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STEAM-COOLED POWER REACTOR EVALUATION
STEAM-COOLED FAST BREEDER REACTOR

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April 15, 1961

Nuclear Development Corporation of America
White Plains, New York

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April 15, 1961

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with the United States Atomic Energy Commission

NUCLEAR DEVELOPMENT CORPORATION OF AMERICA
White Plains, New York

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1. INTRODUCTION

Current interest exists in the development of breeder reactors because of their promise to yield economic power costs within 5 to 15 years while helping to conserve fissionable material resources. This interest does not stop at 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 mils/kwh.

The reactor concept treated in this report combines the advantages of breeder reactors with those of direct cycle high efficiency nuclear superheaters. 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.

This study is sponsored by the USAEC, under Contract AT(30-1)-2303 (Project XII), and is one of several conceptual design and economic studies contracted by the AEC in a comparative study of nuclear superheater reactors.

The economic and technical ground rules used as a basis of this evaluation were specified by the AEC. The salient items 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	Usual AEC schedule

Value of PuO ₂	\$11.50/g of plutonium metal
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, near term plants	30% of fissionable material
Max. burnup, 1975 plants	90,000 MW-d/tonne
Max. central UO ₂ temp	4500°F (k=1)
Max. Inconel fuel cladding surface temp, near term plants	1350°F
Max. fuel cladding surface temp, 1975 plants	1500°F*
Near term reactors	
300 MW _e	Operable by 1967
40 MW _e	Operable by 1965
Long term reactors	
300 and 40 MW _e	Operable by 1975

Cost data developed in the course of this study have been reviewed by Kaiser Engineers and the AEC and are reported in the Kaiser Engineers Report to the AEC, KE 60-31.

* Not established by AEC.

2. SUMMARY

The conclusions reached in this study are as follows:

1. Attractive economics and good breeding characteristics (breeding ratios from 1.27 to 1.42) can be achieved in steam-cooled $\text{PuO}_2\text{-UO}_2$ fueled fast reactors. Low capital costs are obtained by virtue of a compact reactor core and the absence of large heat exchangers and complicated process systems.
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. Additional studies are required to check these calculations and determine the effect on reactor safety.

2.1 NEAR TERM REACTORS

2.1.1 Reactor and Power Cycle, Near Term Design

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. 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.

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 by-pass steam is used to raise the condenser feedwater to 495°F before it enters the Loeffler boiler. The important features of the plant cycle are shown in the simplified flow diagrams, Figs. 3.3 and 3.10.

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. About 10% of the coolant flows in a parallel pass through the radial blanket. Important reactor characteristics are shown in Table 2.1. The core fuel elements are 0.25 in. diameter, Inconel clad oxide rods. $\text{PuO}_2\text{-UO}_2$ fuel yields good breeding ratios, has good corrosion resistance to steam and permits high burnups. Fuel and fuel element fabrication technology are well developed and further advances are expected. Inconel 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 fuel in the blanket is depleted (less than 0.35% U^{235}) UO_2 . For the 300 MW_e reactor, most of the conversion of U^{238} 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.

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, the steam can be led into the fuel storage decay pool where it will condense. The reactor can be cooled by these methods 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 affected by means of control rods located in the core and blanket. They are made of stainless steel clad B_4C rods located in coolant channels interchangeable with those of the fuel. They are activated by individual electric motors located on top of the shield cover.

Refueling of the core takes place approximately every 50 days. An inward radial shift type of fuel management is used to obtain a uniform fuel burnup. It is estimated that refueling of the core can be completed within 48 hr. 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

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, %	34.2	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
Central fuel temp, °F	4500	4500	4500	4100
Rod clearance, spacing, in.	0.060	0.045	0.040	0.035
Clad material	Inconel	Hastelloy-N	Inconel	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

tolerable amount of resonance absorber, such as hafnium or indium, to the fuel.

