GEAP-4418
AEC RESEARCH AND
DEVELOPMENT REPORT
JANUARY, 1964

LIQUID METAL FAST BREEDER REACTOR DESIGN STUDY

(1000 MWe UO_2-PuO_2 FUELED PLANT)

PROJECT ENGINEER
M.J. McNELLY

VOLUME I

(Sections 1.1 - 2.8)

U.S. ATOMIC ENERGY COMMISSION
CONTRACT AT(04-3)-189
PROJECT AGREEMENT 10

ATOMIC POWER EQUIPMENT DEPARTMENT
GENERAL ELECTRIC
SAN JOSE, CALIFORNIA

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AEC-Research and
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Prepared for the
United States Atomic Energy Commission
Under Contract AT(04-3)189
Project Agreement No. 10

Printed in U.S.A. Price $3.00. Available from
the Office of Technical Services, Department
of Commerce, Washington 25, D.C.

GENERAL ELECTRIC
ATOMIC POWER EQUIPMENT DEPARTMENT
SAN JOSE, CALIFORNIA

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ABSTRACT

A design study has been completed for a 1000-MWe sodium-cooled mixed uranium-plutonium oxide fueled fast reactor with a breeding ratio of 1.24 and sodium outlet temperature of 1100 F. The doubling time is less than 16 years. These values are consistent with the AEC objectives of a breeding ratio of approximately 1.2 and a coolant outlet temperature approaching 1200 F.

A detailed description of a reference 1000-MWe fast breeder reactor plant is presented, together with design specifications and the experimental bases for the design. Estimates of the operating characteristics and economic performance of the reference design are given. The design was chosen after a comprehensive survey of physics and fuel design parameters. Detailed optimization was, however, not considered appropriate at this stage of development. The computational methods employed in the study are detailed and the major uncertainties and sources of error appraised. Preliminary definition of the continuing development program required to permit initiation of design of a small prototype of the 1000-MWe design in 1968 for operation by 1972 is presented. All present evidence shows the reactor system to be technically feasible, although much development work remains to be completed. Further, more detailed design studies backed up by extensive hardware experience are required to define better the economic potential of the plant. A number of alternate design features or basic paramenter choices which offer scope for economic performance well beyond that of the reference design are discussed. The development programs necessary to achieve this improved performance are indicated briefly.

1.0 INTRODUCTION

1.1 GENERAL STATEMENT

1.2 STUDY FORMAT AND RESULTS

1.0 INTRODUCTION

1.1 GENERAL STATEMENT

The concept of a large, ceramic fueled fast breeder reactor has been under study and development by General Electric for the AEC over the past 4 years. In the course of this work, the conceptual design of a 500-MWe mixed plutonium-uranium oxide fueled plant [1, 2]* with fuel costs of under 1 mill/Kwhr, and having desirable safety characteristics, [3, 4] was evolved. Parallel development work indicated the general feasibility of the fuel concept for burnups of the order of 100,000 MWD/T at high specific power, [5, 6] and the resistance of the fuel to severe transient disturbances. [7, 8]

Recent increases in the importance attached to the development of breeder reactors in this country, [9] adoption of the fast ceramic reactor as the prime route to breeders in the United Kingdom and West Germany, and progressive increases in understanding of the technology resulting from research and development, particularly in the physics area, have served to make the present 1000-MWe sodium-cooled study timely.

The purpose of the study has been to do conceptual design work on a mixed plutonium-uranium oxide fueled reactor consistent with the latest research and development information, and to evaluate its potential performance in the 1000-MWe size range. A further purpose and consequence of this study is to obtain a better definition of the required development program to permit design of a prototype in 1968 for power operation in 1972. [10] The schedule for the prototype is consistent with over-all nuclear power program considerations which demand effective utilization of plutonium no later than the late 1970's. Briefly, as fast reactors are more effective users of plutonium than thermal systems, [11] their introduction as economic power producers and subsequent growth should follow approximately the accumulation of by-product plutonium from the present thermal reactor systems. Official estimates [12] indicate that by-product plutonium will become available in the early 1970's at the rate of more than 1 ton/yr, and, as shown by similar analyses, an inventory of 100 tons, worth perhaps $1 billion, will have accumulated shortly after 1980. It is, of course, also possible that the advance of the nuclear power industry will be more spectacular than the official estimates. There is thus a major incentive to develop and demonstrate economic large-scale** plutonium fueled fast reactors before 1980.

The degree of speculativeness which reasonably may be incorporated in the design of a reactor which should be of economic consequence by the late 1970's is, in fact, limited severely. This is exemplified by the fact that its prototype must be committed in 1968, only 4 years hence.

The long time scale of reactor development is already familiar to us from thermal-reactor experience. Large, plutonium fueled fast reactors have, in addition, characteristic development difficulties of their own: the toxicity of plutonium, and therefore the slow pace and unusual

* References are listed at the end of each sub-section.
** The size of the reactor which may be used in the 1980's may vary from 400- to 1000-MWe.

1.1 GENERAL STATEMENT

expense of small-scale experimentation; the quantity of plutonium required for critical experiments and other physics experiments calling for special facilities as well as major fabrication and procurement efforts; and the fast flux needed for fuel development, requiring -- at least until a prototype is available -- the operation and maintenance of uneconomic fast reactors with limited test space and experimental facilities.

All of the foregoing considerations suggest that in this era where public enthusiasm for support of atomic power development is waning, and budgetary pressure on research and development funds in general is high, the ingenuity of reactor designers should be exercised to synthesize worthwhile results with minimal development effort. "Solving" design difficulties by inventing new development problems should be avoided except as a last resort.

Following these precepts, a major effort was made in the present study to arrive at an integrated and self-consistent reference design based on the mixed-oxide fuel concept. The heart of the fast oxide reactor concept is to achieve low fuel costs by high specific power and high burnup without the use of unfamiliar materials or radical operating conditions. The properties of mixed-oxide, stainless steel clad fuel were deduced from the preliminary results already in hand on mixed oxides, supplemented by the considerable experience on the closely-related uranium oxide fuel.

The fuel concept used is the simple closed pin with extended upper fission gas reservoir.

The reactor and primary circuit are in a single tank; visual rather than automatic refueling is provided.

Optimization of fuel cycle costs has not been undertaken, in part because the necessary inputs (such as fuel fabrication cost vs. fuel lifetime) are almost totally lacking, and in part because the results are strongly dependent on the economic ground rules* (such as plutonium prices and inventory charges) and might be inappropriate in other circumstances. Further, it is meaningless to consider fuel cycle costs in isolation; the true goal is lower power costs. Certain directions of improvement of fuel cycle costs (notably increased specific power) may, in fact, adversely affect total power costs by increasing the reactor down time and decreasing the plant load factor. Fast reactor fuel costs are of the order of one-fifth, or less, of the capital costs, and minor changes in the latter are usually more significant than further reduction of fuel cycle costs. Thus, it was found desirable to make a limited extension of the study work to permit over-all capital cost estimates to be drawn up and a realistic appraisal to be made of the incentives towards higher performance (e.g., longer fuel burnup or higher specific power).

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* The AEC ground rules, for example, give little value to reduced doubling time.

1.1 GENERAL STATEMENT (Continued)

Fast reactor cross-section data and calculation techniques have been improved substantially in the past few years. The original study of a 500-MWe fast oxide reactor indicated a softer spectrum than that computed more recently, and this in turn resulted in a change in the computed values for a number of the performance parameters, notably the breeding ratio, Doppler coefficient, and sodium coefficient (see Table 1.1).

TABLE 1.1

EFFECT OF NEW DATA AND METHODS ON 500-MWE REFERENCE DESIGN*

	Old Calculation	New Calculation
Mean Energy of Power Spectrum (Kev)	180	210
Fraction of Fissions Below 9 Kev	0.15	0.13
Total Breeding Ratio	1.08	1.36
Atom Percent Pu (239 + 241)	12.8	11.6
Doppler Coefficient ($T \frac{dk}{dT}$)	- 0.011	- 0.0068
ΔK for Sodium	0	+ 0.0063

* Volume composition, 33 percent fuel, 17 percent steel, 50 percent sodium; h/d = 0.5, 15-inch blankets.

This result stimulated an extensive study (see Section 3.3) of the influence of core composition and leakage on the neutron spectrum and reactor properties. It proved possible to tailor the neutron spectrum by design and by the use or omission of BeO to obtain a range of acceptable compromises between the various physics parameters. In general, harder-spectrum cores have higher breeding ratios, larger (positive) sodium coefficients, and smaller Doppler coefficients; softer-spectrum cores have smaller breeding ratios, smaller sodium coefficients, and larger Doppler coefficients.

In the study, considerable attention was paid to establishing safety criteria for the reactor power coefficients. Briefly (see Section 3.2 for a fuller discussion) the criteria are:

1. The reactor power coefficient must be negative in all regions.
2. The Doppler coefficient must be strongly negative.
3. The sodium void coefficient must be negative over most of the axial blanket.

1.1 GENERAL STATEMENT (Continued)

It is possible to meet these criteria with both hard-spectrum and soft-spectrum reactors. The soft-spectrum reactors, however, meet the criteria with greater margins to cover uncertainties at this stage of development. The physics calculation methods and inputs have an inadequate foundation for the class of reactors under investigation. In particular, the calculated dynamic coefficients (Doppler and sodium coefficients) may be considerably in error. Thus it is prudent to adopt, as a base case, a design with enough margin in meeting the safety criteria to cover errors of as much as 100 percent in the dynamic coefficients. These considerations led to the selection of a softened-spectrum reactor for the "reference design."

