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CONCEPTUAL DESIGN AND ECONOMIC
EVALUATION OF A STEAM-COOLED
FAST BREEDER REACTOR

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Project Engineer: G. Sofer

November 15, 1961

Work Performed under Contract AT(30-1)-2303(XII)
with the United States Atomic Energy Commission

UNITED NUCLEAR
CORPORATION

DEVELOPMENT DIVISION
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ABSTRACT

A conceptual design and economic evaluation of 300 and 40 MW_e steam-cooled fast breeder reactor power plants was performed by the Development Division of United Nuclear Corporation (NDA) under contract with the USAEC. A reactor core composed of uranium-plutonium oxide rod-type fuel elements clad with Inconel-X and surrounded by a blanket of depleted UO₂ fuel was studied in some detail. Reactor breeding ratios of from 1.27 to 1.5 and overall system doubling times of from 20 to 30 years are achievable. For the near term (1967) 300 MW_e plant, an energy cost of 7.6 mills/kwh is estimated, based on AEC ground rules for privately financed plants and utilities. This cost may go down to 5.7 mills/kwh by 1975. For the 40 MW_e plant corresponding energy costs are 19.5 and 13.7 mills/kwh, respectively. The R&D program required for this reactor concept is estimated at \$12 million with an additional \$3 million for improvements leading to the 1975 reactor.

Investigation of the operational and safety aspects of the reactor indicated that satisfactory procedures can be used for startup, shutdown, and emergency cooling of the reactor. An increase in reactivity upon flooding can be prevented by incorporating small amounts of high resonance absorption material in the core.

Preliminary calculations indicate a substantial increase in reactivity upon loss of coolant for the 300 MW_e plutonium oxide fueled reactor. To obtain designs with satisfactory voiding characteristics it may be necessary to provide high neutron leakage as in a low L/D core or smaller volume core. Acceptable

voiding characteristics appear possible with a plutonium fueled 40 MW_e reactor cooled with H₂O steam, a plutonium fueled 300 MW_e reactor cooled with D₂O steam, and a 300 MW_e U²³³-thorium fueled 300 MW_e breeder reactor cooled with H₂O steam.

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1. INTRODUCTION

Current interest in the development of breeder reactors stems from their promise to yield economic power costs within 5-15 years, while helping to conserve fissionable material resources. This interest does not end with the development of economic nuclear power on the basis of present AEC schedules for fissionable material cost, but extends to a possible point in the future of nuclear power development, when the easily accessible high grade uranium ores are exhausted, and the cost of fissionable and fertile material may be as much as 30 times higher than present costs. It is important to note that even under such highly inflated fuel material costs, a breeder reactor with a breeding ratio of 1.3 and an average fuel burnup of 60,000 MW-d/tonne can yield a total energy cost of less than 10 mills/kwh.

The reactor concept treated in this report is a fast breeder power reactor fueled with $\text{PuO}_2\text{-UO}_2$ clad with Inconel-X, and cooled with high pressure superheated steam using a direct cycle Loeffler system. This reactor concept attempts to combine the advantages of fast breeder reactors with those of direct cycle high efficiency nuclear superheaters, by using high pressure steam as the coolant instead of liquid metals. Although the density of high pressure steam is lower than that of sodium, the specific heat and allowable velocity are higher. The use of a stainless steel or Inconel clad oxide fuel element permits high cladding temperatures and, consequently, the temperature drop across the steam film can be high enough to result in overall heat transfer rates comparable with those attainable with sodium cooling. A fast reactor core is characteristically small in diam-

eter and, therefore, lends itself to the use of a relatively compact pressure vessel. The use of steam cooling has the major advantages of a direct cycle system with significant savings in capital and operating cost. The amount of hydrogen present in the core with steam cooling is small, and hence its influence on breeding gain is small.

The work reported herein is sponsored by the USAEC, under contract AT(30-1)-2303(Project XII). Two reactor plant sizes are considered: a 300 MW_e central power station plant and a 40 MW_e plant. The plant designs were chosen on the basis of two startup dates: 1965-1967 representing essentially current technology, and 1975 representing improvements in technology which have promise of being achieved in the foreseeable future. An additional report has been issued as NDA 2148-4¹ for the comparative study of nuclear superheater reactors conducted by the USAEC's Division of Reactor Development. The present report includes the material covered in NDA 2148-4 plus an evaluation of the economics of the steam-cooled fast breeder concept. Additional material related to reactor safety and alternate designs is also presented.

The economic ground rules followed in this study for the near term reactors were specified in detail by the Division of Reactor Development to provide a uniform basis for evaluating a number of nuclear superheat reactors. They are summarized in the following:

Annual capital charges, %	14
Annual working capital charges, %	12.5
Plant operating factor, %	80
Fissionable material lease charge per annum, %	4
Value of UF ₆ and natural uranium	April 2, 1960 AEC price schedule
Value of PuO ₂	\$11.50/g
Fabrication cost of PuO ₂ containing fuel elements	Twice cost of corresponding UO ₂ fuel elements
Spent fuel shipping cost	\$16/kg of fuel

Reprocessing plant operating cost	\$16,400/day	
Max burnup for PuO ₂ in depleted UO ₂ , near term plants	50,000 MW-d/tonne initial U plus Pu	
Max burnup for PuO ₂ in depleted UO ₂ or PuC in depleted UC, 1975 plants	90,000 MW-d/tonne initial U plus Pu	
Max central UO ₂ temp, °F	4500 (k=1)	} under steady state conditioning
Max Inconel or Inconel-X Fuel cladding surface temp, near term plants, °F	1350	
Near term reactors		
300 MW _e	Operable by 1967	
40 MW _e	Operable by 1965	
Potential reactors		
300 and 40 MW _e	Operable by 1975	

2. SUMMARY

The conclusions reached in this study are as follows:

1. Attractive economics and good breeding characteristics (breeding ratios from 1.27 to 1.50) can be achieved in steam-cooled $\text{PuO}_2\text{-UO}_2$ fueled fast reactors. Low capital costs (\$204/kw_e for the 300 MW_e near term reactor) are obtained by virtue of a compact reactor core and the absence of large heat exchangers and complicated process systems. High temperature cladding materials such as Inconel-X can be used in this type of reactor, permitting steam temperatures and pressures comparable to modern utility practice (1400 psi/945°F). High burnups, possible with oxide fuels (50,000 MW-d/tonne), contribute to low fuel cycle and, hence, low overall energy costs (7.6 mills/kwh).
2. The reactor designs considered can be prevented from going prompt critical, when fully flooded, by incorporating a tolerable amount of high resonance absorption materials such as hafnium, indium, or others. An increase in reactivity upon loss of coolant is indicated according to preliminary calculations. Acceptable voiding characteristics were obtained with a plutonium fueled 40 MW_e reactor cooled with H₂O steam, a plutonium fueled 300 MW_e reactor cooled with D₂O steam, and a 300 MW_e U²³³-Th fueled breeder reactor cooled with H₂O steam. Additional studies are required to check these results.

2.1 NEAR TERM REACTORS

2.1.1 Reactor and Power Cycle, Near Term Design

The steam-cooled fast breeder reactor design uses a direct cycle where about 1/3 of the superheated steam produced in the reactor is fed directly to the steam turbine. The remainder is used to vaporize feedwater returning from the condenser in a Loeffler boiler. The combined steam flow exits from this boiler at saturation temperature and is returned to the reactor by means of low leakage steam circulators driven by steam turbines. See Fig. 2.1.

Turbine inlet conditions for the 40 MW_e plant are 1400 psig, 920°F, and for the 300 MW_e plant 1400 psig, 945°F. Regenerative feedwater heating and some heating with turbine bypass steam is used to raise the condenser feedwater to 495°F before it enters the Loeffler boiler.

The reactor core, axial and radial blanket, reactor supports, and thermal shielding are enclosed in a pressure vessel. Steam flow through the core and axial blanket is downward. See Fig. 2.2. About 10% of the coolant flows in a parallel pass through the radial blanket. Important reactor characteristics and power generation costs are shown in Table 2.1. For the 300 MW_e plant, the core is 6 ft in diameter and 5 ft high. The core is surrounded by a 1-ft thick blanket of depleted UO₂ (less than 0.35% U²³⁵). The reactor and thermal shielding are enclosed in a pressure vessel of 10.7 ft ID, 31 ft high and 7.5 in. in wall thickness. The core fuel elements are 0.25 in. diameter, Inconel-X clad oxide rods. The PuO₂-UO₂ fuel yields good breeding ratios, has good corrosion resistance to steam, and permits high burnups. The fuel and fuel element fabrication technology are well developed and further advances are expected. Inconel-X cladding permits a steady-state cladding surface temperature of 1350°F, and, therefore, a relatively high core specific power. The alloy has good strength characteristics and appears to have adequate corrosion resistance to steam.

The breeding ratio and critical mass values reported in Table 2.1 are based on fuel loadings of pure Pu²³⁸ isotope. Under isotopic concentrations

Table 2.1 — Steam-Cooled Fast Breeders — Summary of Design Data

	300 MW _e		40 MW _e	
	1967	1975	1965	1975
Turbine Inlet Pressure, psig	1400	2000	1400	1400
Turbine Inlet Temp, °F	945	1060	920	970
Net Plant Efficiency, %	34.7	36.4	31.5	32.8
Core				
Core diameter, ft	5.94	5.43	2.54	2.50
Core length, ft	4.96	4.40	2.65	1.97
Core pressure drop, psi	100	100	100	100
Core Fuel				
Fuel material	UO ₂ -PuO ₂	UO ₂ -PuO ₂	UO ₂ -PuO ₂	UO ₂ -PuO ₂
Fuel rod diameter, in.	0.250	0.200	0.250	0.200
Max. central fuel temp, °F	4500	4500	4500	4100
Rod clearance, spacing, in.	0.060	0.045	0.040	0.035
Clad material	Inconel-X	Hastelloy-N	Inconel-X	Hastelloy-N
Max. clad temp, °F	1350	1500	1350	1500
Clad thickness, in.	0.015	0.0125	0.015	0.0125
Blanket				
Fuel material	Depleted UO ₂	Depleted UO ₂	Depleted UO ₂	Depleted UO ₂
Blanket thickness, ft	1	1	1½	1½
% of total power	4	4	7	12
Nuclear Characteristics				
Critical mass, kg of Pu	1656*	1370*	261*	188*
Core breeding ratio	0.965*	0.95*	0.553*	0.53*
Total breeding ratio	1.29*	1.27*	1.42*	1.38*
Pu enrichment, kg Pu/ kg fuel	0.123	0.128	0.176	0.180
Avg fuel burnup, MW-d/ tonne	33,300	60,000	32,500	60,000
Max. fuel burnup, MW-d/tonne	50,000	90,000	50,000	90,000
Energy Costs, mills/kwh				
Fixed costs	4.6	4.0	13.5	10.2
Fuel cycle	2.6	1.1	4.2	1.8
Operation and maintenance	0.4	0.6	1.8	1.7
Total energy cost	7.6	5.7	19.5	13.7
Capital Cost, \$/kw _e	204	190	603	448

*Based on initial loading of pure Pu²³⁹ and no epithermal poison.

obtained at equilibrium recycle conditions, the breeding ratio of the 1967 300 MW_e reactor is raised from 1.29 to 1.51, and the critical mass drops from 1656 to 1441 kg by virtue of the larger values of η for the higher plutonium isotope Pu²⁴¹ formed.

For the 300 MW_e reactor, most of the conversion of U²³⁸ to plutonium takes place in the core rather than in the blanket (core breeding ratio near 1). This is advantageous in reactor control, requiring less frequent refueling, and results in lower fuel costs than in breeder reactors with smaller cores.

The 300 MW_e plant has a specific power of 0.50 MW_t/kg Pu²³⁹. Although this is a factor of two or more smaller than has been reported for sodium-cooled fast breeder reactors, a substantial reduction in fuel fabrication cost and overall plant capital cost are obtained with this concept which are likely to yield lower power generation costs than in sodium-cooled fast breeder reactors.

2.1.2 Operation and Safety

Tentative schemes for reactor startup and shutdown have been worked out. Analysis shows that these operations are possible without employing an auxiliary cooling gas system.

Emergency cooling of the core is accomplished by flashing the water stored in the steam drums (Loeffler boilers) into steam and permitting this steam to pass through the reactor on its way to the condenser. In case of a break in the main piping from the reactor to the turbine building, the steam can be led into the fuel storage decay pool instead of the condenser. The reactor can be cooled by this method for about 6 min, which is sufficient time to start emergency diesel equipment and auxiliary steam circulator for removal of the afterheat.

Normal control of the reactor is effected by means of control rods located in the core and blanket. They are made of stainless steel clad B₄C rods located in coolant channels interchangeable with those of the fuel. They are activated by individual electric motors located on top of the shield plug.

Refueling of the core for the 300 MW_e reactor takes place approximately every 50 days. An inward radial shift type of fuel management is used to approach a uniform fuel burnup. The reactor is shut down before refueling.

During refueling and other shutdown and startup operations, the reactor can be flooded with water. Reactivity rise is prevented by the addition of a tolerable amount of resonance absorber, such as hafnium or indium, to the fuel. Preliminary calculations based on a six-group diffusion theory indicate that ~0.03 atom of natural hafnium per atom of fuel (Pu + U²³⁸) is needed to provide sufficient epithermal absorption to prevent an increase in reactivity upon full flooding. The effect of this amount of hafnium on critical mass and breeding ratio is expected to be small.

The refueling vehicle is an unshielded machine located on top of the shield cover which transports the irradiated fuel elements under water from the core to the fuel chute leading to the decay pool. The shield plug is lifted prior to refueling. Ports provided in the shield cover and pressure vessel make it unnecessary to remove the vessel head in order to reach the fuel elements.

Voiding of the reactor such as in a loss of coolant accident may be a problem. Preliminary calculations indicate that the temperature and void coefficient of the 300 MW_e reactor are positive. The effects of voiding were calculated using a ten-group diffusion theory. The group structure used, although quite adequate for breeding ratio and critical mass determinations, yields approximate results when assessing the voiding effects because of insufficient energy group subdivision in the region below 5000 eV, which plays an important role in the voiding effect. Additional more accurate calculations of the reactor behavior during voiding accidents are required before the effect of this problem can be fully assessed.

2.2 1975 PLANT

Advanced nickel alloys promising cladding temperatures in excess of 1500°F are under development, promising a reduction in core size and better

steam conditions. These improvements plus the eventual achievement of burnups of the order of 90,000 MW-d/tonne with $\text{PuO}_2\text{-UO}_2$ fuel are expected to decrease power generation cost for the 300 MW_e plant to 5.7 mills/kwh by 1975, and for a 40 MW_e plant to 13.7 mills/kwh. Performance and cost characteristics of the 1975 plants are shown in Table 2.1. An alternate design using PuC-UC fuel is capable of higher overall breeding ratio than the oxide design, but would require the development of carbide fuel with corrosion resistance to steam.

2.3 ALTERNATE DESIGNS

2.3.1 D₂O Steam-Cooled PuO₂-UO₂ Breeder

A 300 MW_e $\text{PuO}_2\text{-UO}_2$ fueled fast breeder reactor cooled with D₂O steam was briefly examined. This reactor yields more favorable flooding and voiding characteristics than the H₂O steam-cooled 300 MW_e design. The breeding ratio for this reactor is slightly better than the reference design. The energy cost is increased by 0.5 mill/kwh from the reference design as a result of D₂O losses and inventory charges plus the use of a D₂O turbine. Additional information on the characteristics of this reactor is given in Sections 4.6 and 7.2.

2.3.2 U²³³O₂-ThO₂ Steam-Cooled Breeder

A 300 MW_e $\text{U}^{233}\text{O}_2\text{-ThO}_2$ fueled fast breeder cooled with H₂O steam was considered because of its promise to yield more satisfactory voiding characteristics than the $\text{PuO}_2\text{-UO}_2$ fueled reactor. A slightly negative void coefficient was calculated for this reactor, but uncertainty in the capture cross section used for thorium in the region between 3500 and 500 ev does not permit assigning a high degree of confidence to this result. No cost estimates were made in conjunction with this reactor concept. See Sections 4.6 and 7.3 for additional information on this reactor concept.

2.3.3 Alternate Fuel Configurations

Plate-type and matrix-type fuel elements penetrated by coolant tubes were examined. It was found that neither of these configurations offers significant

advantages over the fully developed rod-type element chosen for the reference design. The attractiveness of the matrix type element is contingent on any fabrication cost economies obtained over a rod-type element occupying an equivalent fraction of core volume. The heat transfer characteristics of the coolant require a minimum heat transfer surface, which leads to small fuel ligament spacing (70 to 90 mils) and small coolant channel tubes (0.125 in. ID). The matrix fuel element merits further examination as vibratory compaction techniques, capable of handling the above dimensions, are developed. Furthermore, in conjunction with the 1975 design, higher cladding temperatures are expected to be permissible, thus easing the heat transfer surface requirement.

2.3.4 Steam Circulation

Alternate methods for steam circulation were examined with the incentive of eliminating the mechanical steam blowers required in the reference design. A steam ejector system using high temperature steam from the reactor exit header could not be made to provide a sufficient pressure head for steam circulation.

A thermopressor system in which feedwater is sprayed into a segment of the steam circulation line is capable of providing about half the necessary pressure head.

A multipass steam circulation system, in which the necessary pressure is provided by pumping the feedwater only, is feasible only in conjunction with a pressure tube design, which is difficult to adapt for the relatively compact fast breeder cores.

2.3.5 Indirect Steam Cycle

An indirect steam cycle design in which liquid is excluded from the primary loop was briefly examined as a means of eliminating the possibility of core flooding. The added cost of heat exchangers required for this system, and the absence of a ready supply of saturated water, which would flash into steam in the event of a depressurization accident, favors the direct cycle approach.

2.4 R&D PROGRAM

The research and development program for the near term breeders is estimated at \$12.1 million. Of this amount, \$8.2 million would be required for a reactor experiment to establish the dynamic behavior of this reactor.

Development work is required in addition in the following areas:

1. Corrosion tests on Inconel and Inconel-X
2. In-pile irradiation tests of Inconel clad, $\text{PuO}_2\text{-UO}_2$ fuel elements.
3. Development of improved fabrication techniques for $\text{PuO}_2\text{-UO}_2$ fuel elements.
4. Improved knowledge of cross sections for Pu^{239} , Pu^{240} , and Pu^{241} over critical energy ranges.
5. Development of flooding poisons.
6. Development of low leakage, high capacity steam circulators.

The development program required to achieve the higher cladding temperature and increased burnup for the advanced reactors is estimated at \$3 million.

Fig. 2.1 — 300 MW_e steam-cooled fast breeder reactor — simplified flow diagram

Fig. 2.2 — 300 MW_e steam-cooled fast breeder reactor — elevation

3. NEAR TERM PLANTS

This section contains a description and cost estimates of the near term 300 MW_e and 40 MW_e plants. The reactor arrangement and design features of the two plants are similar in most respects. Therefore, a single description is used for both plant sizes. Differences between the two reactors are noted where they exist. The flow sheets and layout drawings of the two designs are shown at the end of Section 3, and are grouped separately for the 300 and 40 MW_e plants. The 300 MW_e plant drawings are shown in Figs. 3.3 to 3.10 and the 40 MW_e drawings are shown in Figs. 3.11 to 3.13.

The fast breeder reactor uses a direct cycle where about 1/3 of the superheated steam produced in the reactor is fed directly to the steam turbine. Most of the remaining steam, after passing through a desuperheater, is mixed with condenser feedwater in an open Loeffler boiler. The combined steam flow emerges from this boiler at saturation temperature and is returned to the reactor by means of low leakage steam circulators. The circulators of the 40 MW_e plant are motor driven while those of the 300 MW_e plant are powered by steam turbines (see Figs. 3.3 and 3.11).