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 may be a problem. Preliminary calculations indicate that the temperature and void coefficient of the 300 MW_e reactor are positive. Additional studies of the reactor behavior during voiding accidents are required before the effect of this problem can be 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. The characteristics of such a plant are shown in Table 2.1. The design is based on PuO₂-UO₂ fuel with a maximum burnup of 90,000 MW-d/tonne. An alternate design with PuC-UC fuel leads to higher power generation costs and would require the development of carbide fuel with corrosion resistance to steam.

2.3 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, PuO₂-UO₂ fuel elements.
3. Development of fabrication techniques for PuO₂-UO₂ fuel elements.
4. Improved knowledge of fast cross sections for Pu²³⁹ and higher Pu isotopes.
5. 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 \$3,000,000.

3. DESCRIPTION OF NEAR TERM PLANTS

This section contains a description 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 3.9 and the 40 MW_e drawings are shown in Figs. 3.10 to 3.12.

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 Fig. 3.3 and 3.10).

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. 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.

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 (Fig. 3.5 and 3.11). The direction of coolant flow through core and blanket is downward. Reactor characteristics are listed in Table 3.1.

Table 3.1 — Summary of Reactor Plant Characteristics —
Near Term Steam-Cooled Fast Breeders

	40 MW _e (1965)	300 MW _e (1967)
Heat Balance		
Net plant power, MW _e	40.0	300.0
Total reactor power, MW _t	127	876
Turbine Cycle Conditions		
Throttle temp, °F	920	945
Throttle pressure, psia	1415	1415
Total steam flow, lb/hr	0.49×10^6	2.69×10^6
Reactor Description		
Reactor vessel		
ID, ft	8.33	10.67
Inside height, ft	16.67	31
Wall thickness, in. (cylindrical portion)	5.85	7.5
Material	Low alloy steel with $\frac{1}{4}$ in. stainless (304) cladding	
Design pressure, psig	1860	1860
Design temp, °F	800	800
Reactor core		
Active equivalent diam, ft	2.54	5.95
Active height, ft	2.65	4.96
Active core volume, ft ³	13.4	138
Total fuel loading, kg fuel	1483	13,460
Avg Pu ²³⁹ content, % by weight	17.6	12.3
Structural material	Stainless steel	Stainless steel
Neutron moderator	None	None
Moderator to fuel ratio	None	None
Blanket		
Material	Depleted UO ₂	Depleted UO ₂
Axial thickness, ft	1.5	1
Radial thickness, ft	1.5	1
Core fuel element		
Fuel material	UO ₂ -PuO ₂	UO ₂ -PuO ₂
Fuel element geometry	Rods	Rods
Clad material	Inconel	Inconel
Fuel rod OD, in.	0.25	0.25

	40 MW _e (1965)	300 MW _e (1967)
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
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

	40 MWe (1965)	300 MWe (1967)
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, nut	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	615,000	615,000
Avg core heat flux, Btu/hr-ft ²	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
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 (perturbed)	137,000	131,000

	40 MW _e (1965)	300 MW _e (1967)
Avg heat flux, Btu/hr-ft ²	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
Suggested Containment		
Design criteria	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	1/2 in. steel	1 in. steel
	shell	shell

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

The core and blanket fuel subassemblies (Fig. 3.7 and Fig. 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 first over the upper axial blanket elements, then through the core elements, and finally through the lower blanket elements. The core fuel cluster supplies 96% of the total heat transferred to the steam.

The number of fuel clusters in the core is based on a maximum 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% U²³⁵) uranium oxide. The fuel is clad with 0.015-in. thick Inconel tubes. The tubes are filled with helium to minimize the temperature drop between fuel and cladding. A small helium-filled space is provided inside the top end of the fuel rods to allow for axial expansion of the fuel, and to minimize the pressure caused by the release of fission product gases. Spacing between adjacent fuel rods 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 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 3.12.) The radial blanket fuel element subassembly 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 and are described in Table 3.2.