The reference design so chosen may understate the performance capabilities of the fast oxide breeder. However, it represents a performance level which can be projected with the greatest degree of certainty and requires the least development to affirm. The extent of the performance improvement possible through advanced fast oxide breeders is also indicated in terms of a series of more advanced designs (see Section 1.2, "Study Format and Results").

In conclusion, the results of this study show that the economic potential of the fast oxide breeder based on materials and concepts already under development is considerable. This is a fortunate circumstance. The soundest policy for the development of more advanced design concepts thus appears to be sequential development reposing on prior exploitation of this economic potential of the near-term reference design. Subsequent development will permit the fast oxide breeder to provide the breeding and doubling properties necessary in the longer term for fuel conservation purposes.

1.2 STUDY FORMAT AND RESULTS

The procedure followed in the 1000-MWe study was detail design and analysis of the reference design case, which was selected after a broad survey as the most well-supported by present evidence. (See Section 3 for further details of the selection process.)

The design specifications are collected in Section 2.1 and described in detail in Sections 2.2 through 2.7. The economic analysis of the reference design is contained in Section 2.8.

Derivation of the design criteria used in Section 2 is detailed in Section 3. The development required to bring the reference design to a point where a prototype may reasonably be committed is given in Section 4.

In Section 5 the performance of more advanced designs is considered, based on progressive relaxations of the design constraints, such as: fuel burnup limit, fuel-clad heat transfer limit, margin for error in dynamic coefficients, fuel specific power, coolant temperatures, etc. In most cases, the advanced performance reactors were quite similar in physical appearance to

1.2 STUDY FORMAT AND RESULTS (Continued)

the reference case, save for minor changes (such as removal of BeO rods). Thus, no additional design work was required to deduce their properties. In one case (the Internal Breeder Reactor) the fuel and hydraulic design were so different from the reference case that a separate preliminary design was made, and is reported in Section 5.3. The additional development work necessary for the advanced designs is discussed in Section 5.4.

Collated in Table 1.2 are the principal results of the four most significant reactor design variants (other variants may be found in Section 5).

Case I is the reference design case. It will be noticed that this design exhibits a breeding ratio of 1.24, thus meeting the study objectives. However, the clad hot spot temperature of 1350 F is at the performance limit for stainless steel, and this limits the sodium outlet temperature to 1100 F.

In Case I refueling takes place twice yearly. Plant availability based on the refueling time alone is expected to be 83 percent. The 70 percent load factor taken in the economic analysis allows for further down time for unspecified maintenance. Preliminary heat balance of a 3500 psi 1000 F/1000 F turbine cycle indicate a net plant output of 1100 MWe for 2500 MWt. Pending further design work on the steam cycle, 1000 MWe was taken as plant output for economic studies.* The capital cost of the plant was estimated at $125/kw.

Capital costs are 2.8 mills/Kwhr, fuel cycle costs under AEC ground rules are 0.57 mills/Kwhr and power costs are 3.71 mills/Kwhr. The breeding ratio is 1.24 and the plutonium doubling time is 15.5 years.

Note that all the design safety criteria could still be met if the measured Doppler coefficient were half that calculated, and simultaneously the measured sodium coefficient were twice that calculated. The Doppler coefficient after loss of sodium in this reactor (thus the one effective during a meltdown accident) is $T \frac{dk}{dT} = -0.008$.

Case II is a reactor very similar to the reference design, but with an unmodified spectrum. The BeO rods are left out. Some small economies in reactor space, coolant circulation, etc., will occur: the beneficial effects on plant capital cost are probably small compared to the precision of the estimates.

* Thus, potentially all the power and fuel cost estimates may be 10 percent too high.

TABLE 1.2

SUMMARY OF RESULTS

Case	I	II	III	IV
Description	Reference Plant	Hard-Spectrum Alternate	Extended Burnup, Hard-Spectrum Alternate	Internal Breeder Concept
Composition				
Sodium Coolant, Volume Percent	47	47	47	35
Fuel (UO_2 + PuO_2), Volume Percent	22	29	29	40
Beryllium Oxide, Volume Percent	7	0	0	0
~~Fuel~~ Fabricated Fissile ~~Fraction~~ Enrichment	0.162	0.13	0.14	0.10
Characteristics				
Mean Energy of Power Spectrum, Kev	120	240	240	240
Breeding Ratio	1.24	1.5	1.4	1.4
Fuel Cycle Specific Power, Kw/kg Fissile	900	1,100	1,100	1,400
Pu Doubling Time, Years	15.5	8	9	5
Economics				
Burnup MWD/T*	100,000	100,000	200,000	200,000
Refueling Interval, Years	0.5	0.5	1.0	5
Load Factor, Percent	70	70	78	95
Fuel Cycle Cost, Mills/Kwhr	0.57	0.29	0.23	(0.07)
Power Cost, Mills/Kwhr	3.71	3.43	3.1	2.3
Dynamics				
Doppler Power Coefficient $\frac{\Delta k}{k} \Big/ \frac{\Delta P}{P}$	-5.6×10^{-3}	-2.8×10^{-3}	-2.8×10^{-3}	-3×10^{-3}
Sodium Power Coefficient $\frac{\Delta k}{k} \frac{\Delta P}{P}$	$+0.02 \times 10^{-3}$	$+0.05 \times 10^{-3}$	$+0.05 \times 10^{-3}$	$+0.15 \times 10^{-3}$
Central Doppler Coefficient $\frac{\Delta k}{k} \frac{\Delta P}{P}$	-7.8×10^{-3}	-3.9×10^{-3}	-3.9×10^{-3}	-4.2×10^{-3}
Central Sodium Coefficient $\frac{\Delta k}{k} \frac{\Delta P}{P}$	$+0.6 \times 10^{-3}$	$+1.4 \times 10^{-3}$	$+1.4 \times 10^{-3}$	$+0.75 \times 10^{-3}$

* MWD/Short ton supplied U + Pu in the fuel

TABLE 1.2 (Continued)

SUMMARY OF RESULTS

Case	I	II	III	IV
Description	Reference Plant	Hard-Spectrum Alternate	Extended Burnup, Hard-Spectrum Alternate	Internal Breeder Concept
Dynamics (Continued)				
Voiding of Upper Blanket Percent of k	-0.4	-0.35	-0.35	-0.2
Doppler Coefficient $T \frac{dk}{dT}$ (Sodium Absent)	-0.008	-0.003	-0.003	-0.005
Development Required				
	Base fuel and physics development (See Table 4.1)	Base fuel, physics development, safety tests	Base fuel, physics development, safety tests, improved fuels, and fission gas venting	Base fuel, physics development, safety tests, improved fuels, fission product gas venting, improved materials, and component performance.

1.2 STUDY FORMAT AND RESULTS (Continued)

Fuel fabrication costs are improved because the extra expense of BeO rods is avoided. The fuel cycle cost is 0.3 mill/Kwhr and power cost is 3.4 mills/Kwhr. The breeding ratio is 1.5, and plutonium doubling time is about 8 years. The safety criteria are met with good margin provided present estimates of the dynamic coefficients are correct. The Doppler coefficient after loss of sodium is $T \frac{dk}{dT} = 0.003$.

Case III is a synthesis of two cases developed in Section 5. It is the same as Case II with doubled fuel burnup (200,000 MWD/U.S. ton). It is probable that some means of venting to coolant will be required to achieve this burnup. Although the breeding ratio is down to 1.4 and the doubling time up to about 9 years, the fuel cycle cost is improved to 0.2 mill/Kwhr because of lower fabrication and reprocessing costs.

The plant load factor is increased because refueling is annual and not semi-annual. As a result, capital costs are improved and power costs are down to 3.1 mills/Kwhr.

The safety parameters do not differ appreciably from Case II.

Cases I, II, and III share a common mechanical design and geometrical aspect. Features of interest include the following:

The core contains nearly 19 volume percent steel. It is composed of 225 hexagonal fuel channels (5 inches on a side, 0.15-inch wall thickness), each enclosing 608 fuel pins. Each fuel channel is mounted independently on the core support structure. The pins are ruggedly clad (0.25-inch OD and 0.015-inch wall thickness), and are supported along their entire length by a slotted tube which gives continuous spring support. The pin support permits axial and radial thermal expansion within the channel. An extensive fission gas reservoir (150 percent of fuel volume) at the upper end of the fuel pins accommodates fission product gases at low pressure beyond the top blanket. The fuel pin and support design minimizes the possibility of local disruption of coolant flow in the channels by fuel swelling or distortion, or by rupture and rapid gas release. The fuel channel design prevents mechanical or thermal propagation across the core of occurrences within any of its constituent fuel assemblies.

Case IV is a radically-different design, with a nonpancaked core, a low (35 percent) sodium coolant volume, and internally-cooled fuel elements. The core breeding ratio is sufficiently high that the reactivity changes during extended burnup are small. Thus, the plant could be operated with infrequent refueling -- perhaps as little as once in a core lifetime (5 years). To take full advantage of this possibility, plant down time for equipment maintenance must also be

1.2 STUDY FORMAT AND RESULTS (Continued)

drastically reduced. Assuming this can be done, load factors as high as 95 percent may be attainable with corresponding benefit to the capital cost contribution to power costs. *

The relatively few, large fuel elements would be fabricated by vibratory compaction. The fuel fabricating costs would be low and the fuel cycle shows a net credit.