Turbine inlet conditions for the 40 MW_e plant are 1400 psig, 920°F, and for the 300 MW_e plant, 1400 psig, 945°F. The turbine generator unit for the large size plant consists of a 316,000 kw tandem compound double flow, 43 in., 1800 rpm turbine directly connected to an 1800 rpm hydrogen-cooled generator having a rating of 371,760 kva, 0.85 power factor, 3 phase, 60 cycle, 20.0 kv, with 30 psig hydrogen cooling. A seven-stage regenerative feedwater cycle is used to

heat the feedwater to a final temperature of 495°F before it is returned to the Loeffler boiler drum.

A fully automatic steam dump system (turbine bypass) is provided for protection of the reactor plant on a sudden reduction in electric power load on the generator, or turbine-generator trip-out at partial or full load. Facilities include control valves, desuperheaters, and complete automatic controls and interlocks with the turbine-governor system.

The important components of the plant cycle are shown in the main power loop flow diagram, Fig. 3.4.

3.1 REACTOR ARRANGEMENT AND SHIELDING

The reactor cylindrical core and surrounding blanket, core supports, control rods, and thermal shielding are enclosed inside a pressure vessel (Figs. 3.5 and 3.6). The direction of coolant flow through core and blanket is downward. Reactor characteristics are listed in Table 3.1.

The core and blanket fuel subassemblies (Figs. 3.7 and 3.8) rest on a bottom grid structure which is secured to the core support cylinder. The core support cylinder is flanged at the top and rests on a shoulder recessed in the pressure vessel flange.

Each fuel element subassembly consists of a hexagonal cluster of rods surrounded by a hexagonal stainless steel shroud tube 0.100 in. thick. The core as well as the blanket subassemblies are 3.74 in. (3.54 in. for the 40 MW_e plant) across flats, so that interchange of core and blanket subassemblies is physically possible. A suitable grappling head is incorporated into the top end fittings for use with the refueling machine.

3.1.1 Core

Each core assembly contains three separate fuel clusters. (See Fig. 3.7.) The upper portion of each coolant shroud contains a cluster of axial blanket fuel elements with the core fuel elements in the center. The lower part of the coolant shroud also contains a cluster of axial blanket fuel elements. The coolant flows

Table 3.1 — Near Term Plant Characteristics

Description	40 MW _e	300 MW _e
Heat balance		
Total reactor power, MW _t	127	876
Gross turbine power, MW _e	46.8	316
Net plant power, MW _e	40	304
Net plant efficiency, %	31.5	34.7
Turbine cycle conditions		
Throttle temperature, °F	920	945
Throttle pressure, psia	1415	1415
Total steam flow, lb/hr	0.49×10^6	2.69×10^6
Condenser back-pressure, in. Hg abs.	1.5	1.5
Final feedwater temperature, °F	575	495
Number of feedwater heating stages	5	7
Reactor description		
Reactor vessel		
Inside diameter, ft	8.33	10.67
Inside height, ft	16.67	31
Wall thickness, (cylindrical portion), in.	5.85	7.5
Material, ASTM	Low alloy steel with $\frac{1}{4}$ in. SS-304 liner	
Design pressure, psig	1,860	1,860
Design temperature, °F	800	800
Reactor core		
Active equivalent diameter, ft	2.54	5.95
Active height, ft	2.65	4.96
Active core volume, ft ³	13.4	138
Total uranium loading, kg U + Pu	1483	13,460
Initial U ²³⁵ enrichment, w/o	Depleted	Depleted
Initial Pu ²³⁹ enrichment, kg Pu/ kg fuel	0.176*	0.123*
Moderator	None	None
Coolant	H ₂ O	H ₂ O
Blanket		
Material	Depleted UO ₂	Depleted UO ₂
Axial thickness, ft	1.5	1
Radial thickness, ft	1.5	1

*Values based on pure Pu²³⁹. See Table 3.4 for values at equilibrium cycle conditions.

Table 3.1 — (Continued)

Description	40 MW _e	300 MW _e
Core fuel element		
Fuel material	UO ₂ -PuO ₂	UO ₂ -PuO ₂
Fuel element geometry	Rods	Rods
Clad material	Inconel-X	Inconel-X
Fuel meat OD, in.	0.218	0.218
Clad thickness, in.	0.015	0.015
Fuel-clad gap (cold), in.	0.001	0.001
Gap filler material	Helium	Helium
Ligament spacing, in.	0.040	0.060
No. of rods per cluster	127	127
Active length, ft	2.65	5.0
Axial blanket fuel element		
Fuel material	UO ₂	UO ₂
Fuel element geometry	Rods	Rods
Clad material	Stainless steel	Stainless steel
Fuel rod OD, in.	0.437	0.405
Clad thickness, in.	0.015	0.015
Fuel-clad gap (cold), in.	0.001	0.001
Gap filler material	Helium	Helium
Ligament spacing, in.	0.088	0.144
No. of rods per cluster	37	37
Active length, ft	1.5	1.0
Radial blanket fuel element		
Fuel material	UO ₂	UO ₂
Fuel element geometry	Rods	Rods
Clad material	Stainless steel	Stainless steel
Fuel rod OD, in.	0.482	0.527
Clad thickness, in.	0.015	0.015
Fuel-clad gap (cold), in.	0.001	0.001
Gap filler material	Helium	Helium
Ligament spacing, in.	0.040	0.050
No. of rods per cluster	37	37
Active length, ft	5.65	7.0
Fuel assembly		
Total no. in core	67	326
Total no. in blanket	248	256
Shape	Hexagonal	Hexagonal

Table 3.1 — (Continued)

Description	40 MW _e	300 MW _e
Distance across flats, in.	3.54	3.74
Shroud material	Stainless steel	Stainless steel
Shroud thickness, in.	0.100	0.100
Lattice arrangements	Hexagonal	Hexagonal
Lattice spacing, in.	3.57	3.77
Reactor control		
Method of control	Rods	Rods
Absorber material	B ₄ C	B ₄ C
No. of shim rods	6 (2 in core, 4 in blanket)	6 (2 in core, 4 in blanket)
Shim rod effective length, ft	See Table 3.2	See Table 3.2
No. of safety rods	4	8
Safety rod effective length, ft	See Table 3.2	See Table 3.2
Control rod OD, in.	See Table 3.2	See Table 3.2
Control rod ID, in.	See Table 3.2	See Table 3.2
Type of drive	Induction motor, lead screw and nut	Induction motor, lead screw and nut
Core performance data		
Core coolant	Steam	Steam
Core coolant outlet temp, °F	925	950
Core coolant inlet temp, °F	610	610
Primary system operating pressure, psig	1550	1550
Primary coolant flow, lb/hr	1.48×10^6	9.44×10^6
Avg. core coolant velocity, ft/sec	120	107
Max. fuel center temp, °F	4500	4500
Max. cladding temp, °F	1350	1350
Max. core heat flux, Btu/hr-ft ² -°F	615,000	615,000
Avg. core heat flux, Btu/hr-ft ² -°F	254,000	201,000
Avg. core power density, kw _t /ft ³	8680	5930
Peak to avg. power ratio (nominal)	1.97	2.48
Avg. specific power, kw _t /kg fuel	79.5	62.0
Avg. specific power, kw _t /kg of Pu	454	505
Fuel management	Inward radial shift	Inward radial shift
Avg fuel burnup in exit fuel, MW-d/tonne	32,500	33,300
Peak to avg burnup ratio	1.54	1.50

Table 3.1 — (Continued)

Description	40 MW _e	300 MW _e
Radial blanket performance data		
Coolant	Steam	Steam
Coolant outlet temp, °F	850	875
Coolant inlet temp, °F	610	610
Coolant inlet pressure, psig	1550	1550
Coolant flow, lb/hr	0.15×10^6	0.87×10^6
Avg coolant velocity, ft/sec	14	20
Max. fuel center temp, °F	2920	2680
Max. cladding temp, °F	1250	1070
Max. heat flux, Btu/hr-ft ² -°F (perturbed)	137,000	131,000
Avg. heat flux, Btu/hr-ft ² -°F	9500	17,400
Fraction of power generated in blanket (beginning of life)	0.07	0.04
Peak to avg. power ratio (initial)	11.8	6.1
Fuel management	Inward shuffling	Inward shuffling
Avg. fuel burnup, MW-d/tonne	715	1150
Nuclear characteristics		
Critical mass, kg Pu ²³⁹	261*	1656*
Enrichment, a/o Pu ²³⁹	17.6*	12.3*
Total breeding ratio	1.42*	1.29*
Core breeding ratio	0.553	0.965
Blanket breeding ratio	0.867	0.325
Median fission energy, kev	160	112
Median absorption energy, kev	77	24
Containment		
Design criteria	Release of pri- mary coolant	Release of primary coolant
Type	Vessel	Vessel
Primary loop coolant inventory, lb	N.A.	160,000
Geometry	Cylinder and hemispherical dome	
Dimensions (OD, ft)	N.A.	125
Design pressure, psig	20	15
Material	$\frac{1}{2}$ in. steel shell	1 in. steel shell

*Values based on pure Pu²³⁹ and a fuel density of 10.0 g/cm³ or 91% of theoretical density. See Table 3.4 for values at equilibrium cycle conditions.

first over the upper axial blanket elements, then through the core elements, and finally through the lower blanket elements. The core fuel clusters supply 96% of the total heat transferred to the steam.

The number of fuel clusters in the core is based on a maximum steady state fuel temperature of 4500°F, and a maximum steady state clad temperature of 1350°F.

The core fuel elements are 0.25 in. diameter fuel rods, arranged in a regular triangular array. The fuel is a mixture of plutonium oxide and depleted (less than 0.35% U^{235}) uranium oxide. The fuel is clad with 0.015-in. thick Inconel-X tubes.

This alloy possesses excellent creep and yield strength properties at 1350°F. Typical strength data for Inconel and Inconel-X are plotted in Figs. 8.1 through 8.4. Corrosion data with supercritical steam at 5000 psi and 1350°F showed Inconel-X to lose less than 1 mil/yr.

Space is provided at the top of each fuel rod to allow for axial expansion of the fuel and to minimize the pressure caused by the release of fission product gases. The 300 MW_e reactor requires a gas space height of about 3 in. while the 40 MW_e reactor requires a gap height of only 1³/₄ in. These dimensions are based on release of 60% of the fission gases, an allowable cladding stress of 20,000 psi and a 9% void space in the fuel. Spacing between adjacent fuel elements is maintained by means of Inconel spacer wires wound in an open spiral around the individual fuel rods. The rod assembly stands on a support shelf in the fuel channel.

The axial blanket fuel elements of the 300 MW_e plant are 0.405 in. diameter fuel rods. The fuel is depleted uranium oxide enclosed in stainless steel tubing. The spacing between rods is maintained by means of ferrules which are brazed to the fuel rods at regular intervals.

3.1.2 Radial Blanket

The blanket of the 300 MW_e reactor is 1 ft thick, and that of the 40 MW_e reactor is 1.5 ft (Figs. 3.6 and 3.13). The radial blanket fuel element sub-

assembly consists of 37-rod clusters of 0.527 in. diameter fuel rods spaced 0.050 in. apart and is shown in Fig. 3.8. The fuel is depleted uranium oxide clad with 0.015-in. thick stainless steel tubing. The spacing between adjacent fuel rods is maintained by means of spacer wires.

3.1.3 Control Rods

There are six shim rods and eight safety rods available for control of the reactor (six shim rods and four safety rods for the 40 MW_e reactor). The rods are made of B₄C clad with stainless steel tubing and are described in Table 3.2.

The control rods are driven from above by a constant speed motor supported on top of the top shield plug. The rod is coupled to the drive through a safety magnet. Reactivity is added to the core by driving the control rod up and out of the core. The safety rods are raised to the upper limit prior to startup and left in this position during operation of the plant. These rods are at the maximum potential for removing reactivity by de-energizing the safety magnets in a reactor scram. Control of the rod drive motors in this group is entirely manual. The shim rods are controlled manually to add reactivity during startup, and to provide shimming action during operation to compensate for changes in reactivity caused by fuel burnup. These rods are released automatically to drop into the core when a reactor scram is required.

3.1.4 Pressure Vessel and Internals

The pressure vessel for this reactor is carbon steel clad with stainless steel (Figs. 3.5 and 3.12). The upper head is removable and is provided with bolted flanges and sealed by a metallic gasket.

The vessel is supported near the bottom. Large flow connections are made in this region. The upper head is penetrated by refueling ports and openings for control rods. The coolant enters via four inlet lines, equally spaced about the periphery near the bottom of the vessel, cooling the cylindrical thermal shields and the core support cylinder as it passes upward. The steam reverses

Table 3.2 — Control Rods — B_4C — 50% Enriched with B^{10}

Plant Size	Rod Type	Design	Active Length, cm	Worth Per Rod, \$	No. of Control Clusters	kg B^{10} Per Rod
300 MW _e	Shim rods in core	7 rod cluster of 0.25 in. diam B_4C clad with SS	120	0.48	2	0.23
	Safety rods in core	7 rod cluster, 0.4 in. diam B_4C	150	1.58	8	0.72
	Shim rods in blanket	7 rod cluster, 0.25 in. diam B_4C	60	0.25	4	0.06
40 MW _e	Shim rods in core	7 rod cluster, 0.37 in. diam B_4C	60	0.48	2	0.12
	Safety rods in core	7 rod cluster, 0.4 in. diam B_4C	80	0.83	4	0.20
	Shim rods in blanket	7 rod cluster, 0.25 in. diam B_4C	60	0.25	4	0.06

direction, entering the region above the core through inlet ports in the core support cylinder. It then flows down through the core and radial blanket fuel assemblies in parallel paths, then through the lower grid plate where the parallel streams rejoin. The openings in the lower grid plate provide orificing to obtain a uniform outlet temperature from all channels. The steam leaves the core through a single discharge line located in the bottom of the vessel.

3.1.5 Thermal Shields

Thermal shields are placed between the blanket and the pressure vessel for the protection of the latter. They are designed to meet the requirement that the integrated neutron flux at the inner pressure vessel wall, of neutrons having energies 1 Mev and above, is not to exceed 10^{19} nvt in 30 yr at 80% plant factor. All thermal shielding is stainless steel, cooled with steam.

Both 300 and 40 MW_e plants have approximately the same fast neutron (≥ 1 Mev) leakage per unit outer surface area of blanket. Thus the radial as well as the bottom axial shielding requirement is the same for both. The top shield in the 300 MW_e plant, however, is somewhat thinner because of the greater attenuation in the steam which takes place between the top blanket and shield surfaces.

The thermal shielding requirements for the 300 and 40 MW_e plants, respectively, are summarized below:

	Total Thickness of Steel	
	300 MW _e Plant	40 MW _e Plant
Radial Shield	1 ft	1 ft
Top Axial Shield	5 in.	7 in.
Bottom Axial Shield	5 in.	5 in.

3.1.6 Reactor Support

The reactor vessel has support brackets equally spaced around the periphery, in close proximity to the hemispherical base of the vessel. A system of steel framework is provided to transfer the weight of the reactor to steel columns.

The base of the vessel is supported on this steel framework, with provisions for radial expansion, while maintaining accurate horizontal alignment. The upper end of the vessel is guided for vertical expansion and alignment without introducing additional stresses in the vessel wall or binding of the control rods.

3.1.7 Reactor Shield

A biological shield is provided around the reactor to prevent radiation levels from exceeding acceptable tolerance in areas accessible to maintenance and operating personnel (Fig. 3.9). The reactor pressure vessel is surrounded by a reinforced concrete biological shield with vertical side walls and flat biological shields above and below the vessel. The side walls are octagonal and are 7 ft thick. Penetrations through the shielding are provided for shield ventilation and coolant pipes. The top shield is separated from the rest of the shield by stepped expansion joints, allowing free lateral expansion of this shield. These steps prevent radiation streaming and lock the shield in place.

3.1.8 Refueling Vehicle

The refueling vehicle is located above the neutron shield plug (Fig. 3.9) and moves on a crane-rail supported by the biological shield structure surrounding the reactor vessel. The vehicle is unshielded and is remotely controlled. It contains a grappling mechanism which can be lowered into the reactor through the refueling ports. The grappling head can be positioned above the respective fuel element and lowered to pick up the fuel element (Fig. 3.12). The vehicle is also equipped with a wrench-type device to open and remove the refueling port closure plug. The refueling operation is performed with the reactor cavity flooded and is described in Section 4.3.

3.1.9 Shielding of External Equipment

Since the reactor coolant is also the working fluid of the steam turbine, the radioactivity associated with the steam will be carried to the turbine and other external equipment. A brief examination of possible radioactivity in the coolant has been made, and the radiation dose rates together with the shielding

required at various points in the system have been estimated.

One major source of radioactivity is N^{16} resulting from (n,p) reaction with O^{16} in the very fast neutron field of the reactor. In considerably smaller amounts neutron-emitting N^{17} is also formed. Although activity levels from this source may be significant, the emitted neutrons can be stopped in very small concrete thicknesses. N^{16} and N^{17} have comparatively short half-lives (7.4 and 4.1 sec, respectively); therefore, rapid decay of this activity results immediately following reactor shutdown. The dose rates due to N^{16} and N^{17} disintegrations are considerably higher for the fast breeder than for a thermal reactor of the same power. The basic reason for this is that higher specific powers in the breeder lead to much larger very fast neutron fluxes.

Another major source of radioactivity is due to deposition of activated solids in the external system. The activity from this source depends on variables such as corrosion rates, materials of construction, amount of entrained solids which pass through the reactor along with the coolant, residence time of the coolant in the reactor core, and the values of the activation cross sections $a(n, \gamma)b$ (where a is the target nuclide in the core and b is the nuclide whose decay produces the external system activation). Relative to thermal reactors, the fast breeders would be expected to have smaller amounts of radioactivity produced by this source. This is true because the breeders, in general, have shorter core residence times, and because of the higher neutron energy spectrums, the activation cross sections are also smaller.

Table 3.3 gives the estimated dose rates due to N^{16} , N^{17} , and active deposits, as well as estimates of the thickness of 150 lb/ft³ concrete shielding required. It will be noted that the deposit activity dose rates do not reach significant levels even after 5 years of reactor operation.

3.1.10 Plant Arrangement

The general arrangement of the nuclear power plant is shown in Figs. 3.9 and 3.10.

Table 3.3 — Dose Rates at the External Surface, and Shielding Requirement of Coolant Cycle Equipment

	Turbine	Condenser	Hot Well	Boilers
N ¹⁶ mr/hr	352	—	497	10.3
N ¹⁷ n/cm ² -sec	5.9	—	1.4	0.5
N ¹⁷ mr/hr	0.6	—	0.1	0.05
Deposit Activity, mr/hr				
1 mo	—	—	—	—
1 yr	0.002	0.0006	—	0.0001
5 yr	0.028	0.083	—	0.0008
Required Concrete* Thickness, ft (to reduce dose to 1 mr/hr)	3.5	2.3†	3.8	1.1

*150 lb/ft³ concrete.

†At the bottom of the condenser.

Plant equipment is housed in two main buildings. The reactor plant and auxiliaries are enclosed with a concrete lined vertical steel shell building, and the turbine-generator and auxiliary equipment are located in a high bay metal siding building.

The reactor building is a concrete lined vertical cylindrical steel shell fitted with a hemispherical top head. The vessel, 120 ft ID by 158 ft overall height, will contain any uncontrolled release of the reactor coolant. A fuel handling pool is provided within the containment vessel, hence a separate fuel handling building is not required.

