denatured, to minimize the value of the material as a weapon component.

c. Additionally, data on the Co-60 spike must be given. What is the plan for sale of Pu? Who is the customer? In what fuel cycle is it to be employed? How is the problem of the relatively short half-life of Co-60 (7 years) handled? What is the form of plutonium in storage?

Response:

Plutonium is assumed to be utilized in a reactor on spiked recyle. In the event of extended storage that would reduce its effectiveness, additional Co-60 would need to be added before shipment to fabrication. It is assumed the plutonium would be in the oxide form. As noted earlier, details of the spiking process and necessary supporting facilities have not been developed.

d. Is the flow sheet valid for BWRs as well as PWRs?

Response:

It is assumed that the flow sheet would be valid for BWRs with a modification of mass flows.

e. Detailed information on gaseous effluents from Thorex fuel reprocessing must be provided.

Response:

Detailed information has not been developed or is not available on gaseous effluents for the processes on the fuel-cycle flow diagrams, except to the extent that they are covered in PSEID Volume VII.

f. What are the fuel-cycle economics? How many reactors are required to justify reprocessing for the use of this cycle? How many reactors must be used to produce the U-233 and Co-60 used in this cycle?

Response

Fuel-cycle economics are analyzed in the NASAP final report. See response to c above for the Co-60 question.

Question 3:

It appears that the first licensing issues to be addressed may be those concerned with the known physical and chemical property differences between thorium and uranium, and the physics behavior of U-233 as opposed to that of U-235. Any modifications in behavior or component design introduced as a result of these initial considerations must then be examined for any effects they might have on the previously licensed reactor and on plant features. The initial evaluation would be assisted by an expanded discussion of the following questions.

a. Fuel Qualification. A comprehensive picture of fuel behavior, growth, densification, fission product migration, transient fuel damage limits, and other safety-related fuel performance information (Section 5.5, final paragraph) is required for qualification in a large, high-performance reactor. To what extent can this information be obtained in Shippingport or from other scheduled tests? Are other fuel development programs visualized? Will there be transient experiments or simulated accident conditions to examine the range of capabilities of this fuel?

b. How extensive in nature is the physics verification program projected to be as a requirement for licensing?

c. According to information presented at the November 7 meeting in Windsor, Connecticut. the composition of the core supplying the U-233 may undergo considerable variation over its lifetime. If this is correct, it would present a problem in the licensing of a fairly wide range of core compositions. What ranges of core compositions (chemical and isotopic) are anticipated for the various prebreeding options and what arrangements would be undertaken for core qualification over these ranges?

Response:

Answers to many of the specific questions raised under this general fuel performance question can be derived from the LWBR program and from the experience gained from the early thorium irradiations in the Shippingport, Indian Point, and Elk River reactors. The answers, however, are not available at the present time.

E. QUESTION ON LWR VARIANTS

Question 1:

Noting pages 3-26, 4-21, and 5-18 (same pages in this volume), some cost and time-magnitude data need to be identified for each on a consistent basis to provide the comparability required.

Response:

Generation of data on cost and time-magnitude was not part of the effort specified for the PSEIDs; however, cost and timing data have been generated on a consistent basis by the various DOE assessment contractors and are being incorporated into the NASAP final report (available only in draft form at the present time). Specifically, information on the research and development costs prior to commercialization and the schedules for carrying out research, development, and demonstration projects with respect to LWR Variants is available in Section 3.2 of Volume IV, title: "Commercial Potential" of the DOE report, Nuclear Proliferation and Civilian Nuclear Power, Report of the Nonproliferation Alternative Systems Assessment Program, July 1979.

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