The control rods are driven from above by a constant speed motor supported on top of the upper 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 3.11.) 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 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

Table 3.2 — Control Rods — B₄C — 50% Enriched with B¹⁰

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 ₄ C clad with SS	120	0.48	2	0.23
	Safety rods in core	7 rod cluster, 0.4 in. diam B ₄ C	150	1.58	8	0.72
	Shim rods in blanket	7 rod cluster, 0.25 in. diam B ₄ C	60	0.25	4	0.06
40 MW _e	Shim rods in core	7 rod cluster, 0.37 in. diam B ₄ C	60	0.48	2	0.12
	Safety rods in core	7 rod cluster, 0.4 in. diam B ₄ C	80	0.83	4	0.20
	Shim rods in blanket	7 rod cluster, 0.25 in. diam B ₄ C	60	0.25	4	0.06

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 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. 3.11). 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 yr of reactor operation.

3.2 NUCLEAR ANALYSIS

The nuclear characteristics for the 300 and 40 MW_e steam cooled fast breeder reactors were calculated by 10-group diffusion theory calculations using a one-dimensional, spherical reactor model. 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. The effects of fission product buildup and higher isotope production are discussed below and in Section 4.3.

The two most important nuclear quantities of Table 3.1 are the critical mass and breeding ratio values. An initial critical mass of 1656 kg of Pu²³⁹ is predicted for the 300 MW_e oxide breeder. This value is about 25% higher than the number reported for a comparable sodium-cooled fast breeder* after correcting for differences in the steel-to-fuel ratio. The reason for this difference is attributed to the spectrum degradation effects of hydrogen, which tends to lower the average neutron regeneration factor, $\bar{\eta}$, for plutonium, and increase the capture-to-fission ratio of U²³⁸. The critical mass for the 300 and 40 MW_e reactors corresponds to 0.50 and 0.46 MW_t/kg Pu²³⁹, respectively.

The initial core breeding ratio for the 300 MW_e oxide design is very close to unity (~0.97), 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.*

*P. Greebler and P. Aline, Parametric Analysis of a PuO₂-UO₂ Fueled Fast Reactor, Proceedings on the Conference on the Physics of Breeding (Oct. 19-21, 1959).

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.

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 PuO₂-UO₂ breeder* 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.

3.2.1 Equilibrium Fuel Cycle

The critical mass and breeding ratio figures given above are based on a core freshly fueled with Pu²³⁹O₂-U²³⁸O₂. 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 the initial Pu²³⁹ loading.

The mixture of 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 Pu²³⁹/U²³⁸ ratio from 0.140 to 0.115. This is due to the buildup of Pu²⁴¹, which has an average η of 2.89 compared with 2.26 for Pu²³⁹. The decreased enrichment of fissile isotopes produces a considerable improvement in the overall breeding ratio.

*Data from NDA 2140-2, Carbide Fuel Development, Phase I Report, Period May 15 to September 15, 1959 (Oct. 15, 1959).

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

1. Critical Mass,* total kg of Pu ²³⁹ and Pu ²⁴¹	1441	
2. 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
3. Median Fission Energy, kev	~101	
4. Initial Breeding Ratios†		
Core	1.165	
Blanket	<u>0.349</u>	
Total	1.514	

* Based on fuel density of 10.0 gm/cm³

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

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

Fig. 3.2 — Average core neutron flux spectrum — 40 MW_e steam-cooled fast breeder

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

Fig. 3.4 — 300 MW_e steam-cooled fast breeder reactor — flow diagram — main power loop

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

Fig. 3.6 — Steam-cooled fast breeder reactor — 300 MW_e — 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

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Fig. 3.10 — 40 MW_e steam-cooled fast breeder reactor — simplified flow diagram

Fig. 3.11 — 40 MWe steam-cooled fast breeder reactor — elevation

Fig. 3.12 — 40 MW_e steam-cooled fast breeder reactor — plan and section through core

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.

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 by-pass 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.

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.

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 by-pass 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-by-pass 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 by-pass 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 MW_e 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 B₄C with 50% B¹⁰ enrichment, clad with type 304 stainless steel tubing. B¹⁰ has a significant absorption cross section for fast neutrons due to the B¹⁰ (n,α) Li⁷ reaction.