The turbine-generator building houses the turbine-generator unit and condenser, and all related equipment, except the main auxiliary transformers and the startup transformer. This building is constructed of reinforced concrete up to and including the main operating floor. Foundations are spread footings bearing on bedrock. The building superstructure is a structural steel frame with insulated metal siding, and insulated metal roof deck with built-up roofing.

A high bay, located near the center of the structure contains the turbine-generator unit. An all-electric traveling bridge crane spans the high bay and serves the turbine-generator and auxiliaries.

The mezzanine floor provides space for the feedwater heaters, high voltage switchgear and motor control centers, shop, air ejector equipment, fan rooms, new fuel storage, offices, locker rooms, health physics, and chemical laboratory. The ground and basement floor levels provide space for low voltage switchgear, battery room, the emergency generator, the condensate demineralizer system, make-up demineralizer system, feedwater pumps, and other turbine-generator auxiliary equipment. A railroad car and truck entrance into the building is provided on the ground floor level. Condensate and demineralized water storage tanks are located on the roof over the feedwater heater area.

Building services include heating and ventilation systems, air-conditioning in the control room, offices and laboratory, plumbing, lighting and electrical services, fire protection and painting.

3.2 NUCLEAR CHARACTERISTICS

The nuclear characteristics for the 300 and 40 MW_e steam-cooled fast breeder reactors were calculated by 10-group diffusion theory using a one-dimensional, spherical reactor model.² Suitable correction factors were applied to allow for deviations from the actual cylindrical geometry. Table 3.1 presents the significant results of the nuclear analysis for the cold, clean condition and correspond to a plutonium loading of pure Pu²³⁹ isotope.

3.2.1 Critical Mass

An initial critical mass of 1656 kg of Pu²³⁹ is predicted for the 300 MW_e oxide breeder. This value corresponds to a specific power of 0.50 MW_t/kg Pu²³⁹ which is about one-half the specific power reported for sodium-cooled fast breeders. A substantial reduction of fuel fabrication cost is obtained in conjunction with steam-cooled breeders because of the lower degree of fuel subdivision required. The lower specific power of the steam-cooled breeder is mainly the result of coolant heat transfer requirement, but is also influenced by the spectrum degradation effects of hydrogen in the core.

3.2.2 Breeding Ratio

The initial core breeding ratio for the 300 MW_e oxide design is near unity, because of a relatively large core and hence a relatively low core leakage compared with smaller size breeders. The high core breeding ratio is advantageous in that it leads to relatively long refueling intervals, and good fuel economics. The 40 MW_e oxide breeder has a somewhat higher total breeding ratio than the 300 MW_e version mainly because of a large blanket thickness. In both reactors the total breeding ratios are not appreciably different from values reported in the literature for sodium-cooled fast breeders.³

3.2.3 Neutron Spectra

Figs. 3.1 and 3.2 present core flux per unit lethargy vs lethargy plots for the 300 and 40 MW_e breeders. The 40 MW_e reactor has a somewhat harder spectrum, because of the higher enrichment required in the smaller reactor to

compensate for the large core leakage.

Fig. 3.1 shows a plot of a spectrum for a sodium-cooled fast $\text{PuO}_2\text{-UO}_2$ breeder⁴ for comparison with the spectrum of the oxide reference design. The major difference between the spectrum of the steam-cooled and the sodium-cooled designs lies in the neutron population of the lower energy groups, where a neutron density as much as 3 times higher is observed for the steam-cooled reactors.

3.2.4 Equilibrium Fuel Cycle

The critical mass and breeding ratio figures given above and in Table 3.1 are based on a core freshly fueled with $\text{Pu}^{239}\text{O}_2\text{-U}^{238}\text{O}_2$. As the discharged fuel is recycled through the reactor a buildup of higher isotopes of plutonium takes place.

Critical mass and breeding ratios for an equilibrium cycle of the 300 MW_e reactor are presented in Table 3.4. A decrease in the critical mass of fissionable isotopes and an increase in total breeding ratio (based on fissionable isotopes) are indicated over values calculated on the basis of pure Pu^{239} loading.

The equilibrium fuel containing higher plutonium isotopes has an average η of 1.55 compared with a value of 1.23 for the 12.3% enriched fuel discussed above. The presence of the higher isotopes enables a reduction in the $\text{Pu}^{239}/\text{U}^{238}$ ratio from 0.140 to 0.115. This is due to the buildup of Pu^{241} , which has an average η of 2.89 compared with 2.26 for Pu^{239} . The decreased enrichment of fissile isotopes produces a considerable improvement in the overall breeding ratio.

In the calculation of the equilibrium concentrations of plutonium isotopes, the relative amounts of these isotopes present in newly refabricated fuel were calculated on the basis of

1. a core fuel cycle of 3.2 years,
2. an average neutron flux in the core of 3×10^{15} per cm²-sec,
3. a blanket consisting only of U^{238} fuel at the start of life.

3.2.5 Effect of U^{235} Present in Fuel

The nuclear characteristics shown in Table 3.1 were obtained neglecting the U^{235} present in depleted UO_2 . A separate ten-group diffusion theory calcu-

Table 3.4 — Nuclear Properties of 300 MW_e Breeder
Reactor under Equilibrium Refueling Conditions

Critical Mass, * total kg of Pu ²³⁹ and Pu ²⁴¹	1441	
Composition of Fuel, a/o	<u>Initial</u>	<u>Final</u>
U ²³⁸	84.6	83.37
Pu ²³⁹	9.76	10.74
Pu ²⁴⁰	4.44	4.60
Pu ²⁴¹	0.94	1.03
Pu ²⁴²	0.25	0.26
Median Fission Energy, kev	~101	
Initial Breeding Ratios†		
Core	1.165	
Blanket	<u>0.349</u>	
Total	1.514	

*Based on fuel density of 10.0 g/cm³

†Defined as atoms of Pu²³⁹ and Pu²⁴¹ produced per atoms of Pu²³⁹
and Pu²⁴¹ destroyed.

lation was performed in order to evaluate the effect of the presence of U^{235} in so-called depleted uranium. A ratio of 0.0037 atoms of U^{235} to 1 atom of U^{238} was assumed. The important nuclear characteristics at the start of core life of the 300 MW (1967) reactor, including the effects of this U^{235} , are given below:

1. Critical mass, kg Pu^{239}	1616
2. Enrichment, a/o Pu^{239}	12.0
3. Initial breeding ratio*	
Core	0.99
Blanket	<u>0.33</u>
Total	1.32
4. Median fission energy, kev	109
5. Median absorption energy, kev	24

The presence of U^{235} is seen to correspond to a decrease in enrichment of 0.25% and an increase in breeding ratio of 0.03.

3.2.6 Effect of Fuel Density

On the basis of the flux spectrum obtained from multigroup calculations, a one-group cross-section set was prepared and used to evaluate the effect of fuel density on the initial critical mass and breeding ratios of the 300 MW_e reactor. The purpose of this effort is to provide a background for determining the merit of fuel fabrication methods, such as vibratory compaction, which yield relatively low fuel densities. The results are presented in Figs. 3.14 and 3.15. It will be noted that the critical mass decreases with a decrease in fuel density, while plutonium enrichment increases. The total breeding ratio is seen to decrease very slightly with decreasing fuel density.

3.2.7 Doubling Time

An overall system doubling time of 29 years is obtained for the 300 MW_e near term reactor and 24 years for the 40 MW_e reactor. The overall system

*Defined as atoms Pu^{239} produced per atom of Pu^{239} destroyed in the core.

doubling time is defined as the number of full power years of operation required to double the in-pile and out-of-pile fissionable material inventory including material tied up in reprocessing and fuel fabrication. This quantity is a function of the breeding gain, burnup rate, reactor specific power and the ratio of total to in-pile fissionable material inventory, as may be seen from the following equation.

$$T = \frac{E}{G B P_s}$$

where T = doubling time

E = kg Pu total/kg Pu in-pile

G = breeding gain, $\frac{\text{net kg Pu produced}}{\text{kg Pu consumed}}$

B = burnup rate, $\frac{\text{kg Pu consumed}}{\text{MW-yr}}$

P_s = core specific power, MW/kg Pu in-pile.

The ratio of total to in-pile inventory, E, was estimated at 1.46 for the 300 MW_e reactor.

3.3 COST ESTIMATES

Cost estimates including capital cost, fixed charges, fuel cycle and maintenance costs for the 300 and 40 MW_e reactor plants are presented in Tables 3.5 to 3.13. A summary of power generation costs is given in Table 3.5.

The data presented in these tables are in agreement with the final revisions and adjustments made by the USAEC and Kaiser Engineers in the effort of presenting the cost of all reactor concepts included in the Steam-Cooled Power Reactor Evaluation on a common basis.⁵

3.3.1 Fixed Charges, Near Term Plants

Fixed charges include annual capital charges at 14% of capital investment, working capital charges at 12.5% working capital requirement, interim replacement charges at 0.3% of capital investment, and nuclear insurance.

Working capital is estimated at 2.7% of the annual operating expense plus

Table 3.5 — Near Term Plants — Summary of Power Generation Costs

	40 MW _e		300 MW _e	
Plant Design				
Gross electric power rating, MW	46.8		316	
Net electric power rating, MW	40		304	
Plant operating factor	0.80		0.80	
Average burnup, MW-d/tonne U	32,500		33,300	
Capital Cost				
Total capital cost, \$	24,130,000	62,000,000		
Unit capital cost, \$/net, kw	603	204		
First core fabrication cost, \$	1,627,000	6,098,000		
Working capital, \$	1,047,000	3,915,000		
	Annual Cost, \$	Mills Per Net kwh	Annual Cost, \$	Mills Per Net kwh
Power Generation Cost				
Investor owned public utility				
Fixed charges				
Capital cost, 14%	\$ 3,378,000	12.05	\$ 8,680,000	4.07
Working capital, 12.5%	131,000	0.47	489,000	0.23
Interim replacement, 0.3%	72,000	0.26	187,000	0.09
Nuclear insurance	<u>200,000</u>	<u>0.71</u>	<u>286,000</u>	<u>0.13</u>
Total fixed charges	\$ 3,781,000	13.49	\$ 9,642,000	4.52

Table 3.5 — (Continued)

	40 MW _e		300 MW _e	
	Annual Cost, \$	Mills Per Net kwh	Annual Cost, \$	Mills Per Net kwh
Operating Cost				
Operation and maintenance	\$ 507,000	1.81	\$ 878,000	0.41
Nuclear fuel cycle				
Fabrication	591,000	2.11	2,940,000	1.38
Shipping	74,000	0.26	261,000	0.13
Depletion	0	0	0	0
Reprocessing	511,000	1.82	2,409,000	1.11
AEC use charge	188,000	0.67	1,540,000	0.72
Plutonium credit	<u>(200,000)</u>	<u>(0.71)</u>	<u>(1,570,000)</u>	<u>(0.72)</u>
Total fuel cycle cost	<u>1,164,000</u>	<u>4.15</u>	<u>5,580,000</u>	<u>2.62</u>
Total operating cost	1,671,000	5.96	6,458,000	3.03
Total Power Generation Costs	\$ 5,452,000	19.45	\$ 16,100,000	7.55

Note: Fuel costs for 300 MW_e plant are based on enrichments that include higher Pu isotopes generated under equilibrium conditions. Costs for 40 MW_e plant are based on Pu as pure Pu²³⁹.

the average value of materials, supplies, and fuel inventory. The annual operating expense includes all the yearly fuel cycle cost, and insurance, operation, and maintenance costs. The average investment in nuclear fuel inventory is assumed to equal 60% of the cost of fabricating one full reactor loading of fuel. The annual cost for the replacement of control rod absorber sections is included in the allowance for interim replacements which is calculated as a fixed percentage (0.3%) of the total capital cost of the plant.

The plant capital cost is computed at current price levels and is representative of plants that could be constructed and operated by June 30, 1967, upon the successful completion of the required research and development program. A summary and detailed breakdown of capital costs is given in Table 3.6.

Costs of reactor plant equipment (Account 22) have been carefully estimated on the basis of layout drawings and preliminary system specifications. Manufacturer's quotations have been obtained for those items where a large degree of uncertainty in cost existed. These costs were reviewed by Kaiser Engineers and the AEC, and the figures given in Table 3.6 are in agreement with those reported to the AEC by Kaiser Engineers.⁵ Accounts 21, 23, 24, and 25 were based on estimates made mainly by Kaiser Engineers. The cost of land and land rights and charges for general and administrative expenses, engineering design, and inspection, contingency and interest during construction are those adapted in the AEC ground rules for the Steam-Cooled Power Reactor Evaluation.⁵

3.3.2 Fuel Costs, Near Term Plants

Core

Fuel costs were calculated for the fast breeder reactors using AEC ground rules established for the Steam-Cooled Power Reactor Evaluation. Nuclear fuel cycle costs are given in Tables 3.7 through 3.9 for the 300 MW_e plant, and in Tables 3.10 through 3.12 for the 40 MW_e plant.

It was assumed that plutonium can be leased from the AEC at 4% per annum in the form of plutonium oxide at \$11.5/g Pu, and the cost of depleted uranium was taken as \$5/kg.

Table 3.6 — Near Term Plants — Estimate of Capital Cost — Summary

Acct. No.	Description	Estimated Cost, \$1000	
		40 MW _e	300 MW _e
	<u>Direct construction cost</u>		
21	Structure and improvements	3,249	5,100
22	Reactor plant equipment	6,945	16,971
23	Turbine-generator unit	3,209	13,179
24	Accessory electric equipment	780	1,850
25	Miscellaneous power plant equipment	<u>206</u>	<u>486</u>
	Total direct construction cost	14,389	37,586
	<u>Indirect construction cost</u>		
	General and administrative*	<u>2,011</u>	<u>4,694</u>
	Subtotal	16,400	42,280
	Engineering, design, and inspection*	2,620	6,170
	Nuclear engineering	1,500	3,000
	Startup costs	190	330
20	Land and land rights	<u>100</u>	<u>360</u>
	Subtotal	20,810	52,140
	Contingency, 10%	<u>2,080</u>	<u>5,210</u>
	Subtotal	22,890	57,350
	Interest during construction*	<u>1,240</u>	<u>4,650</u>
	Total capital cost	24,130	62,000
	Unit capital cost per net kw	603	204
20	Land and land rights		
	201 - Land and privilege acquisition	100	360
21	Structures and improvements		
	211 - Ground improvements	478	768
	212 - Buildings	1,231	1,920
	218 - Main stack	60	75
	219 - Reactor container structures	<u>1,480</u>	<u>2,337</u>
	Total account No. 21	3,249	5,100
*Note:		40 MW _e	300 MW _e
	General and Administrative, %	14.0	12.5
	Engineering, Design and Inspection, %	16.0	14.6
	Interest During Construction, %	5.4	8.1

Table 3.6 — (Continued)

Acct. No.	Description	Estimated Cost, \$1000	
		40 MW _e	300 MW _e
22	Reactor plant equipment		
	221 - Reactor equipment		
	.1 Reactor vessel	872	2,750
	.2 Reactor controls	500	800
	.3 Reactor primary shield	367	387
	Other reactor equipment	<u>228</u>	<u>603</u>
	Total subaccount No. 211	1,967	4,540
	222 - Heat transfer system	1,570	6,645
	223 - Fuel handling and storage equipment	750	1,135
	225 - Radioactive waste treatment and disposal	225	300
	226 - Instrumentation and control	1,100	1,235
	227 - Feedwater supply and treatment system	433	1,445
	228 - Steam, condensate, and feedwater piping	800	1,471
	229 - Miscellaneous reactor plant equipment	<u>100</u>	<u>200</u>
	Total account No. 22	6,945	16,971
23	Turbine-generator unit		
	231 - Turbine-generator	2,407	11,067
	232 - Circulating water system	353	690
	233 - Condenser	320	1,190
	235 - Turbine plant boards, instruments and controls	70	125
	236 - Turbine plant piping (See also Acct. No. 228)	9	24
	237 - Auxiliary equipment for generator	40	64
	238 - Other turbine plant equipment	<u>10</u>	<u>19</u>
	Total account No. 23	3,209	13,179

Table 3.6 — (Continued)

Acct. No.	Description	Estimated Cost, \$1000	
		40 MWe	300 MWe
24	Accessory electric equipment		
	241 - Switchgear	131	148
	242 - Switchboards	117	487
	243 - Protective equipment	41	57
	244 - Electrical structures	48	142
	245 - Conduit	50	128
	246 - Power and control wiring	153	460
	247 - Station service equipment	<u>240</u>	<u>428</u>
	Total account No. 24	780	1,850
25	Miscellaneous power plant equipment		
	251 - Cranes and hoisting equipment	112	242
	252 - Compressed air and vacuum cleaning system	20	33
	253 - Other miscellaneous equipment	<u>74</u>	<u>211</u>
	Total account No. 25	206	486
	Total direct construction cost*	14,389	37,586

*Excludes account No. 20 - land and land rights.

Table 3.7 — 1967, 300 MW_e Plant — Summary of Nuclear Fuel Cycle Costs

	300 MW _e	
	Annual Cost, \$	Mills/kwh
Fabrication	2,940,000	1.38
Spent Fuel Shipping	261,000	0.13
Depletion	0	0
Reprocessing	2,409,000	1.11
AEC Use Charge	1,540,000	0.72
Plutonium Credit	<u>(1,570,000)</u>	<u>(0.72)</u>
Total	5,580,000	2.62
First Core Fabrication Cost	6,098,000	

- Notes:
1. Fuel cycle costs are based on an average exposure of 33,000 MW-d/tonne U.
 2. Fabrication includes cost of conversion from UF₆; losses during conversion and fabrication, shipping and storage on site.
 3. Reprocessing includes separation, conversion of UNH to UF₆, conversion of PuNH to oxide and losses.
 4. AEC use charges include use charge during spent fuel storage, shipping and processing; during in-core residence; and for spare fuel on hand.

Table 3.8 — 1967, 300 MW_e Plant – Nuclear Fuel Cycle Cost Data

Description	300 MW _e	
	Core	Blanket
Total net power output, MW _e	304	
Average plant operating factor	0.8	
Average thermal power output, MW _t	841	35
Average burnup, MW-d/tonne	33,000	1,152
Initial fuel loading (U + Pu), kg	13,460	26,690
Final fuel loading (U + Pu), kg	13,020	26,630
Initial U ²³⁵ enrichment, w/o	Depleted	Depleted
Initial Pu concentration, w/o	15.39*	-
Final plutonium concentration, w/o	16.47*	1.10
Conversion charge from UF ₆ , \$/kg	10.00	10.00
Fuel fabrication price, \$/kg	240.00	58.00
Shipping and insurance charges, new fuel, \$/kg	3.00	3.00
Total recoverable scrap, %	10	10
Irrecoverable loss in conversion, %	0.8	0.8
Irrecoverable loss in fabrication, %	1.2	1.2
Fuel fabrication time, months	1.5	1.5
Core fabrication time, months	2.0	2.0
New fuel shipping time, including storage on site, months	3.0	3.0
Spent fuel decay storage time, including shipping, months	4.0	4.0
Spent fuel shipping and insurance charges, \$/kg	16.00	16.00
Plutonium value, \$/g	11.50	11.50
Value of uranium at initial enrichment, \$/kg U	5.00	5.00
Value of uranium at final enrichment, \$/kg U	5.00	5.00
Uranium burnup cost, \$/kg U	0	0
Plutonium credit, \$/kg U + Pu	61.70	124.90
Average fuel throughput rate, kg U + Pu/month	309	248
Chemical processing batch size, kg U + Pu	6,510	8,786
Chemical separation rate, kg/day	433	1,000
Chemical separation turnaround time, days	8	8
Total chemical reprocessing time, days	93.6	52.6
Spare fuel on hand, kg U + Pu	3,090	3,720

*Includes Pu²³⁹, Pu²⁴⁰, Pu²⁴¹, Pu²⁴².