Power costs are down to 2.3 mills/Kwhr, assuming the $125/Kw capital costs estimated for the reference design is approximately correct for this design as well.

Fission gas venting and improvements in fuel-clad heat transfer and cladding strength are required. The safety criteria are met adequately if present estimates of the dynamic coefficients are correct. (For further details, see Section 5.3.)

* Reducing equipment maintenance down time would also improve the performance of Cases I-III, so part of the benefit shown is available without requiring a new design.

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2.0 DESCRIPTION OF REFERENCE PLANT

2.1 SUMMARY DESCRIPTION
2.2 REACTOR PLANT
2.3 REACTOR SYSTEMS
2.4 STEAM GENERATION AND POWER PLANT
2.5 REACTOR PHYSICS DATA
2.6 SYSTEM DYNAMICS
2.7 PLANT STAFF AND REFUELING OPERATIONS
2.8 ECONOMICS DATA

2.1 SUMMARY DESCRIPTION

2.1 Summary Description

The reference 1000-MWe plant consists of a single sodium cooled reactor unit of 2500-MWt rating. Sodium leaving the reactor core at a mean exit temperature of 1100 F transfers heat to nonradioactive sodium in six secondary coolant loops operating in parallel. Steam, at 3500 psia 1000 F, is raised in six once-through steam generators and flows to a cross compound, four-flow 3600, 1800 rpm, 1000 F reheat turbine. The reactor is fueled with stainless steel clad mixed uranium, plutonium oxide in the form of fuel pins of 0.25 inch OD and 2-foot active length. To provide improved nuclear safety characteristics, BeO pins of the same geometry as the fuel pins and also clad in stainless steel are interspersed with the fuel in the core. The fuel pins and BeO pins are arranged on a close-packed triangular pitch within hexagonal channels to form a fuel assembly, 8.75 inches across flats. 225 fuel assemblies constitute the reference pancake-shaped reactor core which has a fueled region approximately 12 feet in diameter and 2 feet high.

All fuel pins contain depleted uranium above and below the active fuel region to form the axial blankets. A radial blanket consisting of stainless steel clad rods of depleted uranium 0.5 inch in diameter surrounds the core. The reactor control is accomplished using 85 top-entry boron carbide control rods. The reactor arrangement is based on the submerged system concept, all components of the primary sodium coolant circuit being contained within a 52 foot diameter ~40 feet deep sodium-filled tank. The reactor core and blanket are submerged under 20 feet of sodium within a reactor vessel mounted in the center of the primary system tank. The six vertical, shaft-sealed centrifugal pumps and associated intermediate heat exchangers and purification equipment are located, partially submerged, around the reactor vessel (see Figure 2.1).

Manual refueling with direct observation of all operations is provided for using a shielded, inert-gas filled cell and manipulators mounted directly over the reactor vessel. Vertical removal of the vessel head, control drives, and associated shielding, together with adjustment of the sodium coolant level, permits direct viewing and access to the upper end of fuel assemblies within the core. Subsequent sodium level adjustments provide for a transfer canal thus permitting all fuel to be submerged continuously during removal from the core.

The entire reactor, its sodium coolant system, steam raising plant, and refueling facilities are housed within a leak-tight building normally filled below operating floor level with an inert gas.

Reactor operations are estimated to require a permanent staff of 112 and, taking into account the two required refueling shut downs per year, an over-all plant availability of approximately 80 percent is estimated, thus leading to an assumed average load factor in a utility system operation of 70 percent.

The estimated procured cost of the reactor equipment is $7.92 million. The over-all plant cost, including customer expenses, is estimated to be approximately 125 dollars/kw installed.

2.1 Summary Descr Continued)

Fuel cycle costs based on government ownership and reprocessing are estimated at 0.569 mills/kw-hr, and thus an over-all power cost for a unit of this type constructed for operation in the 1980's by a privately owned utility (14 percent capital charges) is approximately 3.7 mills/kw-hr. The computed breeding ratio of the reference plant, at a specific power of 1000 kw/kg of fissile material in core and 900 kw/kg of fissile material within the fuel cycle, is 1.24 . This is estimated to yield a fissile material doubling time of about 15.5 years.

Future improvements, as described in Section 5, are estimated to result in an increase in the breeding ratio to possibly 1.4, thus decreasing the doubling time to less than 10 years (i.e., less than that experienced for growth of the electrical industry), thus enabling the future cost of power from such fast reactors to be largely insensitive to the cost of uranium supply.

Table 2.1 lists important design parameters for the reference plant; these parameter values are not necessarily optimum values in view of the futility of seeking same at the present stage of development of this class of reactor.

TABLE 2.1

DESIGN DATA FOR REFERENCE REACTOR*

1. Key Design Values

Reactor Power Electrical (Net)	MWe	1100**
Thermal (Total)	MWt	2500
Core	MWt	~~2200~~ 2117
Blanket axial	MWt	~~250~~ 291
radial	MWt	~~50~~ 92
Fuel Inventory in Core, 16.2 percent fissile	kgs UO_2 + PuO_2	14,550
Average Core Power Density	Kw/liter	~~365~~ 350
Specific Power for Fuel Cycle	Kw/kg fissile Pu	~ 900
Blanket inventory	kgs UO_2	~~40,000~~ 49,600
Breeding Ratio	-	~~1.257~~ 1.24
Doubling Time for Fuel Cycle	yrs	~~13.5~~ 15.5
Fuel rating (maximum)	Kw/ft	22.7
Peaking factor (maximum)		2.18
Coolant Inlet Temperature	°F	800
Coolant Outlet Temperature	°F	1100

*All values are for hot operating condition unless stated otherwise.

**Results from an assumed thermal rating of 2500 MW and the use of a supercritical steam cycle yielding approximately a 44 percent net station efficiency.

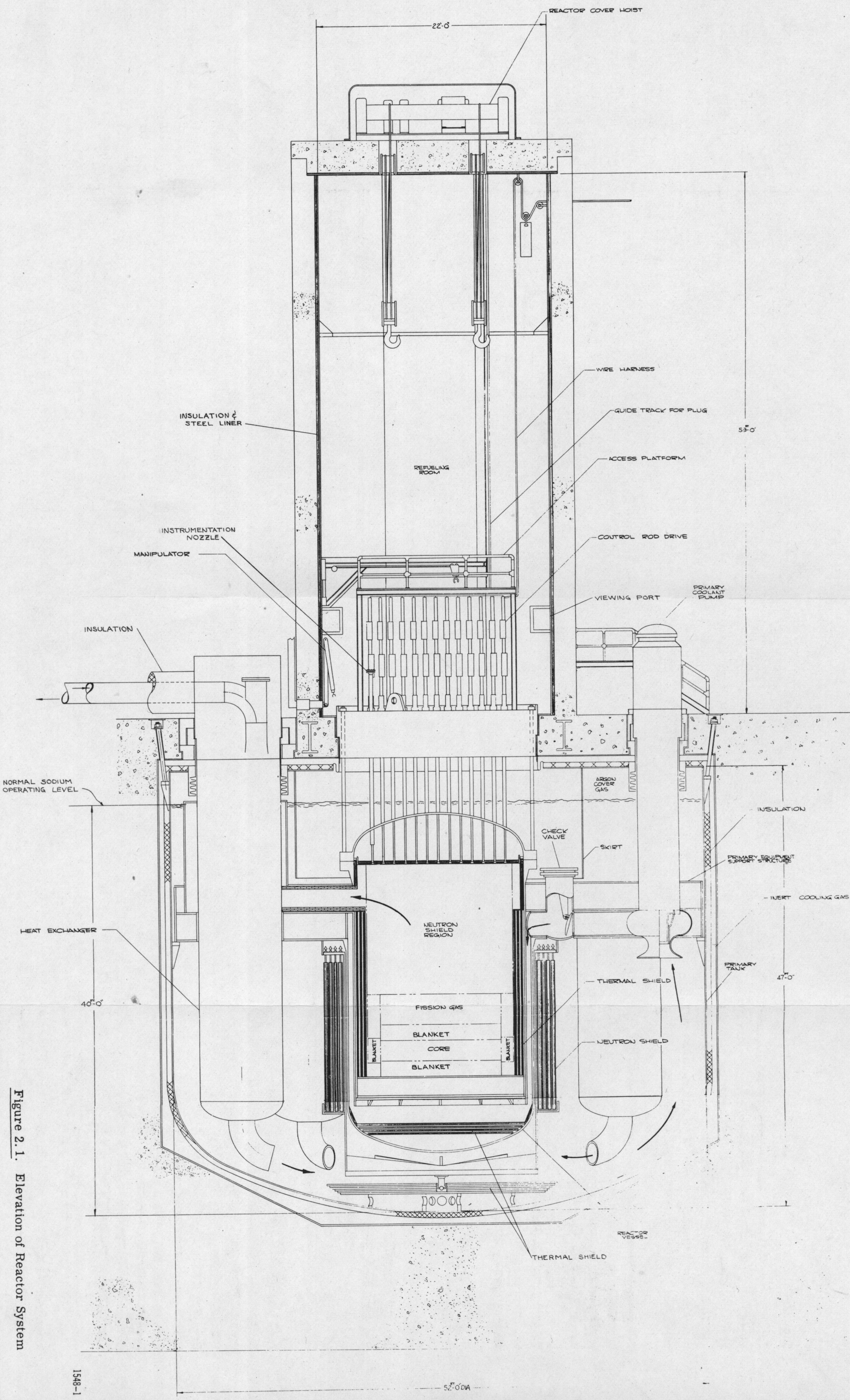

Figure 2.1. Elevation of Reactor System

1548-1

TABLE 2.1 (Continued)