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 B¹⁰ 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 MW_e 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 MW_e 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:

1. burnup of Pu²³⁹, and
2. fission product poisoning.

for the 300 MW_e oxide-fueled plant, the first effect is unimportant relative to the second. The reason for this is that there is little change in the Pu²³⁹ 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% Pu²³⁹ 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 also 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 ($\sim \$4$) at startup. Since it is necessary to limit the available excess reactivity to slightly less than \$1.00, additional excess reactivity must be forthcoming from 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.

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 Section 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 port-holes. 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 3.11.)

After the operations within the reactor vessel are performed, the port-hole 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 by-passes 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 by-passed, the turbine main valve closes and a by-pass 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 by-pass 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 by-pass line is closed and the Diesel-powered steam circulators in parallel with the main steam circulators are used to pump the steam. The turbine by-pass 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 by-passed 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 AND VOIDING

Flooding of the steam-cooled breeder coolant channels with liquid H_2O 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 require-

ments, the behavior of the resonance absorbers indium and hafnium were investigated.

The k_{∞} of the 300 MW_e reference design containing U²³⁸, Pu²³⁹, 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 Pu²³⁹ resonance at 0.3 ev, and a group below the Pu²³⁹ 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 U²³⁸ are needed to provide sufficient absorption to make the reactor safe upon flooding. The addition of the required amount of hafnium is estimated to require an increase in critical mass of less than 2%.

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.

Voiding

In regard to voiding of the reactor (i.e., loss of steam pressure), preliminary calculation on the 300 MW_e oxide breeder indicates that reactivity increases with decreasing coolant density.

Additional studies must be conducted to study this effect more closely and to suggest means of minimizing or reversing the effect on reactivity or loss of coolant. The increase in reactivity with loss of coolant is attributed to the rise in η for the fuel resulting from a hardening of the spectrum, which is not compensated by the effect of increased leakage. The combined effect including leakage is illustrated in 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 — Neutron gain per neutron absorbed in fuel vs neutron energy, for 300 MW_e mixed oxide-fueled steam-cooled fast breeder

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5. 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%.

5.1 PLANT CHARACTERISTICS

The characteristics of the 1975, 300 MW_e and 40 MW_e plants are shown in Table 5.1. Hastelloy-N has been specified 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 6.3 where fuel costs for three fuel rod diameters are compared.)

5.2 NUCLEAR CHARACTERISTICS

The important nuclear characteristics for the 1975 steam-cooled reactor plants are given in Table 5.1.

Both the 300 and 40 MW_e 1975 reactors are smaller than the corresponding near term plants; the 300 MW_e 1975 plant has 11% less fuel volume while the

Table 5.1 — 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. (cylindrical portion)	5.6	9.6
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 5.1 — (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
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

Table 5.1 — (Continued)

	40 MW _e	300 MW _e
Avg specific power, kw _t /kg of Pu	604	577
Fuel management	Inward radial shift	
Avg fuel burnup in exit fuel, MW-d/mt	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.

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.

5.3 CARBIDE-FUELED REACTORS

With the expectation of achieving a higher breeding ratio, a carbide fueled core was investigated. Core characteristics are shown in Table 5.2. Higher core and blanket breeding ratios are obtained with the carbide. The core breeding ratio of the 300 MW_e plant is greater than one (1.08). The excess of breeding ratio above one offsets the decrease in reactivity due to fission product buildup. The net effect is a slow decrease in reactivity with operating time, as shown in Fig. 5.1.

The 300 MW_e carbide core requires a somewhat larger plutonium inventory than the oxide breeder. Fuel cycle costs are slightly higher, as a result of the higher fabrication cost of carbide fuel.

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 nonreactive 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 required with carbides and the higher prospective fuel costs, oxide fuel was selected for the 1975 reactors.

**Table 5.2 — 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

Fig. 5.1 — Effect of fission product poisoning on reactivity for 300 MW_e steam-cooled 1975 carbide-fueled reference design; core breeding ratio (CBR) = 1.08; enrichment = 11.1%.