Table 3.9 — 1967, 300 MW_e Plant — Estimate of Nuclear Fuel Cycle Costs

Cost Item	Cost, \$/kg 300 MW _e	
	Core	Blanket
Fabrication		
Cost of conversion from UO ₂	11.20	11.20
Cost of fissionable material lost in conversion	0	0
Use charge during conversion	0	0
Cost of fuel fabrication	240.00	58.00
Cost of fissionable material lost in fabrication	23.70	0.10
Use charge during fabrication	19.70	0.10
Cost of shipping new fuel	3.00	3.00
Use charge during shipping and storage (new fuel)	11.80	-
Total fabrication cost	309.40	72.40
Spent fuel shipping	16.00	16.00
Depletion	0	0
Reprocessing		
Chemical separation cost	56.10	31.00
Conversion of UNH to UF ₆	4.50	5.50
Conversion of PuNH to oxide	167.70	10.70
Chemical processing losses	36.70	2.60
Total reprocessing cost	265.00	49.80
AEC use charge		
Use charges during spent fuel storage, shipping, and processing	44.00	0.10
Use charges during fuel in-core residence	131.10	0.60
Use charges for spare fuel on hand	31.70	0.10
Total use charges	206.80	0.80
Plutonium credit	(61.70)	(124.90)
Total	735.50	14.10
Yearly throughput rate, kg/yr	3,708	2,970
Annual fuel cycle cost per replacement batch	2,727,000	42,000
Replacement batches per core	2	3
Total annual fuel cycle cost	5,454,000	126,000
Unit fuel cycle cost, mills/kwh	2.56	0.06
First core cost	4,166,000	1,932,000

Table 3.10 — 1965, 40 MW_e Plant — Summary of Nuclear Fuel Cycle Costs

	40 MW _e	
	Annual Cost, \$	Mills/kwh
Fabrication	591,000	2.11
Spent Fuel Shipping	74,000	0.26
Depletion	0	0
Reprocessing	511,000	1.82
AEC Use Charge	188,000	0.67
Plutonium Credit	<u>(200,000)</u>	<u>(0.71)</u>
Total	1,164,000	4.15
First Core Fabrication Cost	1,627,000	

- Notes:
1. Fuel cycle costs are based on an average exposure of 32,500 MW-d/tonne U.
 2. Fabrication includes cost of conversion from UF₆, losses during conversion and fabrication, shipping, and storage on site.
 3. Reprocessing includes separation, conversion of UNH to UF₆, conversion of PuNH to oxide and losses.
 4. AEC use charges include use charge during spent fuel storage, shipping, and processing; during in-core residence; and for spare fuel on hand.

Table 3.11 — 1965, 40 MW_e Plant — Nuclear Fuel Cycle Cost Data

Description	40 MW _e	
	Core	Blanket
Total net power output, MW _e	40	
Average plant operating factor	0.8	
Average thermal power output, MW _t	118	9
Average burnup, MW-d/tonne	32,500	715
Initial fuel loading (U + Pu), kg	1,482	15,950
Final fuel loading (U + Pu), kg	1,434	15,771
Initial U ²³⁵ enrichment, w/o	Depleted	Depleted
Initial Pu Concentration, w/o	17.6	0
Final plutonium concentration, w/o	16.3	1.05
Conversion charge from UF ₆ , \$/kg	10.00	10.00
Fuel fabrication price, \$/kg	240.00	58.00
Shipping and insurance charges, new fuel, \$/kg	3.00	3.00
Total recoverable scrap, %	10	10
Irrecoverable loss in conversion, %	0.8	0.8
Irrecoverable loss in fabrication, %	1.2	1.2
Fuel fabrication time, months	1.5	1.5
Core fabrication time, months	3.0	3.0
New fuel shipping time, including storage on site, months	2.0	2.0
Spent fuel decay storage time, including shipping, months	4.0	4.0
Spent fuel shipping and insurance charges, \$/kg	16.00	16.00
Plutonium value, \$/kg	11.50	11.50
Value of uranium at initial enrichment, \$/kg U	5.00	5.00
Value of uranium at final enrichment, \$/kg U	5.00	5.00
Uranium burnup cost, \$/kg U	0	0
Plutonium credit, \$/kg U + Pu	(212.70)	(119.40)
Average fuel throughput rate, kg U + Pu/month	43.8	98.7
Chemical processing batch size, kg U + Pu	716	5,257
Chemical separation rate, kg/day	261	1,000
Chemical separation turnaround time, days	2.8	5.3
Total chemical reprocessing time, days	41.5	45.6
Spare fuel on hand, kg U + Pu	438	1,482

Table 3.12 — 1965, 40 MW_e Plant — Estimate of Nuclear Fuel Cycle Costs

Cost Item	Cost, \$/kg 40 MW _e	
	Core	Blanket
Fabrication		
Cost of conversion from UO ₂	11.20	11.20
Cost of fissionable material lost in conversion	0	0
Use charge during conversion	0	0
Cost of fuel fabrication	240.00	58.00
Cost of fissionable material lost in fabrication	27.10	0.10
Use charge during fabrication	22.60	0.10
Cost of shipping new fuel	3.00	3.00
Use charges during shipping and storage, new fuel	13.50	0.10
Total fabrication cost	317.40	72.50
Spent fuel shipping	16.00	16.00
Depletion	0	0
Reprocessing		
Chemical separation cost	122.70	32.60
Conversion of UNH to UF ₆	4.40	5.50
Conversion of PuNH to oxide	154.40	10.30
Chemical processing losses	33.30	2.40
Total reprocessing cost	314.80	50.80
AEC use charge		
Use charges during spent fuel storage, shipping and processing	32.90	0.10
Use charges during fuel in-core residence	108.10	0.90
Use charges for spare fuel on hand	33.70	0.10
Total use charges	174.70	1.10
Plutonium credit	<u>+212.70</u>	<u>(119.40)</u>
Total	1,036.60	21.00
Yearly throughput rate, kg/yr	525.6	1184.4
Annual fuel cycle cost per replacement batch	545,000	25,000
Replacement batches per core	2	3
Total annual fuel cycle cost	1,089,000	75,000
Unit fuel cycle cost, mills/kwh	3.88	0.27
First core cost	470,000	1,156,000

Fuel fabrication costs for the near term designs using plutonium are taken as twice the average value of present day stainless steel clad UO_2 fuel element fabrication costs. The cost of a 0.25 in. diameter PuO_2 - UO_2 Inconel-X clad fuel rod was estimated at \$240/kg of fuel. The higher fabrication cost of plutonium-fueled elements allows for the increase in fabrication cost resulting from the use of controlled atmosphere glove boxes in all fabrication steps.

According to recent information from Hanford a fabrication cost of about \$125/kg of plutonium-uranium oxide fuel for 0.25 in. fuel rods is likely to be achieved by 1967. To be on the conservative side, the higher price was used. The use of Inconel-X cladding in the place of stainless steel does not appreciably influence the fabrication costs.

Credit for plutonium is obtained by considering both the initial and final plutonium enrichments, using a value of \$11.50/g of plutonium as PuO_2 . A spare fuel inventory equivalent to 15% of the core inventory is assumed.

Conversion losses, scrap recovery, fabrication time, new fuel shipping time including storage, spent fuel decay storage time, and shipping time are assumed to be the same as for uranium fuel elements. Chemical separation rates for plutonium containing fuel are based on chemical separation rate schedules published by the AEC⁵ for U^{235} . Plutonium enrichment is used in place of U^{235} to determine chemical separation rates, taking into account only the fissionable plutonium isotopes. Operation costs of the chemical separation plant are taken at the AEC published price of \$16,400.

Blanket

The blanket elements are made of depleted uranium, clad with stainless steel. They are considerably larger than the core fuel elements and therefore their fabrication cost per kg of fuel is lower.

Plutonium builds up in the blanket elements with time. The longer the fuel is left in the core the greater the buildup of plutonium in the blanket and the larger the net plutonium credit. However, an excessive buildup may lead to a major shift of the power split between core and blanket. It is also desirable to

obtain plutonium credit at regular (and not infrequent) intervals, in order to be able to claim the credit as part of the equilibrium fuel cycle. The fuel burnup obtainable is low, even over extended periods and therefore the metallurgical burnup limit does not influence the maximum allowable residence time of the blanket elements.

An average blanket residence time of three years was selected for the 300 MW_e reactor, leading to a final Pu enrichment of 1.1%. The same final enrichment was selected for the blanket of the 40 MW_e reactor. The corresponding average residence time is 4.5 years.

3.3.3 Operating and Maintenance Costs

Operating and maintenance costs include all supervisory, technical and operating labor, operating supplies, maintenance materials, and services. The payroll is based on a staff of 44 for the 40 MW_e plant and a staff of 62 for the 300 MW_e plant. These costs are shown in Table 3.13.

Table 3.13 — Near Term Plants — Estimate of Operating and Maintenance Cost

Description	Rate	40 MW _e		300 MW _e	
		No. of Personnel	Annual Cost, \$	No. of Personnel	Annual Cost, \$
Plant management					
Station superintendent	\$15,000/yr	1	15,000	1	15,000
Asst. station superintendent	12,000/yr	-	-	1	12,000
Secretary-typist	2.50/hr	1	5,200	1	5,200
Storekeeper-clerk	2.30/hr	-	-	1	4,800
Subtotal		2	20,200	4	37,000
Technical staff					
Nuclear engineer	\$10,800/yr	1	10,800	1	10,800
Results engineer	8,000/yr	1	8,000	1	8,000
Health physicist	8,400/yr	1	8,400	1	8,400
Laboratory technician	2.85/hr	1	5,900	2	11,900
Instrument technician	2.85/hr	-	-	1	5,900
Subtotal		4	33,100	6	45,000
Operating staff					
Shift supervisor (L)	\$10,800/yr	4	43,200	5	54,000
Senior control operator (L)	3.75/hr	4	31,200	5	59,000
Control operator (L)	3.65/hr	5	38,000	5	38,000
Assistant control operator	3.50/hr	5	36,400	5	36,400
Turbine equipment operator	3.50/hr	5	36,400	5	36,400
Special operator	3.50/hr	5	-	-	-
Watchman	2.25/hr	4	18,700	5	23,400
Janitor	2.25/hr	1	4,700	2	9,400
Subtotal		28	208,600	32	236,600

Table 3.13 — (Continued)

Description	Rate	40 MW _e		300 MW _e	
		No. of Personnel	Annual Cost, \$	No. of Personnel	Annual Cost, \$
Maintenance staff					
Maintenance supervisor	\$10,800/yr	2	21,600	2	21,600
Electrician	3.25/hr	2	13,500	3	20,300
Machinist	3.25/hr	-	-	2	13,500
Pipefitter-welder	3.25/hr	2	13,500	2	13,500
Millwright	3.25/hr	-	-	1	6,400
Instrument mechanic	3.25/hr	3	20,300	4	27,000
Mechanic	3.25/hr	1	6,800	4	27,000
Laborer	2.25/hr	-	-	2	9,400
Subtotal		<u>10</u>	<u>75,700</u>	<u>20</u>	<u>139,108</u>
Total labor		44	337,600	62	457,700
Fringe benefits, 20%			<u>67,400</u>		<u>91,300</u>
Total labor and fringe benefits			405,000		549,000
Maintenance materials and operating supplies					
Normal materials and supplies			<u>102,000</u>		<u>329,000</u>
Total annual operating and maintenance cost			507,000		878,000

Fig. 3.1 — Average core neutron flux spectrum — 300 MW_e fast breeders

Fig. 3.2 Average core neutron flux spectrum – 40 MW_e steam-cooled fast breeders

1967 DESIGN

NDA-TR-11944

Fig. 3.3 — 300 MW_e steam-cooled fast breeder reactor — simplified flow diagram

Fig. 3.5 — 300 MWe steam-cooled fast breeder reactor — elevation

Fig. 3.6 — 300 MW_e steam-cooled fast breeder reactor - section through core

Fig. 3.7 — 300 MWe steam-cooled fast breeder reactor — core — fuel element assembly

Fig. 3.8 — 300 MW_e steam-cooled fast breeder reactor — blanket fuel element subassembly

Fig. 3.9 — 300 MWe steam-cooled fast breeder reactor — building arrangement

Fig. 3.10 — 300 MWe plant general arrangement, plan, and elevation (courtesy of Kaiser Engineers, pp. 209 and 210, KE 60-31)

Fig. 3.11 — 40 MW_e steam-cooled fast breeder reactor – simplified flow diagram

Fig. 3.12 — 40 MWe steam-cooled fast breeder reactor — elevation

Fig. 3.13 — 40 MWe steam-cooled fast breeder reactor — plan and section through core

Fig. 3.14 — Critical mass vs fuel density for oxide fueled cores

Fig. 3.15 — Breeding ratio vs fuel density for oxide fueled cores

4. OPERATION AND SAFETY

It is expected that the operation of the fast breeder reactor plant will be similar to other types of nuclear plants with the following exceptions:

1. Water introduced into the reactor during shutdown must be replaced by steam, initially generated external to the reactor.
2. The control system must be sensitive because of the smaller volumetric heat capacity of steam compared with liquid coolants. Unlike gas-cooled reactors, however, a relatively large supply of coolant is available through flashing of the high temperature water stored in the Loeffler boiler.
3. Reshuffling of fuel elements is required more frequently than with thermal reactors.
4. Because of the presence of hydrogen in the coolant, changes in coolant density could lead to large changes in reactivity.

4.1 STARTUP AND SHUTDOWN

Two types of startup conditions are considered. The first is a cold startup, which occurs during the initial startup, and again when the reactor has been shut down for a considerable period for major overhaul or replacement of power plant components.

The second and more common type of startup takes place approximately every two months after the reactor is shut down for refueling. An appreciable

amount of residual heat is generated and reactor startup from this condition is simpler and faster than startup from a cold condition. In either case, the reactor contains water introduced in the previous shutdown.

4.1.1 Cold Startup

The proposed procedure for cold startup is as follows. The water in the two steam drums (Loeffler boilers) is heated by means of a gas fired startup boiler with a capacity of 11×10^6 Btu/hr. Steam formed in the startup boiler is piped to the Loeffler boiler until all the water is heated and steam pressure in the Loeffler boiler reaches some predetermined level. At this point, the steam from the startup boiler is fed directly to the reactor and draining of the water held in the reactor begins. When all the water has been removed from the core, steam circulation between the reactor and the Loeffler boiler is started by means of the auxiliary steam circulator. The reactor is then made critical and controls are adjusted to give a minimum heat output. The startup boiler is valved out of the circuit.

At some point during the process the steam leaving the reactor is channeled, for a short period, through the bypass line to the condenser. This eliminates any trapped air in the system.

When the water in the Loeffler boilers approaches operating conditions, the steam flow is passed through the condenser, and the main steam circulators are started. Reactor power is raised gradually. The steam flow is switched to the turbine when a reactor power level sufficient to sustain turbine operation is reached.

4.1.2 Warm Startup

The second type of startup occurs after a shutdown of short duration. The manner of startup will depend on the state of the shutdown process, namely, on whether the reactor steam is still passing through the condenser, is flowing directly to the Loeffler boilers, or whether the reactor has already been flooded.

4.1.3 Normal Shutdown

The proposed normal shutdown process starts from a point where the turbine power output is at a minimum. This is of the order of 25% of rated power. The output of the reactor and turbine are gradually decreased, with the turbine following the reactor output. At the cutoff point, the control rods are dropped, the bypass line to the condenser is opened, the turbine throttle is closed, and the steam circulator is gradually slowed down to a recirculation rate of the order of 0.7×10^6 lb/hr. The feedwater steam extraction lines now receive steam from the turbine-bypass line and the feedwater pump output is reduced to match the new flow conditions. Removal of the reactor decay heat in the condenser continues for the next 24 to 48 hr during which time the reactor heat generation drops from 6% immediately after shutdown to 0.5% of full load output.

After the decay heat drops to about 0.5% of the reactor output, the state of the water in the Loeffler boiler is in the order of 250 psi and 400 °F. At this point, the bypass line to the condenser is closed, the feedwater pumps are shut down, and an auxiliary cooler switched to the Loeffler boiler. The auxiliary cooler is connected on the tube side to the water of the Loeffler boiler, and on the shell side to outside cooling water. A closed loop is established between the reactor and the Loeffler boiler with the steam circulator providing steam flow. Heat from the reactor is transferred to the Loeffler boiler and from there to the auxiliary cooler. As shutdown proceeds, the pressure in the Loeffler boiler drops further until it reaches atmospheric. At this point the reactor is flooded. Cooling is provided by circulating the water from the reactor through the shutdown cooler.

4.2 REACTOR CONTROL SYSTEM

For the 300 MWe plant there are 14 control elements, 10 of which are located in the core section of the reactor. Of the core elements, eight are safety rods and two are operating shim rods. The four blanket elements are shim rods used to counteract the shift in power split between core and blanket due to the buildup of plutonium. The neutron absorbing material used is B_4C with 50% B^{10}

enrichment, clad with type 304 stainless steel tubing. B^{10} has a significant absorption cross section for fast neutrons due to the $B^{10}(n,\alpha)Li^7$ reaction. See Table 3.2.

The shim rods consist each of a cluster of seven 0.25 in. rods, operated as one unit. They are 120 cm long. They contain 0.23 kg of boron-10 per rod and are located 1 ft from the vertical center line of the reactor on a diametral line. They are capable of holding down a total of 97 cents of reactivity.

The safety rods are located on two concentric circles of 41 in. and 62 in. diameter, respectively. They have an active section 150 cm long and are capable of holding \$12.5 of reactivity. They contain 0.72 kg of B^{10} per rod.

The four blanket shim rods have a shorter active length and are capable of holding down a total of \$1 of reactivity.

All control elements are operated from the top with individual openings provided in the hemispherical head of the pressure vessel.

The shim rods are used to control the reactor operating power level, and to compensate for net fuel burnup and cold to hot reactivity changes. The safety rods are used to scram the reactor in the event of unusual operating conditions.

The 40 MWe reactor contains 10 control rods, six of which are located in the core. Of these, two are shim and four are safety rods. The blanket again contains four shim rods. The control rod design is similar to that of the 300 MWe plant. The individual rod worth and length are given in Table 3.2.

Either reactor will not become critical until full withdrawal of the safety rods and partial withdrawal of the shim rods has been accomplished.

4.3 FUEL BURNUP AND REFUELING

Two effects of fuel burnup are considered for a reactor fueled with plutonium of pure Pu^{239} isotope:

1. burnup of Pu^{239} , and
2. fission product poisoning.

For the 300 MWe oxide-fueled plant, the first effect is unimportant relative to the second. The reason for this is that there is little change in the Pu^{239} concentration with burnup since the core breeding ratio is very close to 1.0 (≈ 0.97). The

effect of fission product buildup, on the other hand, is important because there is an appreciable neutron flux at energies where the fission product cross sections are quite high (100 to 3000 eV). A plot of k_{eff} as a function of average core burnup is shown in Fig. 4.1. From this plot it may be seen that the buildup of fission products reduces k by about 1.5% at 10,000 MW-d/tonne burnup and that refueling intervals for a one dollar change in reactivity are about 2500 MW-d/tonne. The corresponding time interval at 80% plant capacity is about 50 days.