DESIGN DATA FOR REFERENCE REACTOR

1. Key Design Values (Continued)		
Coolant Flow	lb/hr	95.4×10^6
Core Inlet Pressure (Neglects 10 psi Static Head)	psia	50
Core Outlet Pressure (Neglects 7 psi Static Head)	psia	25
Maximum Coolant Velocity in Core	ft/sec	13.6
Active Core Height	ft	2
Active Core Diameter	ft	11.65
Diameter of Reactor Vessel	ft	18
Height of Reactor Vessel	ft	33
Diameter of Primary Sodium Tank	ft	52
Height of Primary Sodium Tank	ft	47
Number of Primary Coolant Circuits		6
Secondary Loops		6
Reheater Loops		2
Steam Pressure (Turbine Stop Valve)	psia	3500
Temperature (Turbine Stop Valve)	°F	1000
Temperature Reheat	°F	1000
Flow	lbs/hr	7.64×10^6
Turbine Heat Rate	Btu/Kwhr	7570
2. Fuel Pins		
Dimensions — Fuel Clad OD	in.	0.25
Clad Thickness	mils	15
t/d		0.06
Fuel Material - Pellets - Smeared Density of UO_2	gm/cc	9.2
- Maximum operational center void diameter	in.	~~0.058~~ 0.06
Fuel Rating Maximum Linear Power Generation	Kw/ft	22.7
Average Linear Power Generation	Kw/ft	10.4
Mean Specific Power	Kw/kg	150
Surface Heat Flux Maximum Hot Spot on Fuel Pellet Surface	$Btu/ft^2hr \times 10^6$	1.3
Fuel Surface Temperature Point of Maximum Rating	°F	2165
Fuel $\int_{T_o}^{T_c} Kd\theta$ Assumed Rod Fuel Limit	watts/cm	90
~~$\int_{T_s}^{T_c} Kd\theta$ Maximum Required Rod Performance~~	~~watts/cm~~	~~59~~

TABLE 2.1 (Continued)

DESIGN DATA FOR REFERENCE REACTOR

2. Fuel Pins (Continued)		
Maximum Fuel Temperature T_c Steady State	°F	4700
Dimensions - Fuel Length	in.	24
- Blanket Length	in.	2 × 18
- Central Void - Based on Avoiding Melting	vol %	8
- Central Void Diameter Maximum	in.	0.063
- Over-all Clad Length	ft	8
Weight Fuel in Pin (UO_2 + PuO_2	gms	138
Blanket UO_2	gms	206
Gross of Pin (UO_2 + PuO_2)	gms	344
Maximum Fission Product Pressure	psia	1000
Stress Maximum from Fission Product Pressure	psi	8620
Maximum Thermal	psi	21,000
Assumed Yield of Typical 300 Series Stainless Steel Clad at 1200 F	psi	22,000
3. BeO Pins		
BeO Pin OD	in.	0.25
Clad Wall Thickness	mils	15
BeO Pellet Smeared Density	%	~~94~~ 90
BeO Column Length	ft	5
BeO Mass/pin	gm	104
4. Control Rods		
Control Rod Pitch	in.	15.4
Guide Tube-OD	in.	2.34
-ID	in.	2.10
Control Rod-OD	in.	2.00
-Clad Thickness	in.	0.050
Control Material - B_4C -Smeared Density	%	75
-Smeared Density	gm/cc	1.87
B^{10} -Enrichment		Natural
Active B_4C Length	ft	2
Normal Control Travel	ft	3.5

TABLE 2.1 (Continued)

DESIGN DATA FOR REFERENCE REACTOR

5. Fuel Assembly

Number -	BeO Pins		138
-	Fuel Pins		470
Dimensions -	Fuel Pin Spacing Triangular Pitch	in.	0.338
-	Fuel Channel Across Hex Flats	in.	~~8.875~~ 8.75
-	Fuel Channel Across Corner	in.	10.3
-	Fuel Channel Wall Thickness	in.	0.150
-	Spacer Tube OD	in.	0.140
-	Spacer Tube Wall Thickness	in.	0.005
-	Length Total	ft	23
	Lower End Piece	in.	36
	Lower Axial Blanket	in.	18
	Active Fuel Section	in.	24
	Upper Blanket	in.	18
	Fission Gas Reservoir	in.	36
	Fuel Extension and Shield	ft	12
-	Mass Blanket Per Assembly UO_2	kg	98
	Fuel Per Assembly UO_2 - PuO_2	kg	65
	BeO Per Assembly	kg	14
	Total Assembly with Extensions	kg	1200

6. Reactor Assembly

Number of Core Fuel Assemblies			225
Number of Radial Blanket Assemblies			108
Number of Control Rods			85
Number of Composition Zones			3
Fuel Assembly Spacing, Center-to-Center		in.	ok. ~~8.875~~
Control Element Spacing		in.	15.4
Dimensions:	Core Height	ft	2
	Core Equivalent Diameter	ft	11.65
	Core Volume	ft^3	213
	Axial Blanket Thickness (Upper and Lower)	ft	1.5
	Axial Blanket Volume	ft^3	319
	Radial Blanket Height	ft	~~[illegible]~~ 3
	Radial Blanket Thickness	ft	1.26
	Radial Blanket Volume	ft^3	148
	Core and Blanket Over-all Equivalent Diameter	ft	14.17

TABLE 2.1 (Continued)

DESIGN DATA FOR REFERENCE REACTOR

6. Reactor Assembly (Continued)

Core and Axial Blanket Compositions:		
Coolant (Control Inserted)	%	45.9
Fuel at 11.1 gm/cc	%	22.0
Beryllium Oxide at 3.1 gm/cc	%	7.1
Control Material Boron Carbide at 2.5 gm/cc	%	1.1
Structure (Stainless Steel)	%	18.8
Void Associated with Fuel, Beryllia, and Control	%	5.1

7. Active Core

Area - Total Core Cross-Sectional	ft^2	107
- Total Cell - Including One-Third Control Rod	$in.^2$	68
Total Coolant Cross Section	ft^2	50
In-Channel Coolant Cross Section	ft^2	45.3
Dimensions - Total Length Active Fuel in Core	$ft \times 10^3$	211
Composition - Coolant in Channels	%	42.4
- Coolant Outside Channels (Control Removed)	%	4.0
- Stainless Steel Clad	%	9.9
- Other Stainless Steel Structure	%	8.9

8. Thermal Design

Axial Peaking Factor		1.14
Radial Peaking Factor		1.24
Maximum Local Peaking Factor		1.68
Total Factor		2.18
In-Channel Hydraulic Diameter	ft	~~0.0214~~ 0.01
Maximum Velocity	ft/sec	13.6
Maximum Reynolds Number		~~870,000~~ 50,000
Maximum Heat Transfer Coefficient	$Btu/ft^2hr°F$	23,000
Maximum Clad Surface Heat Flux	Btu/ft^2hr	1.14×10^6

9. Radial Blanket

Clad OD	in.	0.5
Clad Thickness	mils	20
Active length of pins	in.	30
Uranium Oxide Diameter	in.	0.45
Blanket Pin Spacer - Triangular Pitch	in.	0.565

TABLE 2.1 (Continued)

DESIGN DATA FOR REFERENCE REACTOR

9. Radial Blanket (Continued)		
Number of Pins per Assembly		208
Number of Assemblies		108
Composition - Coolant Content	%	32.1
- Uranium Oxide Content at 10.9 gm/cc	%	43.9
- Stainless Steel Content	%	17.2
- Void Content	%	6.8
Total Weight of UO_2 in Radial Blankets	kgs × 10^3	~~20.25~~ 17.5
10. Physics - Core		
Fuel Composition as Supplied-Inner Zone (Z1) 239	%	14.11
240		6.95
241		1.56
242		0.84
238		76.54
Fission Product Pairs		0
-Mid Zone (Z2) 239		16.21
240		7.81
241		1.75
242		0.94
238		73.29
Fission Product Pairs		0
-Outer Zone (Z3) 239		18.27
240		8.77
241		1.97
242		1.06
238		69.93
Fission Product Pairs		0
Fuel Composition as Discharged		
-Inner Zone (Z1) 239	%	11.05
240		6.75
241		1.55
242		0.84
238		68.41
Fission Product Pairs		11.40

TABLE 2.1 (Continued)

DESIGN DATA FOR REFERENCE REACTOR

10. Physics - Core (Continued)

Fuel Composition as Discharged			
-Mid Zone (Z2)	239	%	12.05
	240		7.54
	241		1.74
	242		0.94
	238		66.23
Fission Product Pairs			11.50
-Outer Zone (Z3)	239		13.30
	240		8.43
	241		1.96
	242		1.06
	238		63.95
Fission Product Pairs			11.30