6. SELECTION OF NEAR TERM PLANT CHARACTERISTICS

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

6.1 HOT SPOT AND HOT CHANNEL FACTORS

Core heat transfer calculations were based on hot spot and hot channel factors as shown in Table 6.1. 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.

6.2 HEAT TRANSFER AND PRESSURE DROP RELATIONS

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

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

where Cp = 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

Table 6.1 — Steady State Hot Spot and Hot Channel Factors —
Steam-Cooled Fast Breeder Reactor

	$F_{\Delta H_{\text{coolant}}}^*$	$F_{\Delta t_{\text{fuel}}}^\dagger$	$F_{\theta}^\ddagger$
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.

G = mass velocity, lb/hr-ft²

D_e = equivalent hydraulic diameter, ft.

A conservative set of properties was assumed as outlined in NDA 2562-1.

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.

6.3 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 6.2. 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 6.1.

6.3.1 Fuel Rod Diameter

The fuel rod diameter has an important effect on both fuel inventory and fuel cost. In Fig. 6.1 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 increase in fuel lease charge becomes greater than the decrease in fuel fabrication cost (Table 6.3).

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 6.4.) 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

**Table 6.2 — 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

Table 6.3 — 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 6.4 — Effects of Variable Fuel Rod Diameter

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

Clad Rod OD, in.	Fuel Loading, ft ³ Oxide	Leakage $\frac{\text{Volume Coolant}}{\text{Volume Oxide}}$	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.

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. 6.2.

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 the result of reduced leakage from the core with increasing core size. (See Fig. 6.3.)

If the coolant to fuel volume ratio were constant, a similar increase in critical mass with fuel rod 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.

6.3.2 Coolant Inlet Temperature

The lower the reactor inlet temperature the lower the flow rate through the core, and hence the smaller the permissible coolant to fuel volume ratio. The inlet coolant temperature is therefore set a few degrees above saturation.

6.3.3 Coolant Exit Temperature

The selection of the coolant exit temperature has a significant effect on the core size. This effect is shown in Fig. 6.4 using 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 6.3.1.

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.

6.3.4 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.

6.3.5 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 6.5 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 6.3.1 except for the fuel 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.

6.3.6 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. 6.5. 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 6.2.

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.

6.3.7 Steel to Fuel Ratio

It was found that, within a fairly wide range, the amount of stainless steel or nickel structure in the reactor has little effect upon the critical mass and the breeding ratio.

Fig. 6.6 shows a plot of critical mass vs $V_{\text{steel}}/V_{\text{fuel}}$ 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%.

Table 6.5 — Effects of Core Pressure Drop

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

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

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 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. 6.1 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. 6.2 — 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. 6.3 — Effect of fuel rod diameter on breeding ratio. Net plant output 300 MW_e. $\rho_{\text{fuel}} = 10.0 \text{ g/cm}^3$.

Fig. 6.4 — 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. 6.5 — Effect of ligament spacing. 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; maximum clad temperature = 1350°F; maximum fuel temperature = 4500°F; rod diameter = 0.25 in.

Fig. 6.6 — Effect of $V_{\text{steel}}/V_{\text{fuel}}$ on critical mass ($V_{\text{coolant}}/V_{\text{fuel}} = 1.08$)

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

The research and development program for the steam cooled fast breeder reactor is estimated at \$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 7.1.

7.1 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.