Fig. 4.1 also illustrates that an enrichment of 12.6% Pu^{239} is required to make $k = 1$ for an average core burnup of 10,000 MW-d/tonne. Since the core breeding ratio is near one, this is approximately the enrichment required at startup. In a configuration having a fixed number of core and blanket elements, this means that the shim control system would have to hold down about 1.5% excess reactivity (~\$4) at startup. Since it is desirable to limit the available excess reactivity to slightly less than \$1.00, additional excess reactivity may be introduced by replacement of the innermost blanket elements with core elements. This could be done in steps so that at no time does the excess reactivity exceed \$1.00. Alternatives to this approach are:

1. hold down the entire excess reactivity at startup by means of control rods and limit the maximum withdrawal speed of shim control rods to a few cents per day,
2. introduce poisons or lower enrichment fuel at the start of life.

Control of the reactor under equilibrium cycle plutonium isotope concentrations was not studied. However, it is expected that longer periods between refueling will be permissible because of the higher core breeding ratio (1.16) with equilibrium cycle fuel.

Refueling

Prior to refueling, the reactor is shut down, and the reactor cavity and the reactor core are filled with water. The feedwater system is used for flooding the reactor core, and biological shield cooling water is used for flooding the reactor cavity. The fuel carrying section of the refueling vehicle described in Sec-

tion 3.1.8 is submerged in the water which provides shielding for the irradiated fuel elements removed from the core.

In carrying out the refueling operation the shield plug is lifted and the refueling vehicle is moved into position over one of the seven refueling portholes. The grappling arm removes the pressure vessel closure plug. The grappling arm then moves into position to effect the removal and replacement of fuel elements. (See Figs. 3.9 and 3.12.)

After the operations within the reactor vessel are performed, the porthole closure plug is replaced, and the grappling arm is returned to its original position within the refueling vehicle. The vehicle then transports the irradiated fuel elements under water to a fuel chute through which they glide into the decay pool.

4.4 EMERGENCY COOLING

Emergency shutdown procedures are designed to eliminate dangers resulting from various equipment failures or operating failures.

Power Failure

Failure of power to the steam circulators will automatically cause the opening of a valve to the steam line which bypasses the steam circulators. Flow through this line is regulated by a preprogrammed control valve powered by batteries. This valve allows for maximum utilization of the enthalpy of the pressurized water in the Loeffler boilers by permitting the minimum amount of steam (corresponding to the maximum allowable exit temperature of the steam) to pass through the reactor. At the time the steam circulators are bypassed, the turbine main valve closes and a bypass line opens to the condenser. Condenser cooling water and vacuum pumps are switched to an emergency power supply. Water in the two Loeffler boilers (initial capacity 180,000 lb) flashes to steam and passes through the reactor and turbine bypass lines on the way to the condenser. After about 6 min of steam flow the pressure in the Loeffler boilers drops to 250 psi. By this time 33% of the water in the Loeffler boiler has been flashed into steam

and the reactor heat output will have diminished from 6% (after rods are dropped) to 3% of rated output. During this 6-min period, an emergency Diesel engine coupled to the emergency steam circulator can be started. The Loeffler boiler is then valved to a shutdown cooler and the primary and secondary cooler pumps are started on emergency power. The steam circulator bypass line is closed and the Diesel-powered steam circulators in parallel with the main steam circulators are used to pump the steam. The turbine bypass line to the condenser is closed. Cooling is continued by means of the reactor-Loeffler boiler loop, and is terminated as in normal shutdown.

Steam Line Rupture

Rupture of the main pipe on the turbine side calls for a somewhat different emergency shutdown procedure. The control rods are dropped to scram the reactor. The valves leading to the ruptured pipe are closed and the steam is sent to the cooling coil in the decay pool. The pipe has a distributor opening into the decay pool. The water in the pool serves as a heat sink. Simultaneously, the steam circulators are bypassed so that the steam passes through the line containing the preprogrammed control valve. All feedwater pumps are shut to prevent relatively cold water from entering the Loeffler boilers and lowering the enthalpy of the holdup water. This water is flashed to steam as in the emergency power failure shutdown. During this operation (6 min allowable time) the shutdown circulator is started. The Loeffler boiler is connected to the shutdown cooler, and the primary and secondary cooler pumps are started. When the pressure in the Loeffler boiler has dropped to 250 psi corresponding to the flashing of 1/3 of its water into steam, the steam line to the decay pool is closed, and the shutdown steam circulator is valved into the main steam line. Cooling continues in the reactor-Loeffler boiler loop and is terminated as in normal shutdown with the help of the auxiliary shutdown cooler.

4.5 EFFECTS OF CORE FLOODING

Flooding of the steam-cooled breeder coolant channels with liquid H₂O

can cause a rather severe rise in reactivity (in the order of a 20 to 25% increase in k). This rise is due almost exclusively to the neutrons which escape the U^{238} resonance region and which are subsequently absorbed at the 0.3 ev Pu^{239} resonance and lead to additional fissions.

In the normal operating state the median fission energy is 112 kev. All neutrons are absorbed within or above the U^{238} resonance region (6 to 10,000 ev). Full flooding of the reactor degrades the neutron spectrum to the extent that the bulk of the neutrons are absorbed between 0.1 and 100 ev. Roughly 50% of the neutrons which enter the U^{238} resonance region escape and are absorbed in the 0.3 ev Pu^{239} resonance, as there is little competition with plutonium for neutrons below 6 ev. It has been estimated that if the number of neutrons leaving the U^{238} resonance region could be reduced to about 30%, the reactor would shut itself down upon flooding. However, to achieve this value, the coolant to fuel volume ratio would have to be reduced to about 0.5, a value well below that required by heat transfer and heat transport considerations.

A promising solution is to add a permanent poison to the core which would have the ability to absorb neutrons below the U^{238} resonance region, yet not be detrimental under normal operating conditions. In the light of these requirements, the behavior of the resonance absorbers indium and hafnium were investigated.

The k_{∞} of the 300 MW_e reference design containing U^{238} , Pu^{239} , hydrogen and the resonance poison was calculated by means of a six-group diffusion theory. These groups consisted of two fast groups, a "resonance" group extending from 0.1 Mev to 6 ev, a 0.6 ev to 0.3 ev group, a group representing the Pu^{239} resonance at 0.3 ev, and a group below the Pu^{239} resonance. Cross sections for these groups were obtained from measurements or calculations of resonance integrals allowing for self shielding effects.

It was found that 0.1 atoms of indium or 0.03 to 0.04 atoms of natural hafnium per atom of fuel ($Pu + U^{238}$) are needed to provide sufficient absorption to

make the reactor safe upon flooding. The addition of the required amount of hafnium is expected to have a small influence on the critical mass and breeding ratio under normal operation. Other possible elements whose oxides could be mixed with the fuel to make the reactor safe from flooding include silver, europium, and many rare earths. With such materials incorporated in the fuel, it is possible to tolerate partial as well as full reactor flooding, with appreciable savings in the auxiliary systems required for startup, shutdown, and refueling.

Flooding calculations described above are based on an infinite reactor and may be considered applicable to the 300 MW_e reactor where leakage is small. The flooding characteristics of the 40 MW_e core, where there is considerable leakage, were not investigated.

4.6 CORE VOIDING

Core voiding characteristics of the steam-cooled breeder were calculated by means of the 10 group, one-dimensional diffusion model used for calculating critical mass and breeding ratio. The group structure used, although adequate for breeding ratio and critical mass calculation, is perhaps inadequate for accurate determination of voiding effects. This is the result of insufficient energy group subdivision in the region of U²³⁸ resonances. Results obtained must, therefore, be considered preliminary in nature, subject to verification by accurate calculations.

According to these calculations, complete loss of coolant from the 300 MW_e core with an initial Pu²³⁹ - U²³⁸ loading would produce a reactivity increase of \$12. This effect is reduced to about \$8 when equilibrium concentrations of higher plutonium isotopes are present in the core.

The positive void coefficient is less severe with a "pancake" core (small height to diameter ratio), with a 40 MW_e reactor and a D₂O steam-cooled reactor. A U²³³-thorium fueled reactor was found to have a negative voiding effect.

On the basis of these calculations it appears that a combination of such design modification might yield a negative void coefficient for the 300 MW_e reactor.

The 40 MW_e H₂O-cooled reference reactor is likely to have a negative void coefficient under equilibrium fuel loading.

4.6.1 Near Term Reference Designs

The effects of coolant voiding for the 300 MW_e and 40 MW_e, H₂O steam-cooled breeders are given in Figs. 4.2 and 4.3 as a function of steam density change. The reactivity increases nearly linearly with decreasing steam density, ρ_s .

The following approximate formulae may be written.

300 MW_e Reference Design (Pu²³⁹-U²³⁸ loading):

$$\frac{\Delta k_{\text{eff}}}{\beta k_{\text{eff}}} = -12.14 \frac{\Delta \rho_s}{\rho_s} \text{ dollars}^*$$

40 MW_e Reference Design (Pu²³⁹-U²³⁸ loading):

$$\frac{\Delta k_{\text{eff}}}{\beta k_{\text{eff}}} = -1.36 \frac{\Delta \rho_s}{\rho_s} \text{ dollars}^*$$

300 MW_e Reference Design (equilibrium fuel concentration):

$$\frac{\Delta k_{\text{eff}}}{\beta k_{\text{eff}}} \approx -8 \frac{\Delta \rho_s}{\rho_s} \text{ dollars}^*$$

The voiding problem in these reactors arises because the fuel is more reactive at higher neutron energies. For Pu²³⁹, η increases with increasing energy over the range of interest, and parasitic captures in U²³⁸ decrease sharply. The hydrogenous coolant thus acts indirectly as a poison by degrading the neutron spectrum. This effect is manifested in a rise in the value of k_{∞} for the core with reduced coolant density. It is partially offset by the increase in neutron leakage from the core at the higher energy spectrum obtained with reduced coolant density. The effect of coolant voiding on an infinite reactor core is shown in Fig. 4.2.

An understanding of these effects may be obtained by examining the effect of coolant voiding on neutron spectra, shown in Figs. 4.4 and 4.5, and the ef-

*The delayed neutron fraction, β , is taken equal to 0.00412.

fect of neutron energy on the neutron regeneration factor, η , and the neutron absorption cross section of the fuel materials used, shown in Figs. 4.6 and 4.7.

The increase in leakage upon voiding results from the increase in the average diffusion length with removal of hydrogen from the core. For the 300 MW_e H₂O steam-cooled breeders, the spectral effect is much larger than the leakage effect. As the reactor size decreases, the leakage term becomes more important and the void coefficient decreases.

The reduction in void coefficient under equilibrium fuel cycle conditions is attributed mainly to the buildup of Pu²⁴¹, for which its neutron regeneration factor, η , unlike Pu²³⁹, does not increase monotonically with increasing neutron energy. See Table 4.1.

Table 4.1 — Regeneration Factors and Capture Cross Section of Fissile and Fertile Isotopes

Group	Group Range, ev	η		$\sigma_{n,\gamma}$ (barns)	
		Pu ²³⁹	Pu ²⁴¹	U ²³⁸	Pu ²⁴⁰
1	$10^7 - 3 \times 10^5$	2.93	2.89	0.105	0.123
2	$3 \times 10^5 - 6.7 \times 10^4$	2.42	2.56	0.231	0.366
3	$6.7 \times 10^4 - 3.4 \times 10^3$	2.06	2.58	0.377	1.30
4	$3.4 \times 10^3 - 450$	1.86	3.16	0.950	4.06

4.6.2 Alternate Reactor Designs

Improvements in Void Coefficient

Three approaches may be taken to minimize or reverse the sign of the void coefficient: (1) the neutron leakage from the core may be increased, (2) other coolant or fuels may be used, and (3) absorbing or moderating materials may be added to the core. The following possible modifications suggest themselves:

1. Increase leakage with a core height to diameter ratio substantially larger or smaller.
2. Use D₂O steam which leads to less degradation of the neutron spectrum.

3. Use U^{233} - Th^{232} fuel which promises to be less sensitive to changes in neutron spectrum.

These methods were investigated in some detail by means of a four-group hand calculation method. The calculation method is described in Section 4.6.3.

Pancake Design

The void coefficient of reactivity decreases with increased neutron leakage. Such an increase can be obtained by increasing the core surface to volume ratio. Details of a "pancake" design, a core with a small length to diameter ratio, are presented in Section 7.1. A reactivity increase upon voiding of +0.023 or \$5.60 was calculated for this design. This rise in reactivity, although less than one half that of the reference core, is still too large for the reactor to be considered safe. Additional modifications may, therefore, be required. Recalculation based on more refined methods should first be made.

D₂O-Cooled Reactor

The loss of a coolant with lower moderating effectiveness than H₂O steam would lead to a smaller positive component of the void coefficient. A 300 MW_e D₂O steam-cooled reactor was investigated. The reactivity increase corresponding to a complete loss of D₂O steam for a Pu²³⁹-U²³⁸ loading is only +0.012 or +2.91 dollars.

U²³³-Th²³² Breeder Reactor

For enrichments in the range of interest, η of U²³³-Th²³² fuel is not very sensitive to changes in neutron energy.⁶ One may, therefore, expect a reduction in the positive component of the void coefficient.

A 300 MW_e mixed oxide Th²³² and U²³³-fueled H₂O steam-cooled breeder was briefly investigated. This design is described in Section 7.3. The void coefficient was found to be negative. A reactivity change of -0.0035, about minus one dollar, is obtained for complete loss of coolant. The capture cross sections of thorium in the energy range between 3500 and 500 ev were not sufficiently reliable, however, to inspire a high degree of confidence in this result.

4.6.3 Four-Group Method Used for Calculating Void Coefficients

Calculation of void coefficients made for the purpose of comparison of the alternate designs with the reference design were carried out by means of a four-group analysis for a one-region reactor. Savings due to the blanket were taken into account in the determination of the geometrical buckling. The choice of energy groups was based on consideration of fission threshold of the fertile material and the importance of transfer cross sections with energy. Cross sections were obtained by condensing the 10-group set used for criticality calculations. The lower energy limit for this set was 450 ev. Neutrons below this level were, therefore, treated as members of the lowest energy group. The low energies are not important for critical mass calculations, but may have a significant effect on reactivity changes due to coolant voiding, particularly in the borderline cases such as the low L/D design and the D₂O steam-cooled reactor.

Neutron balance equations were written for each group and neutron fluxes calculated starting with the highest group. The reactivity was evaluated as:

$$k_{\text{eff}} = \int_0^{\infty} z(E) A(E) dE = \sum_{i=1}^{i=4} z_i A_i = \sum_{i=1}^{i=4} (z_i \mu_{ai})_{\text{fuel}} \phi_i$$

where z_i = number of neutrons produced per neutron absorbed in fuel for group i ,

A_i = absorption in fuel for group i ,

μ_{ai} = fuel microscopic absorption cross section for group i ,

ϕ_i = flux for group i .

At the operating point, $k_{\text{eff}} = 1$.

For complete loss of steam ($\rho_s = 0$), the void coefficient is

$$\left[\frac{\Delta k_{\text{eff}}}{k_{\text{eff}}} \right]_{\rho_s=0} = \sum_{i=0}^4 (z_i \mu_{ai})_{\text{fuel}} \Delta \phi_i$$

where $\Delta \phi_i = \phi_i^{\text{void}} - \phi_i^{\text{operating}}$.

For small decreases in steam density, the void coefficient was calculated by means of a perturbation theory assuming that only transfer cross sections are perturbed.

An important component of the steam coefficient of reactivity is the negative leakage effect. The effect of increased neutron streaming obtained upon voiding can be included if the diffusion coefficient is calculated by the method of Benoist (for an infinite array of rods):

$$D' = \frac{1}{3\left(1 + \frac{V_2}{V_1}\right)\mu_1} \left[P_{11} + \mu_1 \frac{P_{12}}{\mu_2} + \frac{V_2}{V_1} \left(P_{21} + \mu_1 \frac{P_{22}}{\mu_2} \right) \right]$$

where μ_1, μ_2 = transport cross sections of regions 1 and 2,

V_2/V_1 = volume ratio of region 2 to region 1,

P_{ij} = probability that a neutron born in region i makes its next collision in region j .

Removal and transport cross sections were averaged and the diffusion coefficient calculated in each group for both the operating and voided case.

It was found that the correction due to neutron streaming reduced the effect of voiding for the 300 MW_e reference design by about 25 cents. This improvement is insignificant compared with the approximations and uncertainties present in void coefficient calculations.

The effect of higher plutonium isotopes on the void coefficient of the 300 MW_e reactor was estimated by accounting only for changes in the absorption and fission cross sections of the fuel at its equilibrium composition. Group to group transfer cross sections of the higher isotopes were not available.

4.7 TEMPERATURE COEFFICIENT

The temperature coefficient for the 300 MW_e reference reactor due to Doppler broadening is estimated at $-1.2 \times 10^{-5}/^{\circ}\text{C}$; the temperature coefficient resulting from the expansion of the fuel and cladding is about $+0.6 \times 10^{-6}/^{\circ}\text{C}$. The net effect of temperature on fuel and cladding is, therefore, negative. The coolant temperature coefficient is $\sim +1.2 \times 10^{-4} (\Delta k/k)/^{\circ}\text{C}$, and is therefore likely to have a dominant effect in a power excursion. See Fig. 4.2.

Fig. 4.1 — Effect of fission product poisoning on reactivity for 1967 300 MW_e reference design. Core breeding ratio = 1.0.

Fig. 4.2 — Reactivity effect of coolant voiding — 300 MWe reference design

Fig. 4.3 — Reactivity effect of coolant voiding —
40 MW_e reference design

Fig. 4.4 — Normalized average neutron flux as a function of neutron energy — 300 MW_e reference design

Fig. 4.5 — Normalized average neutron flux as a function of neutron energy — 40 MW_e reference design

Fig. 4.6 — Variation of neutron regeneration factor, $\bar{\eta}_F$, with neutron energy

Fig. 4.7 — Variation of average microscopic absorption cross sections with neutron energy

5. SELECTION OF NEAR TERM PLANT CHARACTERISTICS

Selection of the reference designs was based on nuclear, heat transfer, and cost optimizations described in the following.

5.1 HOT SPOT AND HOT CHANNEL FACTORS

Core heat transfer calculations were based on hot spot and hot channel factors as shown in Table 5.1. These values were selected after careful survey of such factors used in the nuclear industry, and after making adjustments appropriate to the steam-cooled breeder design. The overall factors used in the design of the reactors are the product of the individual factors, which corresponds to the assumption that the worst possible conditions occur simultaneously in the hottest channel.

In connection with $F_{\Delta H}$, the coolant enthalpy rise factor, it should be pointed out that the possible 27% increase in coolant enthalpy rise indicated in the table corresponds to an increase of about 36% in the coolant temperature rise. The reason for the difference is that the specific heat of steam decreases significantly with increase in temperature.