11. Physics - Blankets

Axial - As Supplied	235	%	0.3
- As Discharged	239		3.91
	240		0.32
	241		0.02
	242		~0
	235		0.18
	236		0.04
	238		94.46
Fission Product Pairs			1.07
Radial - As Supplied	235		0.3
- As Discharged (4.6 year residence)	239		2.44
	240		0.09
	241		~0
	242		~0
	235		0.23
	236		0.02
	238		96.64
Fission Product Pairs			0.58

TABLE 2.1 (Continued)

DESIGN DATA FOR REFERENCE REACTOR

12. Physics - Dynamics Data

	Over-all Temperature Coefficient	$\frac{\Delta k}{k\,^\circ F} \times 10^6$	~~-7.2~~ -6.7
	Over-all Power Coefficient	$\frac{\Delta k}{k}$ per $\frac{\Delta P}{P}$	-0.0073
Doppler -	Parameter A_{Dopp}	$T \frac{dk_{Dopp}}{\cancel{dT} \, dT}$	-0.01
-	Temperature Coefficient-Hot Operating (1500 K) Refueled Core	$\frac{\Delta k}{k\,^\circ F} \times 10^6$	-3.5
-	Power Coefficient-Hot Operating (1500 K) Refueled Core	$\frac{\Delta k}{k} \frac{\Delta P}{P} \times 10^3$	-5.6
Sodium -	Temperature Coefficient-Hot Operating (1500 K) Refueled Core	$\frac{\Delta k}{k\,^\circ F} \times 10^6$	+0.15
-	Power Coefficient-Hot Operating (1500 K) Refueled Core	$\frac{\Delta k}{k} \frac{\Delta P}{P} \times 10^3$	+0.023
Minimum Local Ratio of Doppler/Sodium Power Coefficients		$\frac{\Delta k_{Dopp}}{\Delta k_{Na}}$	12
Loss of Sodium -	Maximum Partial ~~Worth~~ Loss	$	~~$~~ +4.2
-	Total Core ~~Worth~~ Loss	$	~~-0.1~~ +2.4
-	~~Upper~~ Axial Blanket ~~Worth~~ Loss	$	~~[illegible]~~ -2.4
-	~~Maximum or One Complete Change~~ Complete Loss, Central Channel	$	+0.04
-	~~Worth for Center of Worst Channel~~ Partial Loss, Central Channel	$	+0.05
Fuel Slump ~~Maximum Worth 100 Percent Density~~ From 88 to 100 Percent Density		$	~~[illegible]~~ +4.5
Doppler Available to Point of Fuel Damage			
(6500 F) -	From Operating Condition	$	1.5
-	From Shutdown Condition	$	5.4

13. Coolant - Primary

Temperature	- Inlet	°F	800
	- Mixed Mean Outlet	°F	1100
	- Maximum Channel Outlet	°F	1300
	- Maximum Local (Surface Hot Spot)	°F	1332
Flow	- Total Through Reactor	lb/hr	9.54×10^7
	- Total In Core	lb/hr	9.15 ~~8.22~~ $\times 10^7$

TABLE 2.1 (Continued)

DESIGN DATA FOR REFERENCE REACTOR

13. Coolant - Primary (Continued)

~~Radial~~	- Blanket Flow	lb/hr	0.4 ~~1.23~~ $\times 10^7$
	- Leakage Flow	lb/hr	0.08
	- Control Rod Flow	lb/hr	0.08
	- Maximum Single Bundle	lb/hr	5.05 ~~4.8~~ $\times 10^5$
Velocities - Maximum In Channel			
	(ρ Sodium = 50 lb/ft^3 at 1100 F)	ft/sec	13.6
	- Mean In Channel	ft/sec	11.1
	- Past Control Rod	ft/sec	2.5
	- Mean In-Radial Blanket	ft/sec	1
Pressure Loss-Hottest Channel			
	- Total Core	psi	25
	- Inlet Nozzle	psi	$\sim$0.8
	- Lower Blanket	psi	4.5
	- Core	psi	6
	- Upper Blanket	psi	4.5
	- Fission Gas Reservoir	psi	9
	- Upper Extension and Nozzle	psi	$\sim$0.2
	- Primary Circuit ΔP	psi	40
Pressure	- Inlet Plenum*	psia	50
	- Outlet of Core**	psia	25
Pump Discharge (Neglecting Static Head)		psia	55
Cover Gas		psia	15
Coolant Mean Circuit Time- Pump Inlet to Intermediate Heat Exchanger Outlet		sec	10
	- Pump Inlet to Pump Inlet	sec	120
Total Coolant Circuit Inventory within Primary Tank		lbs	3×10^6
Coolant Composition - Sodium			Reactor Grade
	- Na_2O	ppm	5
	- Carbon	ppm	<10
Maximum Surface Heat Flux		Btu/ft^{2}hr $\times 10^6$	1.2
Minimum Subcooling - At Intermediate Heat Exchanger		°F	$\sim$500
	- At Core Inlet	°F	$\sim$1150

*Neglects 10 psi of static sodium head

**Neglects 7 psi of static sodium head

TABLE 2.1 (Continued)

DESIGN DATA FOR REFERENCE REACTOR

14. Heat Removal Equipment

Number Circuits		6
Total Surface Area Intermediate Heat Exchanger	ft^2 ~~$\times 10^3$~~	91,000
Mean ΔT - Intermediate Heat Exchanger	°F	63.8
Pumping Power- Primary Circuit	MW	6.8
- Intermediate Circuit	MW	4.2
Total Surface Area - Steam Generator	ft^2 ~~$\times 10^3$~~	88,800
- Reheat	ft^2 ~~$\times 10^3$~~	~~56~~ 36,000
Steam - Pressure (Turbine Stop Valve)	psia	3500
- Temperature	°F	1000
- Reheat	°F	1000
- Feed temperature	°F	500
Mean ΔT's - Steam Generator	°F	~~150~~ 130
- Reheater	°F	115
Heat Loads - Intermediate Heat Exchanger	Btu/hr $\times 10^3$	86
- Boiler	Btu/hr $\times 10^3$	71.8
- Reheater	Btu/hr $\times 10^3$	14.2

15. Equipment Specifications

Radial Neutron Shield Thickness	ft	2
Diameter of Reactor Vessel	ft	18
Height of Reactor Vessel	ft	33
Design Pressure for Reactor Vessel	psi	100
Number of Coolant Circuits		6
Diameter of Sodium Tank	ft	52
Height of Sodium Tank	ft	44
Design Pressure of Sodium Tank	psi	1
Inventory of Sodium	lbs	3×10^6
- Level Hot	ft	40
- Level Cold	ft	37
Thickness of Tank Walls	in	2
Thickness of Tank Head	in	2
Thickness of Top Shield	ft	6
Cooling for Top Shield		None

TABLE 2.1 (Continued)

DESIGN DATA FOR REFERENCE REACTOR

15. Equipment Specifications (Continued)

Height of Refueling Cell	ft	50
Diameter of Refueling Cell	ft	23
Shield Thickness of Refueling Cell	ft	3
Distance of Sodium Level Below Operating Floor	ft	10
Distance of Core Top Below Operating Floor	ft	33
Maximum Lift Capability of Refueling Manipulator	tons	2

16. Power Plant

Turbine Heat Rate	Btu/Kwhr	7570
Steam Flow	lb/hr 10^6	7.64
Condenser Pressure	in. hg	1.5
Heat Rejection	Btu/hr $\times 10^9$	5.0
Auxiliary Power Required (Excludes Boiler Feed Pumps)	MW	6
Boiler Feed Pumps (Turbine Driver)	MW	32
Plant Net Efficiency	%	44

17. Fuel Cycle*

		Core and Axial Blanket	Radial Blanket
Average In-Core Residence	months	29	59
Total Cooling Time at Plant	months	4	4
- Shipping Time	months	1	1
- Reprocessing Time	months	1	1
- Refabrication Time	months	2	2
- Storage Time	months	1	1
- Fueling Time	months	1	1
Mean Ratio In-Core to ~~Out-of-Core~~ Total Inventory Time (not dollar value)		0.77	0.87

Basis

Fabrication Cost	Fabrication Plant in GEAP-3824
Shipping Cost	50 cent/ton-freight-mile; for 750 miles
Reprocessing Cost	AEC Reprocess Plant, $17,000 per day
Depletion Charges	10.00 dollars/gm Pu-239 and 241 as nitrate
Inventory Charges	Leased fuel at 4.75 percent per year
Losses	1 percent in Fabrication and Reprocessing
Fabrication Working Capital	6 percent per year on Book Value

*See section 2.8 for further details. These fuel cycle costs do not incorporate some minor revisions later made in physics data and blanket design. These revisions would have been negligible effects on fuel cost.