Autoclave tests of Inconel and Inconel-X in pure water at 500-680°F have been reported by various facilities.^{1,2,3,4,5} These tests show that corrosion rates are very low (< 1 mil per yr) but that the film produced is very loose in the 600-680°F range water. Inconel also showed no tendency for stress-corrosion cracking and exhibited no galvanic effects with 300-series stainless steels.³ These results were also obtained in a test using hydrogenated, ammoniated, and aerated, high-pressure, high-temperature (600°F), high-velocity water.^{6,7} However, in this test, some pitting was found in the Inconel under stress.⁷

Low corrosion rates were obtained with Inconel in a 1000 hr exposure at 585°F to ammoniated or lithiated steam.⁸ Tests with 500°F steam yielded no pits or cracks and a low corrosion rate (< 1 mil per yr) after 500 hr.¹ Inconel showed little attack in a hot water-steam loop⁹ and did not react appreciably with water vapor at 2000°F.¹⁰

Although no corrosion data of Inconel in 1350-1500°F steam is available, a major supplier felt that Inconel would be satisfactory in steam under oxidizing conditions.¹¹ This was based on low oxidation in air, and the results of superheater tube studies which indicated macroscopically and microscopically

Table 7.1 — 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 develop- ment	2,000,000	▲	▲
6. Steam circulator development	100,000	▲	▲
Total	\$ 12,100,000		

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

that very little attack occurred.¹² 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.

This apparent lack of creep strength was investigated. Some creep and stress-to-rupture properties of Inconel are plotted in Fig. 7.1. These high-temperature properties are similar to those of Type 304 stainless steel.¹³

Before Inconel 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.

7.2 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 have 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 35,000 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.

7.3 CROSS SECTIONS

Theoretical and experimental studies are required to obtain accurate values for the epithermal cross sections of Pu^{239} , higher plutonium isotopes and nickel. These studies need not be charged against this particular reactor concept, but may be supported by general funds for breeder reactors.

7.4 REACTOR SAFETY

A number of studies and experiments are required to ascertain the safety of oxide-fueled breeder reactors. These involve investigations of flooding characteristics, density 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 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 the overall temperature coefficient of reactivity for the oxide breeder is positive mainly because of the positive void coefficient. These computations did not include the negative effects of axial streaming. More refined calculations, 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.

7.5 MECHANICAL EQUIPMENT DEVELOPMENT

The breeder requires relatively frequent reshuffling of fuel elements. To minimize reactor down time it is important to develop means and equipment for performing the refueling operations, with a minimum reactor down time.

7.6 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. 7.1 — Creep and stress-to-rupture vs temperature for Inconel.
(Data obtained from Handbook on Inconel published by International
Nickel Co., Inc., Sept. 1959.)

8. 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 8.1. Its cost is estimated at \$3,000,000; this amount is based on the assumption that a reactor test facility has already been built for the near term development program and will be available for reactor experiments with alternate fuel configurations.

8.1 CLADDING DEVELOPMENT

For the 1975 design, cladding temperatures of 1500°F are proposed for the 300 and 40 MW_e designs. Either Inconel-X or Hastelloy-N promises to be the best material for this service. Hastelloy-N was developed at Oak Ridge as a container material for molten fluoride salts. A comparison of the high-temperature properties of Inconel and Hastelloy-N is shown in Figs. 8.1, 8.2, and 8.3. These figures indicate that Hastelloy-N may well replace Inconel in the advanced versions of the steam-cooled breeder reactor. Additional tests must confirm this, however. Experience at Oak Ridge indicates that Hastelloy-N performed better than Inconel in 1500°F helium, air, and molten salt.¹⁴

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.¹⁵ On a cost per pound basis, this would be \$9 and \$34 for Inconel and Hastelloy-N, respectively.

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.

Table 8.1 — 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 of reactor experiment	2,000,000	▲	▲	
Total	\$ 3,000,000			

*Incremental cost above near term research and development program
presented in Table 7.1.

8.2 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.

8.3 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.

8.4 FLOODING, TEMPERATURE AND POWER COEFFICIENTS

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

Fig. 8.1 — Yield strength vs temperature. (Curves were plotted from data published by International Nickel Co. and Haynes-Stellite Co.)

Fig. 8.2 — Elongation vs temperature. (Data obtained from curves in Chemical Engineering, May 4, 1959.)

Fig. 8.3 — Rupture data — annealed, cold drawn material — at 1500°F.
(Curves were plotted from recent data published by International Nickel
Co., Inc., and Haynes-Stellite Co.)

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