5.2 HEAT TRANSFER AND PRESSURE DROP RELATIONS

The heat transfer coefficient was calculated by means of the McAdams, Kennel, Addoms relation

$$h = 0.0214 \left(\frac{k}{De} \right) \left(\frac{DeG}{\mu} \right)^{0.8} \left(\frac{Cp\mu}{k} \right)^{1/3}$$

Table 5.1 — Steady-State Hot Spot and Hot Channel Factors —
Steam-Cooled Fast Breeder Reactor

	$F_{\Delta H_{\text{coolant}}}$ *	$F_{\Delta t_{\text{fuel}}}$ †	F_{θ} ‡
Fuel Rod Characteristics			
Pellet diameter	1.004	1.005	1.005
Pellet density	1.02	1.03	1.03
Pellet enrichment	1.01	1.02	1.02
Rod spacing and bowing	1.05	—	1.05
Thermal conductivity of fuel	—	1.10	—
Subtotal	<u>1.09</u>	<u>1.16</u>	<u>1.11</u>
Coolant Flow Characteristics			
Plenum chamber conditions	1.02	—	1.02
Heat transfer coefficient	—	—	1.05
Density factor	1.04	—	1.03
Subtotal	<u>1.06</u>	<u>0</u>	<u>1.10</u>
Nuclear Factors			
Uncertainty in radial flux	1.05	1.05	1.05
Uncertainty in local flux	—	1.05	1.05
Subtotal	<u>1.05</u>	<u>1.10</u>	<u>1.10</u>
Precision of Power Measurement	1.05	1.05	1.05
Overall Factor			
Product of individual factors	1.27	1.34	1.41

* $F_{\Delta H}$ — factor applicable to coolant enthalpy rise from inlet to outlet.

† $F_{\Delta t_{\text{fuel}}}$ — factor applicable to fuel temperature drop from center to surface of fuel pellet.

‡ F_{θ} — factor applicable to coolant film temperature drop from cladding surface to the bulk of the coolant.

where C_p = specific heat, Btu/lb-°F

k = thermal conductivity, Btu/hr-ft-°F

μ = viscosity, lb/hr-ft,

The above factors are taken at the mean film temperature, and

G = mass velocity, lb/hr-ft²

D_e = equivalent hydraulic diameter, ft.

A conservative set of properties was assumed as outlined in Reference 7.

The friction pressure drop in the core was computed by means of the conventional Fanning equation for flow inside smooth tubes, using the equivalent hydraulic diameter. The calculated pressure drop was increased by 25% to allow for spacer wire and surface roughness effects. The overall pressure drop reported also includes the axial blanket pressure drop, entrance and exit losses and change in momentum effects. To obtain a constant coolant exit temperature from all channels, orificing was assumed.

5.3 SELECTION OF CORE PARAMETERS, NEAR TERM DESIGNS

The core parameters selected for the near term designs of the 40 and 300 MW_e plants are shown in Table 5.2. The range of each variable considered is also indicated. The maximum clad surface temperature and maximum central fuel temperature were computed using the steady state hot spot and hot channel factors listed in Table 5.1.

5.3.1 Fuel Rod Diameter

The fuel rod diameter has an important effect on both fuel inventory and fuel cost. In Fig. 5.1 this effect is shown graphically for the 300 MW_e reactor. Two curves are shown, one representing the cladding surface temperature limitation, the other the fuel temperature limitations. The two curves cross at a fuel rod diameter of 0.25 in. As the fuel rod diameter is increased, the total fuel inventory and the critical mass increase, while the fabrication cost per kg of fuel decreases. The fuel cost reaches a minimum at about 0.25 in., at which point the

Table 5.2 — Selection of Core Parameters, Near Term Fast Breeder Reactors

	40 MW _e Reactor	300 MW _e Reactor	Range Permitted
Coolant Inlet Temp, °F	610	610	> saturation
Coolant Exit Temp, °F	925	950	≤1050
Coolant Exit Pressure, psia	1450	1450	≤2000
Max. Coolant Velocity, ft/sec	210	240	≤250
Core Pressure Drop, psi	100	100	≤100
Max. Inconel Clad Surface Temp, °F	1350	1350	≤1350
Max. Central Fuel Temp, °F	4500	4500	≤4500
Fuel Rod Diameter, in.	0.250	0.250	≥0.150
Ligament Spacing, in.	0.040	0.060	≥0.040
Coolant to Fuel Volume Ratio	0.70	0.99	≤1.0
Core Length to Diameter Ratio	1.03	0.83	0.8-1.1

increase in fuel lease charge becomes greater than the decrease in fuel fabrication cost (Table 5.3).

A 0.25 in. diameter rod has been chosen for the near term designs. Besides being near optimum costwise, it is not smaller than currently manufactured uranium oxide rods and therefore should not require fabrication method development.

The effect of fuel rod diameter upon the nuclear properties may be understood upon examination of three surface temperature-limited designs having clad rod diameters of 0.150, 0.200, and 0.270 in., respectively. (See Table 5.4.) These designs were based upon a maximum cladding temperature of 1250°F and, therefore, do not relate directly to the reference 300 MW_e design. The three reactors; all have the same pressure drop across the core, the same steel to fuel volume ratio, and a constant ligament spacing of 50 mils between the fuel rods. The latter condition causes the coolant/fuel volume ratio to increase as the rod size decreases, an effect which tends to influence the overall result by increasing the degradation of the neutron spectrum.

The neutron leakage from the core is relatively small for the 300 MW_e reactor. Therefore, an increase in fuel volume results in almost a proportionate increase in critical mass, the fuel enrichment decreasing only slightly with increasing fuel volume.

As this reactor is surface temperature-limited, the fuel volume and core size increase with fuel rod diameter. For the 300 MW_e reactor there is only a small decrease in neutron leakage; the plutonium enrichment, therefore, changes only slightly with increasing fuel volume. The resulting increase in critical mass is illustrated in Fig. 5.2.

The increase in overall breeding ratio with an increase in rod diameter reflects mainly the hardening of the spectrum attributable primarily to the decrease in coolant to fuel ratio. The breeding ratio in the core increases with increasing fuel rod diameter, while the blanket breeding ratio decreases. This is

Table 5.3 — Effect of Fuel Rod Diameter on Core Fuel Cost

Near Term 300 MW _e Plant			
Fuel Rod OD, in.	0.20	0.25	0.30
Fuel Volume, ft ³	42.8	54.0	78.5
Fabrication Cost, \$/kg	341	275	227
Fuel Lease Charge, \$/kg	143	180	261
Burnup and Fuel Recovery Costs, \$/kg	230	230	230
Core fuel cost, \$/kg	<u>714</u>	<u>685</u>	<u>718</u>
Core fuel cost, mill/kwh	2.39	2.29	2.40

1975 300 MW _e Plant			
Fuel Rod OD, in.	0.15	0.20	0.25
Fuel Volume, ft ³	30.3	40.8	51.4
Fabrication Cost, \$/kg	214	151	115
Fuel Lease Charge, \$/kg	203	260	318
Burnup and Fuel Recovery Costs, \$/kg	233	233	233
Core fuel cost, \$/kg	<u>650</u>	<u>644</u>	<u>666</u>
Core fuel cost, mill/kwh	1.15	1.14	1.18

Table 5.4 — Effects of Variable Fuel Rod Diameter

$$V_{\text{steel}}/V_{\text{fuel}} = 0.48$$

Clad Rod OD, in.	Fuel Loading, ft ³ Oxide	Leakage <u>Volume Coolant</u> <u>Volume Oxide</u>	Required Enrichment	Pu ²³⁹ Alpha*	Median Fission Energy, kev	Fraction of Fissions Occurring in U ²³⁸	Core Leakage per Neutron Produced in Core	Critical Mass, kg Pu	Overall Breeding Ratio
0.15	50.6	1.53	13.5	0.333	80	0.129	0.152	1530	1.27
0.20	67.6	1.08	12.0	0.317	93	0.128	0.134	1900	1.31
0.27	91.0	0.81	11.1	0.304	102	0.129	0.106	2440	1.33

*Capture to fission cross section ratio.

the result of reduced leakage from the core with increasing core size. (See Fig. 5.3.)

If the coolant to fuel volume ratio were constant, a similar increase in critical mass with fuel rod diameter would occur. The enrichment would also decrease slightly with increasing fuel volume. However, the neutron spectrum would tend to soften very slightly with increasing fuel volume, because of the effect of decreasing enrichment which leads to an increase in the ratio of scattering to fission macroscopic cross sections.

5.3.2 Coolant Inlet Temperature

A low reactor inlet temperature decreases the flow rate through the core, reduces the required coolant to fuel volume ratio, and minimizes flooding and voiding problems. The inlet coolant temperature is, therefore, set only a few degrees above saturation.

5.3.3 Coolant Exit Temperature

The selection of the coolant exit temperature has a significant effect on the core size. This effect is shown in Fig. 5.4 for a constant fuel rod diameter, rod spacing, maximum clad temperature, and core pressure drop. In order to minimize the heat transfer surface and hence the fuel volume and critical mass, it is desirable to select the lowest acceptable coolant exit temperature. For 0.250 in. OD rods this temperature is 950 °F for the 300 MW_e reactor, and 925 °F for the 40 MW_e reactor. These coolant temperatures lead to sufficiently high power densities that the limiting central fuel temperature of 4500 °F is reached.

A lower fuel volume leads to a lower plutonium inventory. The total breeding ratio remains nearly constant but the core breeding ratio decreases with decreasing core size. These effects were discussed in Section 5.3.1.

The reduction in fuel use charges for the lower exit coolant temperature is largely offset by the increase in fuel throughput, caused by a decrease in thermal cycle efficiency. However, the lower core volume requires a somewhat smaller pressure vessel. Energy costs, therefore, go down slightly with reduced coolant exit temperature in the range between 1050 °F to 900 °F.

5.3.4 Coolant Pressure

The coolant pressure has little effect on core size for constant core pressure drop. An increase in coolant pressure results in a heavier pressure vessel and also increases the ratio of hydrogen atoms to fuel atoms in the core. The latter effect causes a decrease in breeding ratio, and a slight increase in enrichment due to spectral softening. Offsetting these unfavorable effects is a slight increase in cycle efficiency, a reduction in steam pumping power, and a small reduction in critical mass because of the smaller fuel volume.

For the reference designs a coolant exit pressure of 1450 psia was selected. This is approximately the optimum pressure for a nonreheat steam plant operating with steam temperature between 900 and 950 °F.

5.3.5 Core Pressure Drop

An allowable increase in core pressure drop can be utilized to decrease either the fuel volume or the coolant to fuel volume ratio, at the expense of increased pumping power. A cost analysis shows that the fuel cost remains fairly constant when the fuel volume is decreased by an increase in core pressure drop. The accompanying decrease in pressure vessel cost is offset by the increased cost of the pumping so that over a wide range of pressure drops the total energy cost remains fairly constant. It would therefore appear that a larger pressure drop leading to a smaller fuel inventory could be tolerated without economic penalty. For both the 300 MW_e and 40 MW_e reference designs the pressure drop is about 100 psi. The maximum fuel temperature is at its limit, i.e., 4500 °F. A further increase in pressure drop, therefore, could only be utilized to reduce the coolant to fuel volume ratio. The pressure drop of 100 psi corresponds to a steam velocity of about 240 ft/sec. It would not be desirable to go to much higher velocities because of possible erosion difficulties.

An increase in core pressure drop for a core with a fixed core geometry causes the critical mass and the core breeding ratio to decrease, while the total

breeding ratio remains approximately constant. These effects are illustrated in Table 5.5 for several typical surface temperature-limited reactor designs. The results of increasing pressure drop are similar to those of reducing fuel rod diameter presented in Section 5.3.1 except for the following: (1) the variations of fuel loadings (core sizes), enrichment, and breeding ratios occur over a narrower range, and (2) the spectrum becomes hardened with reduced fuel volume, because the effect of increase in enrichment is not offset by a large increase of the hydrogen to fuel atom ratio.

5.3.6 Ligament Spacing

An increase in ligament spacing, for a fixed core pressure drop must be accommodated by increases in the core length to diameter ratio as well as the coolant to fuel volume ratio. The total fuel volume is also increased. This is shown in Fig. 5.5. The ligament spacing for each core, therefore, must be selected so that core L/D ratio and coolant to fuel volume ratio are satisfactory from a nuclear standpoint. A core L/D ratio much different from 1 leads to significant increases in the critical mass. A high coolant to fuel volume ratio would degrade the neutron spectrum and adversely affect the breeding ratio. The acceptable ranges of these variables are shown in Table 5.2.

On the basis of the core L/D ratio and the coolant to fuel volume ratio, the 40 MW_e core could be designed with a spacing somewhat smaller than the 0.040 in. selected. However, it is felt that 0.040 in. is close to the lower limit rod spacing for 0.25 in. diameter rods from fabrication as well as heat transfer and flow distribution considerations. The 300 MW_e core requires a 0.060 in. ligament spacing in order to keep the core L/D ratio above 0.8. These considerations were not governing in the design of the pancake reactor described in Section 7.1 where a large critical mass was accepted for safety reasons.

5.3.7 Steel to Fuel Ratio

It was found that, within a fairly wide range, the amount of stainless steel

Table 5.5 — Effects of Core Pressure Drop

$V_{\text{coolant}}/V_{\text{fuel}} = 1.08$; $V_{\text{structure}}/V_{\text{fuel}} = 0.48$; fuel rod diameter = 0.200

Allowable Pressure Drop, psig	Fuel Loading, ft ³ Oxide	Required Enrichment	Median Fission Energy, kev	Critical Mass, kg Pu	Core Breeding Ratio	Total Breeding Ratio
76	51.1	12.4	107	1590	0.96	1.31
109	44.9	12.6	111	1415	0.95	1.31
156	38.9	12.8	112	1243	0.93	1.31

or nickel structure in the reactor has little effect upon the critical mass and the breeding ratio.

Fig. 5.6 shows a plot of critical mass vs steel to fuel volume ratio for surface temperature-limited designs all having the same coolant to fuel volume ratio. Three cladding thicknesses are indicated on the plot corresponding to the three reactors investigated. It is seen that over the range studied, a 230% increase in cladding thickness results in a mere 7% increase in critical mass. The corresponding breeding ratio decrease is only about 4.5%.

The effects of increasing the steel to fuel ratio are largely the result of increased inelastic scattering by the nickel and iron atoms in the steel (Inconel-X has a similar effect). This results in an increase in the value of α for plutonium, and a decrease in the U^{238} fission fraction. The latter effect is the result of inelastic scattering of fast neutrons to energies below the fission threshold for U^{238} . In the reactors under consideration, the presence of hydrogen reduces the relative importance of these effects compared to sodium-cooled reactors.

Fig. 5.1 — Effect of fuel rod diameter on fuel volume.
 Net plant output = 300 MW_e; coolant exit pressure = 1450 psia; coolant exit temperature = 950 °F; coolant inlet temperature = 610 °F; coolant pressure drop = 100 psi; coolant/fuel volume ratio = 1.0.

Fig. 5.2 — Effect of fuel rod diameter on critical mass. All designs are surface temperature limited, and have the same ΔP across core. $\rho_{\text{fuel}} = 10.0 \text{ g/cm}^3$.

Fig. 5.3 — Effect of fuel rod diameter on breeding ratio. Net plant output 300 MW_e. $\rho_{\text{fuel}} = 10.0 \text{ g/cm}^3$.

Fig. 5.4 — Effect of coolant exit temperature on core parameters. Net plant output = 300 MW_e; coolant exit pressure = 1450 psia; coolant pressure drop = 100 psi; coolant inlet temperature = 610 °F; maximum clad temperature = 1350 °F; rod diameter = 0.25 in.; rod spacing = 0.06 in.

Fig. 5.5 — Effect of ligament spacing. Net plant output = 300 MWe; coolant exit pressure = 1450 psia; coolant exit temperature = 950 °F; coolant inlet temperature = 610 °F; coolant pressure drop = 100 psi; maximum clad temperature = 1350 °F; maximum fuel temperature = 4500 °F; rod diameter = 0.25 in.

Fig. 5.6 — Effect of $V_{\text{steel}}/V_{\text{fuel}}$ on critical mass
($V_{\text{coolant}}/V_{\text{fuel}} = 1.08$)

6. 1975 REACTORS

The 1975 fast breeder designs are based on a maximum steady state cladding temperature of 1500°F. Nickel alloys which permit such temperatures are already under development.

Other improvements in technology and operating experience with this type of reactor may permit 1975 designs with vapor suppression type of containment, and simplified fuel handling equipment. Increased burnup capability of $\text{PuO}_2\text{-UO}_2$ fuel from 50,000 to 90,000 MW-d/tonne and a reduction in fuel element fabrication costs would cut fuel cycle costs by more than 50%.

6.1 PLANT CHARACTERISTICS

The characteristics of the 1975, 300 MW_e and 40 MW_e plants are shown in Table 6.1. Hastelloy-N has been chosen as the cladding, although other nickel alloys such as Inconel-X or Hastelloy-X may also be satisfactory. The higher surface temperature (1500°F) has been utilized to raise the core exit temperature of the 300 MW_e plant from 950 to 1075°F. The coolant pressure was correspondingly raised from 1450 to 2075 psia. Steam conditions for the 40 MW_e plant were set at 1450 psia and 975°F, as there is little incentive to increase the pressure for the small size plant.

Inasmuch as a reduction in fuel fabrication costs is expected for 1975 (no grinding of pellets), the optimum fuel rod diameter is 0.20 in. instead of 0.25 in. (This is illustrated in Table 5.3 where fuel costs for three fuel rod diameters are compared.)

Table 6.1 — Reactor Plant Characteristics — 1975 Steam-Cooled Fast Breeders

	40 MW _e	300 MW _e
Heat Balance		
Total reactor power, MW _t	122	825
Turbine Cycle Conditions		
Throttle temp, °F	970	1060
Throttle pressure, psia	1415	2015
Total steam flow, lb/hr	0.46×10^6	2.53×10^6
Reactor Description		
Reactor vessel		
ID, ft	8.0	10.1
Inside height, ft	15.0	29.8
Wall thickness, in.	5.6	9.6
(cylindrical portion)		
Material	SS clad carbon steel	
Design pressure, psig	1860	2500
Design temp, °F	800	800
Reactor core		
Active equivalent diameter, ft	2.50	5.43
Active height, ft	1.97	4.40
Active core volume, ft ³	9.66	130
Total fuel loading, kg fuel	1170	11,620
Avg Pu ²³⁹ content, w/o	18.0	12.8
Structural material	SS	SS
Neutron moderator	None	None
Blanket		
Material	Depleted UO ₂	
Axial thickness, ft	1.5	1.0
Radial thickness, ft	1.5	1.0

Table 6.1 — (Continued)

	40 MW _e	300 MW _e
Core fuel element		
Fuel material	UO ₂ -PuO ₂	UO ₂ -PuO ₂
Fuel element geometry	Rods	Rods
Clad material	Hastelloy-N	Hastelloy-N
Fuel rod OD, in.	0.200	0.200
Clad thickness, in.	0.0125	0.0125
Ligament spacing, in.	0.035	0.045
No. of rods per cluster	217	217
Active length, ft	1.97	4.40
Axial blanket fuel element		
Fuel material	UO ₂	UO ₂
Fuel element geometry	Rod	Rod
Clad material	SS	SS
Active length, ft	1.5	1.0
Radial blanket fuel element		
Fuel material	UO ₂	UO ₂
Fuel element geometry	Rod	Rod
Clad material	SS	SS
Active length, ft	4.97	6.40
Fuel assembly		
Total no. in core	50	256
Shape	Hexagon	Hexagon
Distance across flats, in.	3.67	3.81
Shroud material	SS	SS
Shroud thickness, in.	0.100	0.100
Lattice arrangement	Hexagonal	Hexagonal
Lattice spacing, in.	3.70	3.84

Table 6.1 — (Continued)

	40 MW _e	300 MW _e
Core Performance Data		
Core coolant	Steam	Steam
Core coolant outlet temp, °F	975	1075
Core coolant inlet temp, °F	610	650
Core coolant inlet pressure, psia	1550	2175
Primary coolant flow, lb/hr	1.23×10^6	6.51×10^6
Avg core coolant velocity, ft/sec	100	78
Max. fuel center temp, °F	4500	4100
Max. cladding temp, °F	1500	1500
Max. core heat flux, Btu/hr-ft ²	672,000	592,000
Avg core heat flux, Btu/hr-ft ²	292,000	197,000
Avg specific power, kw _t /kg of Pu	604	577
Fuel management	Inward radial shift	
Avg fuel burnup in exit fuel, MW-d/tonne	60,000	60,000
Peak to avg burnup ratio	1.5	1.5
Nuclear Characteristics		
Critical mass, kg Pu ²³⁹	188*	1370*
Enrichment, % Pu ²³⁹	18.0	12.8
Total breeding ratio	1.38*	1.27*
Core breeding ratio	0.53	0.95
Blanket breeding ratio	0.85	0.32

*Initial values, starting with Pu²³⁹, and a fuel density of 10.0 g/cm³, 91% of theoretical density.