TABLE 2.1 (Continued)

DESIGN DATA FOR REFERENCE REACTOR

17. Fuel Cycle (Continued)

Core	$/kg		Mills/eKwhr	
Fabrication Cost (including Fabrication losses)	298		0.248	
Shipping Costs	41		0.034	
Reprocessing Cost	30		0.025	
Depletion Charges Net	386		0.322	
Inventory Charges	229		0.191	
Losses (during reprocessing)	13		0.011	
Fabrication Working Capital	24		0.019	
Subtotal		1021		0.850
Axial Blanket	**$/kg**		**Mills/eKwhr**	
Fabrication Cost	41		0.052	
Shipping Cost	41		0.052	
Reprocessing Cost	30		0.038	
Depletion Charges Net	-344		-0.430	
Inventory Charges	0		0	
Losses	3		0.004	
Fabrication Working Capital	3		0.004	
Subtotal		-226		-0.280
Radial Blanket	**$/kg**		**Mills/eKwhr**	
Fabrication Cost	75		0.044	
Shipping Cost	20		0.012	
Reprocessing Cost	30		0.017	
Depletion Cost	-138		-0.080	
Inventory Charge	0		0	
Losses	2		0.001	
Fabrication Working Capital	11		0.006	
Subtotal		0		0
Total				0.570

2.2 REACTOR PLANT

2.2.1 CORE CONFIGURATION

2.2.1.1 Fueling Pattern

2.2.1.2 Mechanical Arrangement

2.2.2 FUEL AND BLANKET ASSEMBLIES

2.2.2.1 Fuel Spacer

2.2.2.2 Inlet Nozzles

2.2.2.3 Shield Section

2.2.2.4 Radial Blanket Assembly

2.2.3 FUEL AND BeO PIN CONSTRUCTION

2.2.3.1 Materials Specifications

2.2.3.2 Fabrication

2.2.4 THERMAL DESIGN

2.2.4.1 Heat Removal

2.2.4.2 Fuel Temperatures

2.2.4.3 Clad Stresses

2.2.5 CONTROL RODS

2.2.5.1 Poison Section

2.2.5.2 Connecting Rod and Guide Tube

2.2.5.3 Drive Mechanism

2.2.6 CORE INSTRUMENTATION

2.2.6.1 Thermocouples

2.2.6.2 Ion Chambers

2.2 REACTOR PLANT

The reference reactor design is a pancake-shaped core of mixed uranium and plutonium oxides containing a small amount of beryllium oxide to adjust the neutron spectrum, thus yield the desired combination of economic performance and adequate safety characteristics. Details of the studies of design alternates and of the parameter surveys which form the basis for the reference design are given in Sections 3.1 - 3.5.

A core structural design has been evolved which gives continuous support to the fuel pins on three sides and along their entire length. The reference fuel design allows for gross fission gas release from the fuel material to an end-gas reservoir. Both of these factors are of major importance in view of the proposed fuel burnup of 100,000 MWD/T and the unavoidable high levels of combined thermal and pressure stresses in the clad (see Section 3.4.4).

The mixed mean exit coolant temperature is 1100 F for the reference design. It is the highest temperature considered practical, using series 300 stainless steel structural materials, if an economically-attractive power rating (1000 Kw/kg fissile) is to be attained in the core. The peak surface temperature for the clad is computed to be 1332 F using a conservative compilation of some nine separate local peaking factors. The coolant temperature rise across the core is from 800 to 1100 F and is limited both by peak temperature considerations and from analysis of transient performance (see Section 3.2).

2.2 REACTOR PLANT (Continued)

2.2.1 CORE CONFIGURATION

The reactor assembly of stainless steel clad mixed UO_2 PuO_2 fuel and BeO pins is surrounded by a depleted UO_2 blanket and the entire structure is cooled by upward-flowing sodium. The core has a height of 2.0 feet and an equivalent diameter of 11.65 feet. The upper and lower axial blankets have a thickness of 18 inches and a diameter equal to that of the core. The radial blanket has an equivalent thickness of 15.2 inches and a height of 36 inches. The over-all equivalent diameter of the core and blanket is 14.17 feet. The circumscribed diameter is 14.83 feet.

2.2.1.1 Fueling Pattern

The core is made up of 225 fuel assemblies, hexagonal in cross section. Both the upper and lower axial blanket regions are fabricated as an integral part of the fuel assemblies. The radial blanket consists of 108 assemblies also of hexagonal cross section. Refueling is on a multi-batch scatter reload pattern within the various core and radial blanket zones.

The core is divided into three zones of different plutonium concentrations for the purpose of controlling the power distribution (see Section 2.5 for further data). The arrangements of the core zones and the blankets are shown isometrically in Figure 2.2.1.1. Zones 1, 2, and 3 contain 75, 72, and 78 assemblies, respectively.

2.2.1.2 Mechanical Arrangement

The 225 core assemblies and associated 108 radial blanket assemblies are mounted on a lower grid structure which is approximately 17 feet in diameter. The core support grid houses the inlet nozzles of the fuel assemblies: it is 24 inches deep and thus can provide the necessary lateral stability for the assemblies and the control rod guide tubes it supports. The core support grid hangs from a core shroud, which separates inlet from exit coolant flow streams. The shroud is attached to the upper portion of the reactor vessel (see Section 2.3.2 for further details), thus the whole structure expands in a downward direction when raised in temperature.

The fuel assemblies in the core are arranged in a pattern of modules, each consisting of three assemblies plus one control rod, as illustrated on Figure 2.2.1.2.(a). A cross section of the module, Figure 2.2.1.2.(b), shows the design of the hexagonal fuel assemblies which each have one recessed corner oriented to form the channels for the control rod guide tubes.

The hexagonal fuel assemblies and radial blanket assemblies are spaced on a triangular pitch of 8.875 inches. This provides a nominal 1/8-inch gap between assemblies to facilitate handling and accommodate manufacturing tolerances. The spacing is maintained by the lower support grid. Relative lateral movement between fuel assemblies is minimized by:

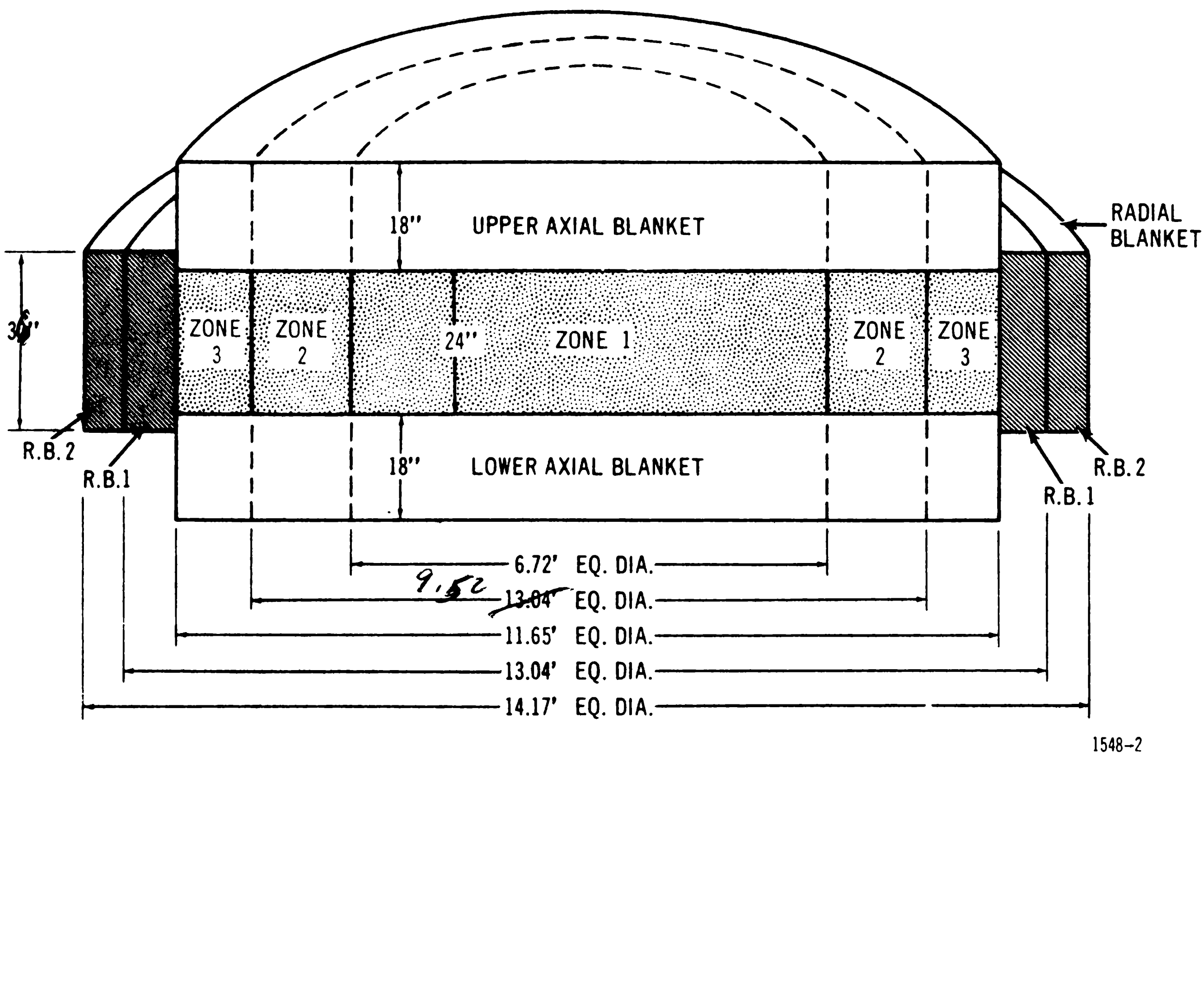

Figure 2.2.1.1. Core and Blanket Arrangement

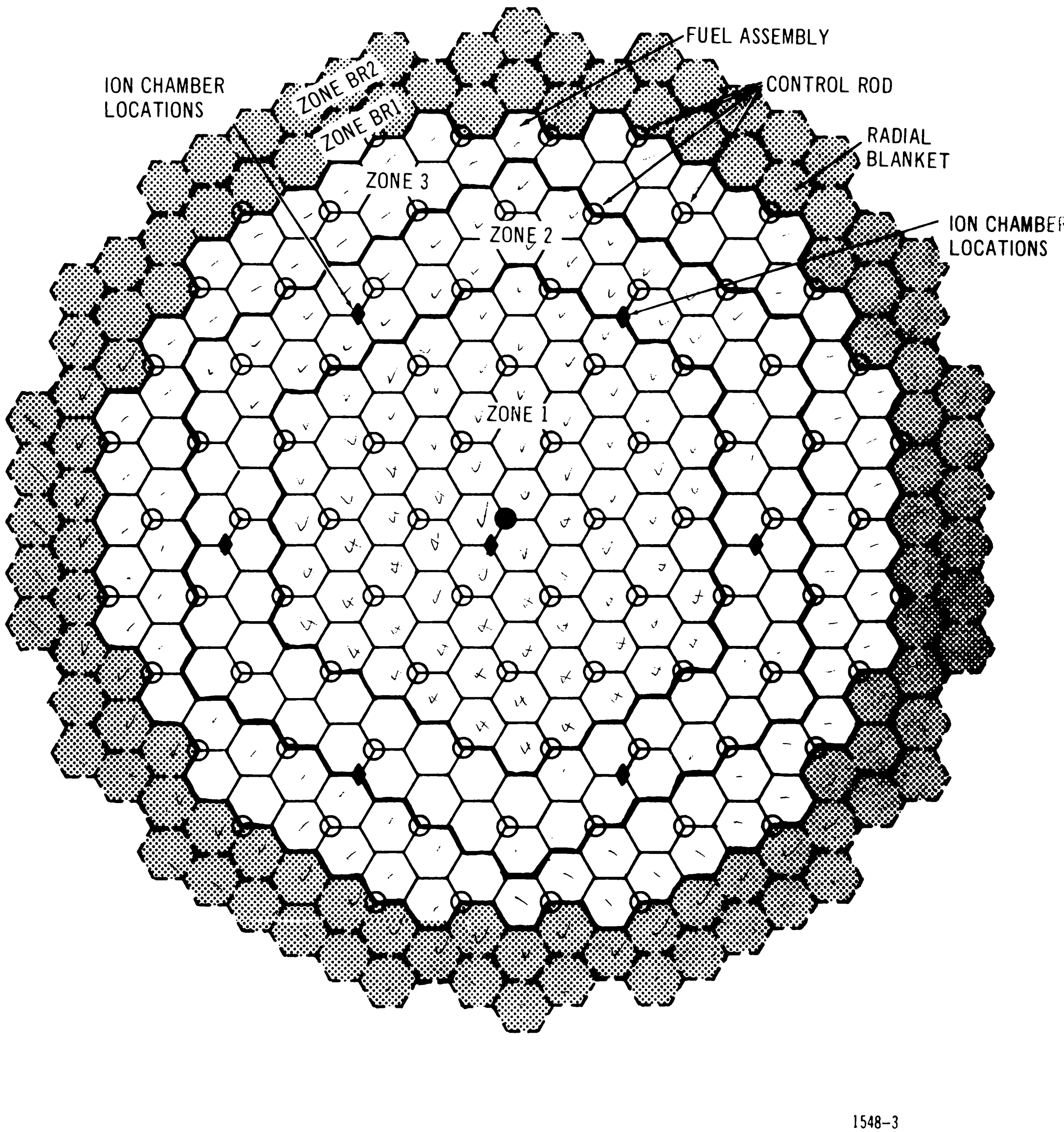

Figure 2.2.1.2(a). Core Layout

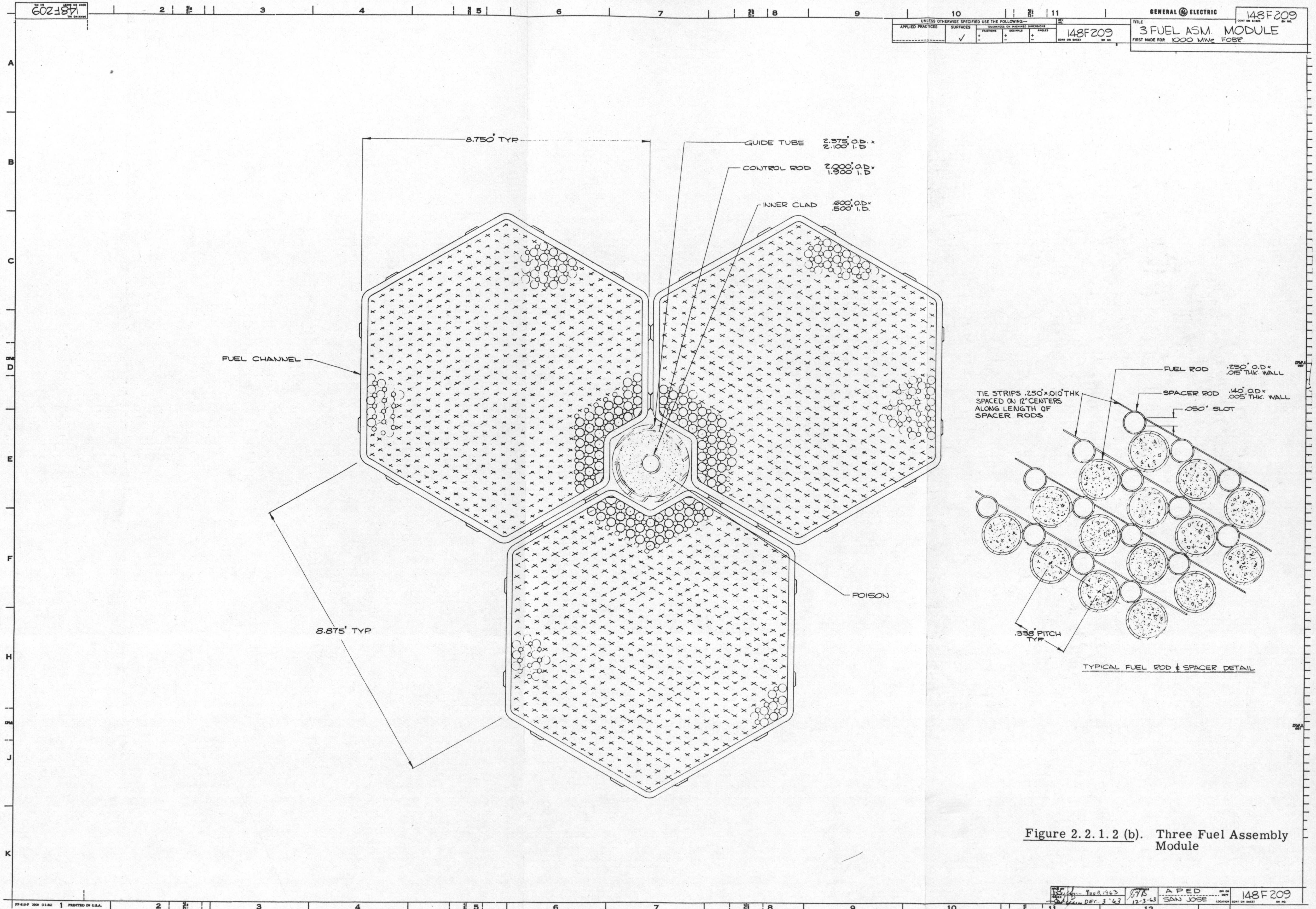

Figure 2.2.1.2 (b). Three Fuel Assembly Module

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2.2.1 CORE CONFIGURATION

2.2.1.2 Mechanical Arrangement (Continued)

1. accurate positioning of the lower end in the support grid,
2. close spacing between channels near core center elevation by pads on fuel channel walls,
3. close spacing between assemblies in the shielding region above the core, and
4. a continuous shroud at outer boundary of radial blanket assemblies.

The spacing of the fuel assemblies and control rods results in a triangular control-rod pattern with a pitch of 15.37 inches. There are 97 available locations within the core and radial blanket regions; 85 are planned for use for the dual-purpose control and safety poison rods. The remaining 12 positions located between the core and radial blanket fuel assemblies have dummy guide tubes, some of which will be used as reference points to facilitate fuel handling in the refueling process. The remaining dummy guide tubes may be used to house neutron sources, etc.

2.2 REACTOR PLANT (Continued)

2.2.2 FUEL AND BLANKET ASSEMBLIES

Each fuel assembly consists of a fuel region, upper axial blanket, lower axial blanket, fission gas reservoir, and neutron shielding. The assembly has an over-all length of 23 feet, a hexagonal cross section 8.75 inches across the flats, and weighs 1,200 kgs (2650 pounds).

The components of the assembly are listed as follows:

	Quantity
Outer hexagonal channel	1
Fuel rods	470
BeO rods*	138
Filler rods	7
Fuel rod spacers	650
Fuel rod support bars	29
Nose piece	1
Channel to neutron shield adaptor	1
Fuel assembly handle	1
Neutron shield rods	17
Flow sample tube	1
Intermediate grid for shield rods	2

The arrangement of the assembly is shown on Figure 2.2.2.