6.2 NUCLEAR CHARACTERISTICS

The important nuclear characteristics for the 1975 steam-cooled reactor plants are given in Table 6.1.

Both the 300 and 40 MW_e reactor cores are smaller than the corresponding near term plants; the 300 MW_e 1975 plant has 11% less fuel volume while the 40 MW_e plant contains 30% less fuel. The lower fuel volume, made possible mainly by the use of a smaller fuel rod diameter, leads to lower critical masses for the 1975 designs.

The reason for the larger percentage reduction in fuel volume for the smaller plant is as follows. There is no incentive to use a 2000 psig turbine of 40 MW_e size; hence the turbine steam throttle temperature is set at 970°F, 90°F lower than in the 1975 300 MW_e reactor. The increase in the allowable clad temperature from 1350 to 1500°F results in a higher reduction in fuel volume for the 40 MW_e 1975 reactor. The turbine throttle steam temperature for the 300 MW_e plant is set at 1060°F, which is justified for a 2000 psig turbine pressure.

The total breeding ratios for the 1975 reactors are slightly lower than for the respective near term reactors.

6.3 ENERGY COSTS

Power generation costs for the 1975 fast breeders are given in Table 6.2. Reduction in capital costs are expected to result from (1) the smaller core sizes and hence smaller pressure vessels, (2) the use of vapor suppression type of containment, and (3) simplification of startup, shutdown, refueling, and fuel handling equipment. Total savings from these improvements were estimated to be 0.3 mill/kwh.

Reductions in fuel cost are expected from an increase in maximum burn-up to 90,000 MW-d/tonne, and from a reduction in fuel fabrication cost as high density nonsintered fuel fabrication methods become more fully developed. The fabrication cost of plutonium enriched fuel elements in 1975 was assumed to be equal to the present day cost of enriched uranium fuel elements.

Table 6.2 — Summary — Capital and Energy Generation Costs

Reactor Type: Steam-Cooled Fast Breeder

Date: 1975

1. Plant Design Data

a. Net electric power rating, MW	40.0	310
b. Plant operating factor	0.80	0.80
c. Reactor thermal rating, MW	122	825
d. Specific power kwt/kg Pu	604	577
e. Type of fuel	UO ₂ -PuO ₂	UO ₂ -PuO ₂
f. Type of cladding	Hastelloy-N	Hastelloy-N
g. Avg burnup, MW-d/tonne	60,000	60,000

2. Capital Costs

a. Total plant, \$	17,900,000	60,000,000
b. Unit power, \$/kw	448	190

3. Power Generation Costs (mills/kwh)

a. Fixed charges, 14%	8.97	3.90
b. Nuclear fuel cycle		
Fabrication	0.28	0.27
Shipping	0.03	0.02
Depletion	0.90	0.17
Reprocessing	0.34	0.22
AEC use charge	0.38	0.46
Blanket fuel cost	-0.11	-0.08
Total fuel cycle costs	<u>1.82</u>	<u>1.06</u>
c. Operation and maintenance	1.66	0.56
d. Working capital, 12.5%	0.56	0.23
e. Insurance	<u>0.73</u>	<u>0.13</u>

4. UNIT POWER COST (mills/kwh)	13.74	5.72
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6.4 CARBIDE FUELED REACTORS

A carbide fueled reactor core was investigated for the 1975 designs, with the expectation of achieving a higher breeding ratio than with the oxide fuel. A brief listing of reactor and economic characteristics of the PuC-UC designs is given in Table 6.3, and compared to the oxide fueled designs. Higher breeding ratios are obtained with the carbide fuel, because of its higher fuel density. The core breeding ratio of the 300 MW_e plant is greater than one (1.09). The excess of core breeding ratio above one offsets the decrease in reactivity due to fission product buildup. The net effect is a slow decrease in reactivity with burnup, as shown in Fig. 6.1.

Because of a minimum heat transfer surface requirement, proper utilization of the superior power generation capability of carbide fuel can not be made in this reactor. The fuel volume used is the same as with the oxide design, resulting in higher critical mass than the oxide design.

Uranium carbide reacts with steam or water. To reduce the possibility of a serious accident in the event of a cladding failure it would be necessary to provide adequate protection such as by coating individual fuel particles with non-reactive oxides or carbides in conjunction with a vibratory compaction fabrication procedure. Failure of the clad and subsequent failure of a few coatings would be detected by the monitors before coolant contamination became significant. These coatings must be mechanically strong, resistant to thermal shock, abrasion and corrosion by steam, and must be nonreactive with UC-PuC. The most promising coating materials at this time appear to be Al₂O₃, SiC and TiC.

In view of the need of a more extensive research and development program for carbides when used in a steam-cooled reactor, and the higher prospective fuel costs, oxide fuel was selected for the 1975 reactors.

Table 6.3 — Comparison Between 1975 Oxide and Carbide
Fueled Fast Breeders

	40 MW _e		300 MW _e	
	PuO ₂ -UO ₂	PuC-UC	PuO ₂ -UO ₂	PuC-UC
Core Diameter, ft	2.50	2.32	5.43	5.43
Active Height	1.97	1.86	4.40	4.40
Critical Mass, kg Pu ²³⁹	188	188	1370	1590
Core Breeding Ratio	0.53	0.60	0.95	1.09
Blanket Breeding Ratio	0.85	0.90	0.32	0.33
Total Breeding Ratio	1.38	1.50	1.27	1.42
Max. Fuel Burnup, MW-d/tonne	90,000	90,000	90,000	90,000
Avg Fuel Burnup, MW-d/tonne	60,000	60,000	60,000	60,000
Fuel Cycle Costs, mills/ kwh	1.82	2.62	1.06	1.17
Power Generation Costs, mills/kwh	13.7	14.5	5.7	6.1

Fig. 6.1 — Effect of fission product poisoning on reactivity for 300 MW_e steam-cooled 1975 carbide-fueled breeder reactor; core breeding ratio (CBR) = 1.08; enrichment = 11.1%.

7. ALTERNATE DESIGNS

In the course of this study a number of design alternatives was considered for various purposes. A description and evaluation of these designs is given in this section.

The following alternate reactor designs were briefly examined for possible improvement in void coefficient.

1. Low L/D core similar in other respects to the reference design.
2. $\text{PuO}_2\text{-UO}_2$ fueled reactor cooled with D_2O steam.
3. $\text{U}^{233}\text{O}_2\text{-ThO}_2$ fueled reactor cooled with H_2O steam.
4. $\text{PuO}_2\text{-UO}_2$ fueled reactor cooled with quality H_2O steam.

In addition to the above a number of design variations of the reference concept were considered, aimed at cost reduction or increased design flexibility:

1. A pressure tube reactor.
2. Alternate fuel element configuration including plate or matrix type fuel.
3. Steam circulation by means of multipass, steam ejector, and thermopressor systems.
4. An indirect cycle design considered as a means of reducing flooding hazard.

7.1 LOW L/D CORE

An improvement in the void coefficient of the 300 MW_e fast breeder reactor can be obtained with a core which has a large neutron leakage and a small

coolant to fuel ratio. These conditions can be obtained in a core with a small height to diameter ratio. Physical and operating characteristics of such a reactor are presented in Table 7.1. Its nuclear characteristics are given in Table 7.2. The 8.5 ft core diameter surrounded by a 0.25 ft blanket will require a 12 ft ID pressure vessel. The breeding ratio of the low L/D reactor is about the same as that of the reference reactor.

The breeding ratio and critical mass for the low L/D reactor were obtained by means of ten-group diffusion calculations using an equivalent slab model. The effect of voiding was based on the four-group method described in Section 4.6.3.

In this design, advantage is taken of the large core cross sectional area to reduce the ligament spacing between rods so as to yield a lower coolant to fuel ratio than in the reference design, without leading to an unduly high pressure drop. The reduced coolant to fuel ratio and increased neutron leakage from the core contribute to a lower, though still positive, void coefficient.

7.2 D₂O-COOLED, PLUTONIUM FUELED FAST BREEDER REACTOR

The core geometry and the coolant conditions of this design are identical to those of the 300 MW_e, H₂O steam-cooled reference design, described in Table 3.1. Its nuclear characteristics are presented in Table 7.2.

Heavy water steam has considerably smaller slowing down power ($\xi\Sigma_s$) than light water steam. Fewer neutrons are slowed down to lower energies where the ratio of captures to fissions is relatively large. This results in a smaller critical enrichment and a slightly larger overall breeding ratio.

The D₂O-cooled breeder described above has \$2.90 excess reactivity when completely voided. The effect of voiding may become negative in a low L/D core. It is also expected to be lower for an equilibrium cycle isotopic fuel composition.

Power Generation Cost

It is estimated that a 300 MW_e D₂O-cooled fast breeder plant will require a D₂O inventory of about 150 metric tons (tonnes). D₂O purification equipment will

Table 7.1 — Physical and Operating Characteristics
of a 300 MW_e Low L/D Core

Coolant Inlet Temp, °F	610
Coolant Inlet Pressure, psia	1550
Coolant Exit Temp, °F	950
Max. Surface Temp, °F	1350
Fuel Meat OD, in.	0.218
Ligament Spacing, in.	0.035
Total Number of Rods	100,000
Max. Coolant Velocity, ft/sec	200*
Core Pressure Drop, psi	50
Clad to Fuel Volume Ratio	0.294
Coolant to Fuel Weight Ratio	0.0025†
Core Diameter, ft	8.50
Core Height, ft	2.25
Core Length to Diameter Ratio	0.265
Fuel Volume, ft ³	61.0
Max. Steady State Fuel Temp, °F	4500
Radial Blanket Thickness, ft	0.25
Axial Blanket Thickness, ft	1.5

*Compared to 240 ft/sec for the reference design.

† Compared to 0.0036 for reference design.

Table 7.2 — Nuclear Characteristics of 300 MW_e Fast Breeder Reactors

Design	1967 Reference	Low L/D	D ₂ O Steam	U ²³³ -Th
Fuel	PuO ₂ -UO ₂	PuO ₂ -UO ₂	PuO-UO ₂	ThO ₂ -U ²³³ O ₂
Coolant	H ₂ O	H ₂ O	D ₂ O	H ₂ O
Critical mass, kg fissile metal	1656*	1899*	1575*	1272†
Enrichment, a/o fissile element	12.3	12.1	11.7	10.3
Total breeding ratio	1.29	1.30	1.31	1.30
Core breeding ratio	0.96	0.96	1.01	0.92
Blanket breeding ratio	0.33	0.34	0.30	0.38
Median fission energy, kev	112	135	129	N. A.
Median absorption energy, kev	24	50	51	N. A.
Estimated excess reactivity on complete coolant voiding, %	+12.1	+5.6	+2.9	-1.0

*Based on a fuel density of 10.0 g/cm and an initial Pu²³⁹-U²³⁸ loading.

† Based on a fuel density of 9.2 g/cm, and an initial U²³³-Th²³² loading.

be required. D₂O losses represent an additional cost item. However, fuel cycle costs for the D₂O-cooled breeder are slightly lower as a result of the lower critical mass. The cost differential between the 300 MW_e D₂O-cooled reactor and the light water steam-cooled reference design is estimated as follows:

	mills/kwh
D ₂ O inventory charges at 12.5% per year	0.42
D ₂ O losses at 2% per year	0.07
D ₂ O purification, refueling systems and other equipment modifications	0.05
Saving in fuel cycle cost (0.6% reduction enrichment)	<u>(0.07)</u>
Net increase in energy cost	0.47
Total power generation cost	8.02

7.3 U²³³-Th²³² BREEDER REACTOR COOLED WITH H₂O STEAM

The nuclear characteristics of a 300 MW_e light steam-cooled U²³³-thorium fueled reactor were examined in an effort to obtain an improvement in the void coefficient. The identical geometry of the 300 MW_e (1967) plutonium fueled breeder reactor was used.

Critical mass and breeding ratio were calculated using one-group diffusion constants derived on the basis of ten-group core and blanket spectra of the 300 MW_e plutonium reactor. An equivalent spherical model was used in this calculation. The effect of voiding was based on the four-group calculations described in Section 4.6.3. The important nuclear characteristics of the steam-cooled thorium breeder are presented in Table 7.2.

Fig. 7.1 shows a comparison of the fission energy distributions of the two fuels, and indicates a shift to lower energies for the U²³³-Th²³² system. The overall breeding ratio of the thorium reactor is approximately equal to the 1967 plutonium reference design. Its critical mass is appreciably lower.

The void coefficient of the thorium fueled core appears to be negative as indicated in Section 4.6. However, a more accurate investigation of the characteristics of the thorium breeder is required before a reliable comparison of the voiding characteristics of the two systems can be made.

7.4 $\text{PuO}_2\text{-UO}_2$ FUELED REACTOR COOLED WITH QUALITY H_2O STEAM

A reactor cooled with wet steam was investigated with respect to improved voiding and flooding characteristics. A range of coolant inlet qualities from 50 to 100% was examined, in conjunction with an exit coolant temperature of 950 °F.

The inlet quality was selected on the basis of the largest maximum permissible temperature drop across the coolant film, which leads to the smallest core size. Fig. 7.2 shows the maximum permissible temperature drop as a function of coolant inlet enthalpy. This curve is based on a hot channel factor of 1.25, and a maximum cladding surface temperature of 1350 °F. The maximum cladding temperature occurs in the region of the coolant channel where the coolant is superheated. The curve drops off at lower inlet enthalpies because the coolant enthalpy rise through the core is relatively large. The film temperature drop is fairly constant over a wide range of inlet enthalpies. At the 70% inlet quality chosen for this design, the film temperature drop is near the maximum. This quality is sufficiently high so that the two-phase mixture is above "burnout" at all times.

Two reactor designs with a coolant inlet quality of 70% are shown in Table 7.3. One design uses a fuel matrix type element, the other the rod type element. As a result of the lower coolant velocity with quality steam, a rod spacing of 0.035 in. can be tolerated, compared to 0.060 for the reference design. Although the average coolant density is higher for the coolant near the inlet of the core, there is an improvement in coolant to fuel weight ratio over the reference design. The overall core dimensions do not change appreciably.

While the quality steam design with rod type elements has some advantages over the reference design, there is insufficient improvement in the void coefficient to constitute a complete solution to the voiding problem.

Table 7.3 — Steam-Water Fast Breeder Reactor Cooled with Quality Steam —
300 MW_e Plant

Coolant	Superheated Steam	Quality steam	Quality steam
Fuel Element Geometry	Rod fuel	Rod fuel	Matrix fuel
Coolant Inlet Temp or Quality	610 °F (sat.)	70%	70%
Coolant Inlet Pressure, psia	1550	1550	1550
Coolant Exit Temp, °F	950	950	950
Max. Cladding Surface Temp, °F	1350	1350	1350
Fuel Element Rod OD or Coolant Channel ID, in.	0.25	0.125	0.25
Rod Spacing or Fuel Ligament Spacing, in.	0.060	0.095	0.035
Total Number of Rods or Tubes	41,370	78,450	45,000
Max. Coolant Velocity, ft/sec	240	215	215
Core Pressure Drop, psi	100	115	115
Clad to Fuel Volume Ratio	0.294	0.144	0.294
Coolant to Fuel Weight Ratio	0.0036	0.00184	0.0025
Core Diameter, ft	6.0	6.16	5.75
Core Height, ft	5.0	5.33	4.56
Core Length to Diameter Ratio	0.83	0.865	0.79
Fuel Volume, ft ³	53.7	85.5	53.7
Max. Steady-State Fuel Temp, °F	4500	2590	4500

The matrix fuel design shown in Table 7.3 permits a larger reduction of the coolant to fuel weight ratio when used in conjunction with quality steam as the coolant. A difficulty with this design is the relatively small dimensions of the coolant channel (0.125 in.) and fuel ligament spacing between adjacent coolant tubes (0.095 in.). Another disadvantage is the high fuel volume compared to the reference design, which could be reduced only at the expense of smaller fuel ligament spacing or smaller diameter coolant channels.

7.5 PRESSURE TUBE REACTOR

A design of the steam-cooled breeder reactor in which the coolant flows through pressure tubes was briefly examined in the interest of achieving cost reduction resulting from the elimination of the pressure vessel and the use of matrix fuel, and to lift the power plant size restriction inherent in pressure vessel reactors. Also, refueling operations would be simpler in a pressure tube design, and the multipass system of recirculating steam described in Section 7.7 might be applicable.

An elevation cross section of a possible design of a 300 MW_e pressure tube fast breeder reactor is shown in Fig. 7.3. Saturated steam is supplied to the inlet steam drums located in the upper header room. From the steam drums, the steam flows through the inlet steam headers into the pressure tube extensions which take it to the individual pressure tubes. The steam passes downward through the core and blanket. The lower header room contains the outlet steam headers which collect the superheated steam and feed it to the turbine. A plan view of the reactor, showing the arrangement of the steam drums and headers, is presented in Fig. 7.4.

A possible fuel element for this type of reactor is the fuel matrix shown in Fig. 7.5 and discussed in Section 7.6. The matrix element is of hexagonal shape and contains 19 coolant tubes. A small clearance is provided between adjacent elements to allow for thermal expansion.

A difficulty with the pressure tube design arises from the refueling operation. The close spacing of the elements necessitates removal of the entire pres-

sure tube whenever a fuel element is to be removed. In the design shown, removal of a pressure tube requires breaking the tube at the top and bottom. To gain access to some of the inner tubes, it is necessary to remove other tubes first. Unless a more suitable design can be developed, this operation would be extremely cumbersome and time consuming.

7.6 ALTERNATE FUEL ELEMENT CONFIGURATIONS

In addition to rod type elements, a plate type fuel and matrix fuel penetrated by coolant tubes were examined. (See Fig. 7.6). It was found that neither of these designs offers significant advantages over the rod type geometry at the present time. It is possible, however, that the matrix fuel geometry would lend itself more readily to vibratory compaction of fuel and thereby offer the possibility of lower fuel costs upon further advances in this method of fabrication.

In a surface temperature-limited core, such as the steam-cooled breeder, the total amount of heat transfer surface required is largely independent of fuel geometry. For a given fuel volume, it is desirable to minimize the coolant volume, both from the standpoint of overall core volume and from flooding and voiding considerations. In comparing geometries it is, therefore, necessary to compare configurations which have similar steam to fuel volume ratios. Such a comparison is made in Table 7.4, though at conditions somewhat different from those of the reference design.

It will be noted that for nearly comparable coolant to fuel ratios the flat oxide plate requires a meat thickness of 0.109 in. and a plate spacing of 0.071 in. Oxide plates of this thickness are still in the development stage and can not be considered suitable for near term cores. Plates made of a stainless steel-PuO₂-UO₂ matrix would lead to core dimensions similar to those of the oxide plates. However, the steel to fuel ratio would be very high (of the order of seven), and the reactor would have a breeding ratio less than one.

The matrix fuel element with a coolant to fuel ratio comparable to the reference design is shown in Fig. 7.5 and has a coolant tube ID of 0.160 in. and a fuel ligament spacing of 0.070 in. The fuel ligament spacing of this element is

Table 7.4 — Comparison Between Rod, Flat Plate, and Matrix Fuel Elements

Coolant Exit Pressure, 1800 psi
Coolant Exit Temperature, 975°F
Max. Surface Temperature, 1250°F

Type of Element	Rod	Flat Plate	Fuel Matrix
t = Fuel "meat" diameter, thickness, or ligament spacing, in.	0.240	0.109	0.070
δ = Cladding thickness, in.	0.0135	0.010	0.010
S = Coolant channel dimension, in.	0.050	0.071	0.160
Mass velocity, (lb/hr-ft ²) × 10 ⁻⁶	1.25	1.33	1.31
Core pressure drop, psi	60		50
Max. flow velocity, ft/sec	150	160	157
Surface area, ft ²	21,000	20,000	20,400
Core diameter, ft	6.1	6.1	5.9
Core height, ft	6.5	6.5	6.3
Fuel volume, ft ³	96	91	86
Steam/fuel volume ratio	0.67	0.67	0.70
Clad/fuel volume ratio	0.235	0.401	0.186
Max. fuel temp, °F	3000	2000	1570

too small for successful fabrication by vibratory compaction on the basis of current technology.

The matrix fuel shown in conjunction with the quality steam design, Table 7.3, also requires small coolant tube diameter and ligament dimensions. A small ligament spacing is characteristic of this type of geometry for this particular type of reactor by virtue of the surface heat transfer limitation. The fuel matrix geometry leads to a considerably lower clad to fuel volume ratio and a lower fuel temperature than either the flat plate or rod type geometry. In reactors where the latter considerations are governing the fuel matrix element may be of considerable interest.

7.7 STEAM CIRCULATING DEVICES

In a Loeffler cycle steam-cooled reactor plant, more than one half of the steam leaving the reactor must be recirculated in order to vaporize the condenser feedwater. The most direct way of recirculating this steam is by means of steam blowers. This is the method incorporated in the reference design.

Steam blowers require a comparatively large power input. Also, some development work is necessary to provide low steam leakage shaft seals for existing pump designs. For these reasons, alternate methods of steam circulation have been examined. These are (1) the multipass system, (2) the steam ejector, and (3) the steam thermopressor. Schematics of the corresponding circuits are shown in Fig. 7.7. None of these systems appears particularly attractive for the present application.

7.7.1 Multipass System

In the multipass system, the entire coolant pressure rise is provided in the liquid phase. The feedwater is brought to a high pressure by means of conventional pumps, and enters a closed type heat exchanger where it is vaporized by steam from the reactor. The saturated steam makes one pass through the reactor, returns to the heat exchanger where it gives up its heat to the feedwater. The steam is once more passed through the reactor, then through the heat exchanger,

and then again through the reactor. From there the entire steam flow goes to the turbine. At this point, the steam is at a considerably lower pressure than the feedwater, but no pumping of the steam is required.

The pumping power required for this system is small, because only the liquid is pumped. The large steam blowers are eliminated. However, a fairly costly heat exchanger is required in their place. Moreover, this system requires multipass flow through the reactor, creating a difficult design problem with a pressure vessel reactor. The multipass system might be feasible with a pressure tube reactor, but an unmoderated reactor such as the steam-cooled fast breeder does not lend itself to a simple pressure tube design.

7.7.2 Steam Ejector

In the steam ejector system, the feedwater is pumped to a high pressure, vaporized in a heat exchanger by means of recirculated reactor steam and then accelerated in a convergent-divergent nozzle. The resulting high velocity vapor stream is mixed with the steam from the heat exchanger and the mixture is decelerated. The exit pressure of the combined steam flow will, in general, be higher than the initial pressure of the recirculated steam.

For an initial steam pressure of 975 psi and a feedwater pressure of 2400 psi, an ejector pressure rise of 50 psi was calculated. This pressure gain is insufficient to overcome a pressure drop of about 150 psi expected in the reactor, piping, and heat exchanger.

The steam ejector system, like the multipass system, requires a surface type heat exchanger for vaporizing the feedwater.

7.7.3 Steam Thermopressor

The steam thermopressor is a pumping device that could replace both the steam drums (Loeffler boiler) and the steam blowers. Its operation is derived from the principles of compressible flow which indicate that when a flowing gas stream is cooled, the total pressure increases, provided the heat transfer can be accomplished in the absence of adverse frictional pressure loss. The evaporation

of water droplets in the bulk of steam leads to large heat transfer rates and pressure increases may be obtained.

The steam thermopressor consists of a converging nozzle, in which the recirculated steam is accelerated. This is followed by a large diameter duct in which the feedwater is accelerated, sprayed into the flowing steam, and evaporated. A sketch of such a device is shown in Fig. 7.8.

Calculations were made for a reactor coolant pressure of 2000 psi and a feedwater pressure of 5000 psi. With a 13.5 in. duct diameter and a heat transfer rate of 10^6 Btu/hr-ft² between steam and water droplets, a pressure rise of about 25 psi was calculated. This rise is insufficient to overcome the expected pressure drop in the other components of the circuit. Accordingly, the steam thermopressor concept is not considered applicable to the plants of interest here.

7.8 INDIRECT CYCLE

Some consideration was given to the relative merits of the direct cycle plant of the reference design (in which reactor steam passes directly to the turbine) and an indirect cycle plant. In the latter only superheated steam circulates in the primary loop. The primary steam is made to raise superheated steam in an intermediate heat exchanger. The indirect cycle has the advantage that it would minimize the danger of flooding of the reactor. However, since flooding effects can be offset by the addition of resonance poisons in the core, there is no compelling reason to go to the indirect cycle with its higher capital cost and lower thermal efficiency. An additional consideration in favor of the selection of a direct cycle is the presence of a large reservoir of water in the Loeffler boilers in the reactor. These are useful in emergency cooling of the reactor in case of pump failure or a depressurization accident. (See Section 4.4.)

Fig. 7.1 — Energy distributions of fissions in $\text{Pu}^{239}\text{O}_2\text{-U}^{238}\text{O}_2$ and $\text{U}^{233}\text{O}_2\text{-Th}^{232}\text{O}_2$ 300 MW_e breeders

Fig. 7.2 — Maximum permissible coolant film temperature drop as a function of coolant inlet enthalpy

Fig. 7.3 — Steam-cooled fast breeder reactor — alternate arrangement—
pressure tube reactor elevation

Fig. 7.4 — Steam-cooled fast breeder reactor —
alternate arrangement — pressure tube reactor
plan and sections

Fig. 7.5 — Steam-cooled fast breeder reactor — fuel element for pressure tube reactors

Fig. 7.6 — Alternate fuel element configurations

Fig. 7.7 — Alternate steam circulating devices;
—steam; ---water.

Fig. 7.8 — Steam thermopressor; region I, steam acceleration; region II, condensate acceleration; region III, droplet evaporation; region IV, mixture deceleration.

8. RESEARCH, DEVELOPMENT, AND TEST PROGRAM, NEAR TERM DESIGNS

The research and development program for the steam cooled fast breeder reactor is estimated to cost \$12,100,000. Of this amount, \$8,200,000 is the cost of a reactor experiment which would be required to study dynamic and safety characteristics of this type of reactor. A detailed breakdown of development costs as well as a time schedule is presented in Table 8.1.

8.1 CLADDING DEVELOPMENT

For the near term designs, the cladding must withstand a maximum steady state surface temperature of 1350°F. It appears that either Inconel or Inconel-X will be suitable for this service. The higher strength characteristics of Inconel-X make it the preferred material.

Autoclave tests of Inconel and Inconel-X in pure water at 500-680°F have been reported by various facilities.⁸⁻¹² These tests show that corrosion rates are very low (<1 mil per yr), but that the film produced is very loose in 600-680°F water. Inconel showed no tendency for stress-corrosion cracking and exhibited no galvanic effects with 300-series stainless steels.¹⁰ These results were also obtained in a test using hydrogenated, ammoniated, and aerated, high-pressure, high-temperature (600°F), high-velocity water.^{13,14} However, in this test, some pitting was found in Inconel under stress.¹⁴

Low corrosion rates were obtained with Inconel in a 1000 hr exposure at 585°F to ammoniated or lithiated steam.¹⁵ Tests with 500°F steam yielded no pits or cracks and a low corrosion rate (<1 mil per yr) after 500 hr.⁸ Inconel

Table 8.1 — Research, Development, and Test Program
for the Near Term Fast Breeders

Development Item	Cost, \$*	Lapsed Time, yr	
		1961	1965
1. Reactor experiment			
A. Design and construction	6,200,000	▲	▲
B. Operation	2,000,000		▲
2. Cladding development and test (including out-of-pile steam cooled loop, steam purchased, not generated)	500,000	▲	▲
3. Fuel element development			
A. Fabrication methods development	700,000	▲	▲
B. In-pile capsules	350,000		▲
4. Development of flooding poisons	250,000	▲	▲
5. Mechanical equipment development	2,000,000	▲	▲
6. Steam circulator development	<u>100,000</u>	▲	▲
Total	\$12,100,000		

*Over and above present plutonium and uranium oxide development program.

showed little attack in a hot water-steam loop¹⁶ and did not react appreciably with water vapor at 2000°F.¹⁷

Although no corrosion data for Inconel in 1350-1500°F steam is available, a major supplier felt that Inconel would be satisfactory in steam under oxidizing conditions.¹⁸ This was based on low oxidation in air, and the results of superheater tube studies which indicated macroscopically and microscopically that very little attack occurred.¹⁹ In these studies, impact strength and elongation were also determined. Both these properties remained fairly constant at room temperature and in 1200-1500°F, 20 ft/sec steam. However, the Inconel tubes bulged excessively from high-temperature creep.

Some creep and stress-to-rupture properties of Inconel and Inconel-X are plotted in Figs. 8.1 to 8.4. The high temperature properties of Inconel-X are superior to those of Inconel and type 316 stainless steel. The high temperature properties of Inconel, on the other hand, are similar to those of type 304 stainless steel. The only steam corrosion data found with Inconel-X at 1350°F were with supercritical steam at 5000 psi.²⁰ The corrosion rate for Inconel-X, about 1 mill/year, appears to be higher than for other stainless steels. However, it is still low enough to make Inconel-X a feasible cladding material. The corrosion rate could probably be improved with selective heat treatment.

Before Inconel-X could be used in a full scale reactor some in-pile and out-of-pile strength and corrosion tests will be required. The approximate test conditions would be as follows:

1. Short term corrosion and erosion tests in flowing steam under:
 - a. temperatures up to 1500°F,
 - b. pressures up to 2500 psi,
 - c. velocities up to 300 ft/sec.
2. Strain-cycling tests in steam atmosphere at temperatures up to 1500°F and pressures up to 2500 psi.
3. Creep and stress-to-rupture tests in steam atmosphere at temperatures up to 1500°F and pressures up to 2500 psi.

4. Out-of-pile tests of fuel assemblies representative of the core elements under conditions approximating actual reactor operation.

8.2 FUEL ELEMENT DEVELOPMENT

Fabrication methods for rod type uranium-plutonium oxide fuel elements are currently being developed in conjunction with the fabrication of the second core of the PRTR (Plutonium Recycle Test Reactor) at Hanford. A new plutonium fuel fabrication facility was completed there recently. A majority of the steps used in the fabrication process has already been proven out. The incentive for any future development program will therefore lie principally in the achievement of lower fabrication costs without compromising important fuel characteristics such as density, fission gas retention, and inertness and physical stability in the presence of the coolant brought about by a defected cladding.

Improved Fabrication Techniques

Although the cost of plutonium fuel elements will always be greater than that for a similar configuration using uranium, because of the special precautions required for handling the former, this cost differential can be reduced to a point where it will cease to be a major consideration. New methods of fabricating oxide elements are already under development and promise substantial cost reductions. Among these are the following:

1. Vibratory Compaction

The process of vibratory compaction currently involves vibration of the cladding tube to frequencies up to 5000 cps with accelerations of 15-20 g. The fuel, in the form of particles, is added continuously to the vibrating tube while the frequency is varied continuously over a range which straddles the resonant frequency of the containing tube. Vibratory compaction is particularly suited to plutonium containing elements since it reduces the number of fabrication steps required and lends itself to remote handling. In order to increase the average attainable fuel density, additional work is required to optimize process variables such as particle size, distribution and shape, frequency and thrust. Experiments will also be required to obtain data on the thermal conductivity of both unirradiated

and irradiated fuel elements produced by vibratory compaction.

2. Pelletizing and Drawing

In this process, fuel pellets are loosely placed into a fuel tube which is then drawn tightly over the pellets reducing the diameter of the larger pellets slightly. This method promises to provide high fuel density without requiring grinding of fuel pellets.

3. Hot and Cold Swaging

The fuel tube is filled with oxide powder. The tube diameter is then reduced, whereby the powder inside is compacted to a high density.

Irradiation Tests

Within the next year or two, considerable information will become available on the burnup capability of UO_2 - PuO_2 fuel. However, individual fuel element configurations or fuel elements produced by a particular fabrication process will still require irradiation testing to ascertain the maximum burnup of 50,000 MW-d/tonne desired for the near term plants. Capsule testing of prototype fuel elements will be necessary. The cost of this program given in the table does not include the cost of the required reactor facility. It has been assumed that this facility will be available free of charge for neutrons.

8.3 CROSS SECTIONS

Theoretical and experimental studies are required to obtain accurate values for the cross sections of Pu^{239} , Pu^{240} , Pu^{241} and nickel in the high and intermediate neutron energy range. These studies need not be charged against this particular reactor concept, but may be supported by general funds for breeder reactors.

8.4 REACTOR SAFETY

A number of studies and experiments are required to ascertain the safety of oxide-fueled breeder reactors. These involve investigations of flooding and voiding characteristics, and temperature and power coefficients.

Reactor Flooding

Calculations indicate that the 300 and 40 MW_e breeders will generate a large amount of excess reactivity in the event the core becomes flooded with water. One solution to this problem is to place permanently a neutron absorber in the core that will successfully compete with plutonium at low energies but absorb few neutrons under normal operating conditions. Although it is clear that materials such as hafnium, silver, indium, gadolinium and others would be suitable for this purpose, further study and experimentation would be required to select the best material.

Alternately, it may be possible to control adequately the reactivity increase resulting from progressive core flooding with a control rod system and a water soluble poison in the Loeffler boilers. This poison would move into the reactor in the event that the water from the Loeffler boiler is carried into the reactor, but would stay in the Loeffler boiler under normal operation. Again this scheme would require further study and possibly testing.

Temperature, Power, and Void Coefficients

Multigroup diffusion calculations indicate that, as a result of the strongly positive void coefficient, the overall temperature coefficient of reactivity for the oxide breeder is positive. These computations did not include the negative effects of axial streaming. More refined calculations using a larger number of energy groups in the region between 10,000 and 1 ev, together with reactor dynamics experiments will be required to determine actual characteristics. Experimental verification of the sign and magnitude of the coolant density coefficient of reactivity is required. The reactor experiment would require expenditures and efforts of the order of magnitude of one of the BORAX experiments for boiling water reactors.

8.5 MECHANICAL EQUIPMENT DEVELOPMENT

The breeder reactor requires relatively frequent reshuffling of fuel elements once every 2 to 6 months. To reduce reactor down time, it is important to

develop means and equipment for performing the refueling operations with a minimum reactor down time.

8.6 MAIN STEAM CIRCULATORS

The design shown requires comparatively large main steam circulators operating at high steam pressures. Some development work will be required to provide low steam leakage shaft seals for existing circulator designs. However, this problem is well within the capabilities of present day technology.

Fig. 8.1 — Creep and stress-to-rupture vs temperature for Inconel and Inconel-X

Fig. 8.2 — Yield strength vs temperature for Inconel and Inconel-X

Fig. 8.3 — Per cent elongation vs temperature for Inconel and Inconel-X

Fig. 8.4 — Rupture data for Inconel and Inconel-X at 1350 and 1500 °F

9. RESEARCH, DEVELOPMENT, AND TEST PROGRAM, 1975 DESIGNS

Development work beyond that required for the near term designs will be mainly for the purpose of obtaining higher cladding temperatures and increased fuel burnup.

The proposed program is summarized in Table 9.1. Its cost is estimated at \$3,000,000; this amount is based on the assumption that a reactor test facility would be built for the near term development program and would be available for reactor experiments with alternate fuel configurations.

9.1 CLADDING DEVELOPMENT

For the 1975, 300 and 40 MW_e designs, steady-state cladding temperatures of 1500°F are proposed. Inconel-X, Hastelloy-N, and Hastelloy-X may be considered for this service. The high temperature properties of these materials are shown in Figs. 9.1 to 9.5.

Hastelloy-N was developed at Oak Ridge as a container material for molten salt.²¹ Data published by Haynes Stellite report the oxidation penetration of Hastelloy-N in air at 1600°F to be less than 1.5 mils/yr.

Hastelloy-X has been chosen as the cladding material for the ML-1 gas-cooled high temperature reactor²² where a cladding temperature of 1750°F is used.

The relative cost of Inconel and Hastelloy-N as cladding material has been obtained using 1/4 in. diameter tubing with a 12 mil wall as a basis. The present costs are approximately \$30 and \$115 per 100 ft length, respectively.²³ On a cost per pound basis, this would be \$9 and \$34 for Inconel and Hastelloy-N, respectively.

Table 9.1 — Research, Development, and Test Program
for the 1975 Fast Breeders

Development Item	Cost, \$ *	Lapsed Time-Yr		
		1965	1970	1975
1. Cladding development (incremental cost over near term program)	350,000		▲	▲
2. Fuel fabrication development (incremental cost over near term program)	300,000			▲
3. Fuel irradiation tests (capsules)	350,000			▲
4. Continued operation or reactor experiment	<u>2,000,000</u>	▲	▲	
Total	\$3,000,000			

*Incremental cost above near term research and development program presented in Table 8.1.

In addition to routine testing similar to that required for the near term designs cladding, research and development work will be required to investigate and develop the corrosion and erosion resistance of these alloys in a steam atmosphere at, or slightly above, 1500°F.

9.2 FUEL ELEMENT FABRICATION

A relatively small effort will be required to develop techniques for fabricating smaller fuel rod diameters for cores with higher power densities.

9.3 IRRADIATION TESTS

In-pile irradiation testing of oxide elements will be required. For 1975, burnups in excess of 90,000 MW-d/tonne are desired. It will be necessary to conduct capsule experiments to determine the effects of this high burnup.

9.4 REACTOR SAFETY

Extension of the dynamic behavior experiments to the higher specific powers of the 1975 designs will be required.

Fig. 9.1 — Yield strength vs temperature. (Curves were plotted from data published by International Nickel Co. and Haynes-Stellite Co.)

Fig. 9.2 — Elongation vs temperature. (Data were obtained from curves in Chemical Engineering, May 4, 1959 and Reactor Handbook.)

Fig. 9.3 — Rupture data, annealed material, at 1500 °F;
 — Hastelloy-N; ---- Inconel; --- Hastelloy-X;
 -.-.- Inconel-X. (Curves were plotted from recent
 data published by International Nickel Co. and Haynes-
 Stellite Co.)

Fig. 9.4 — Stress to produce secondary creep rate of 0.1% in 1000 hr vs temperature. (Data were obtained from reactor handbook on materials.)

Fig. 9.5 — Stress to produce rupture in 1000 hr vs temperature
(Data from reactor handbook of materials.)

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