The major structural member in the core region is the stainless steel fuel assembly channel. It has a thickness of 0.15 inch. As shown in Figure 2.2.1.2.(b), when each group of three assemblies is properly oriented, a hexagonal opening 2.34 inches wide is formed between them for the control rod. The indentation occupies the equivalent space of sixteen fuel rods. The weight of the 670 fuel pins and associated 138 BeO pins is supported in the assembly by means of their lower end plugs which contain "T"-sectioned slots. The "T" slot slips over the "T"-sectioned support bar attached to the lower end of the fuel assembly thus locating the pins and supporting their weight.

A number of the BeO pins are arranged in the fuel assembly in contact with the channel wall at the outer edge of the assembly, and the remainder are dispersed within it. Since their coolant requirement is moderate compared to that of the fuel rods, the elimination of the outer ring of spacers and the resulting 30 percent reduction in coolant space per rod is permissible.

2.2.2.1 Fuel Spacer

Continuous spacing of the fuel rods throughout their entire length is provided by slotted stainless steel tubes with a 0.140-inch OD and a 0.005-inch wall thickness.

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*BeO is dispersed throughout the core for the purpose of making the Doppler coefficient more negative.

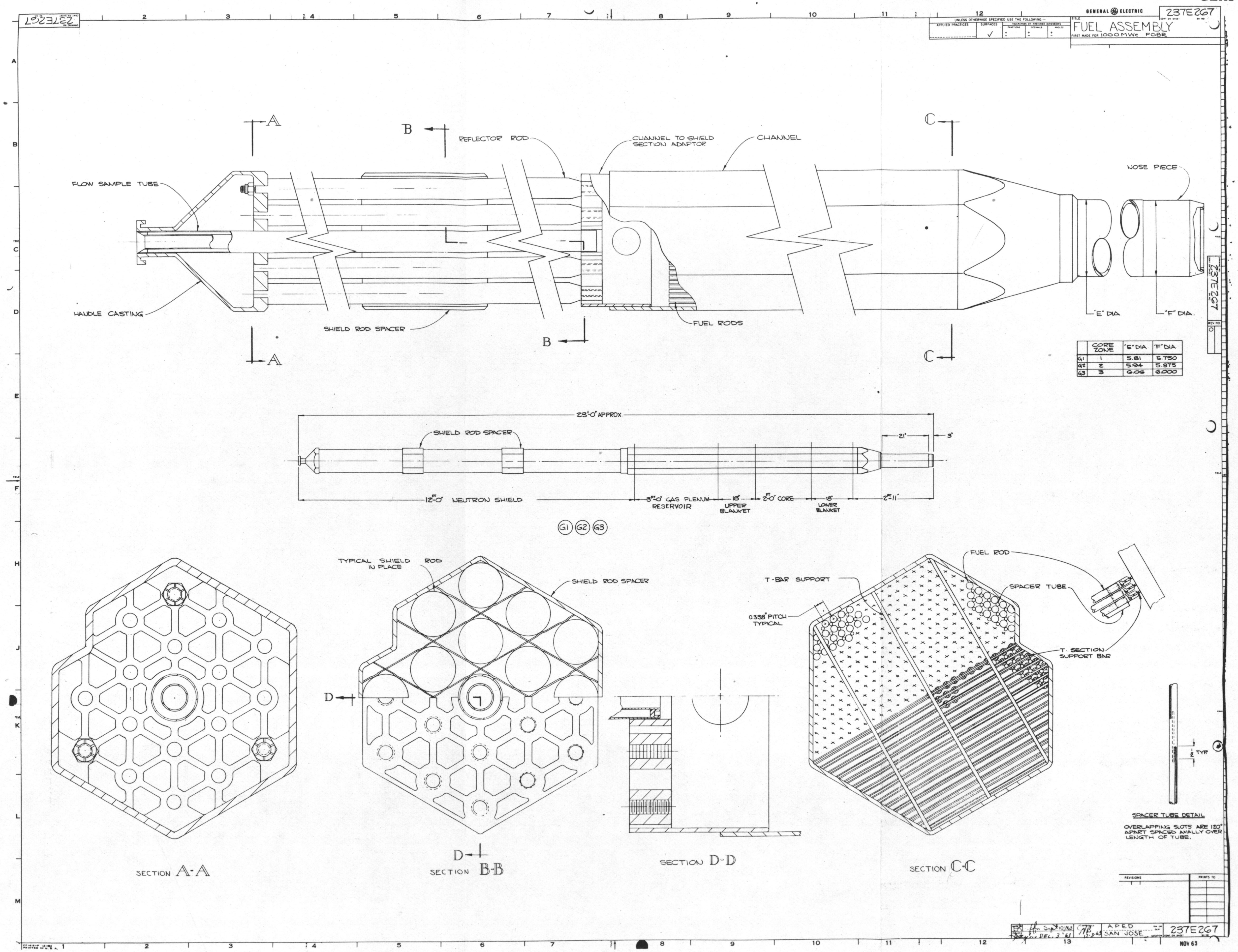

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Figure 2.2.2. Fuel Assembly

2.2.2 FUEL AND BLANKET ASSEMBLIES

2.2.2.1 Fuel Spacer (Continued)

There is nominally one spacer tube provided for each rod. The spacer tube at a typical cross section [illustrated on Figure 2.2.1.2. (b)] appears as a split ring in contact with three surrounding fuel rods. The fuel and BeO rods are supported by three equal-spaced continuous-line contacts along their length on a 0.338-inch triangular pitch, thus leaving ~88-mil spacing between heated surfaces. The use of a slotted tube results in a flexible support permitting differential expansion, initial assembly, and interchannel coolant mixing. The spacer tubes are the same length as the fuel and BeO rods, and are held in place axially by a series of tie strips on 1-foot centers along the full length of the spacer tubes.

Different spacer members are required for the corner rods adjacent to the fuel assembly channel. Solid steel bars are assumed tentatively for these locations. The fuel rods and BeO rods are not attached at their upper end since they are supported at the lower end and are positioned tightly throughout their length by the spring action of the spacers. This allows for vertical thermal expansion of the individual rods and an unobstructed exit for the coolant from the core region.

2.2.2.2 Inlet Nozzles

The nose piece, channel-to-neutron-shield adaptor, and fuel assembly handle are stainless steel castings finish machined for their specific application.

The nose piece is the vertical support member for the fuel and BeO rod bundle and is attached mechanically to the fuel assembly channel which gives lateral support to the rods. The region in which the attachment is made to the fuel rod support bars and fuel assembly channel is of the same cross section as the fuel assembly channel. Directly below this region a transition is made to the cylindrical portion which fits into the core support grid. This section is approximately 2-1/2 feet long with a machined lower end which guides the nose piece into, and accurately positions it within, the lower plate of the 2-foot-deep grid structure. The fuel assembly load is taken on the upper plate by the conical surface of the nose piece. This surface contacting the mating surface of the grid structure forms the seal between the inlet plenum and region outside the fuel assembly channels. Positioning the fuel assemblies at two elevations in the support grid will result in a "free-standing" fuel element without causing problems during fuel handling operations, since a cylindrical fit is used at one of the locations and a tapered seat without close diametral clearances is used at the other.

The problem of preventing the higher-enriched assemblies in the outer zone of the core from being inserted erroneously in any of the other two zones of higher neutron flux is accomplished by utilizing a larger-diameter cylindrical section nose piece for the higher-enrichment fuel assembly. This does not eliminate the possibility of installing a lower-enrichment fuel assembly in a high-enrichment fuel zone. However, this does not present a safety problem and any errors of this nature will be detected by the core temperature instrumentation (see Section 2.2.6).

2.2.2 FUEL AND BLANKET ASSEMBLIES

2.2.2.2 Inlet Nozzles (Continued)

The lower ends of the nose pieces which pass beneath the grid serve as fixed orifices for proper flow distribution across the core. The size of the openings in each nose piece is based on the coolant requirement for that specific location. Although the size of the flow control orifice holes as well as the nose piece diameter will be different for the various core regions. the same basic casting will be used.

2.2.2.3 Shield Section

The channel-to-neutron-shield adaptor serves the dual function of: 1.) a transition piece from the fuel channel to the shielded rod section. and 2.) a handle for the fuel rod section after the shield rods are removed for spent fuel shipment and reprocessing.

The channeled coolant flow from the fuel assembly passes through the transition piece to the open shield rod section above the core where bulk coolant mixing and cross flow take place. A small amount of this coolant leaving the fuel region passes up through a central insulated tube in the shield section to the top of the assembly where a thermocouple is located for monitoring core outlet temperature. The 17 shield rods surrounding the flow sample tube are attached to the upper end of this transition piece whereas the flow sample tube is only guided at this end.

The central rib across the casting. shown on Figure 2.2.2, Section D-D, serves as the handle when the shield section is removed. This provision simplifies the spent fuel handling and reprocessing problems by reducing the fuel assembly to less than half its original length.

A neutron shield region is located above the core to protect the reactor vessel head from excessive activation. The 12-foot-high shielding region is in the form of steel rods sized and spaced to result in approximately 50 percent steel and 50 percent coolant flow space. As mentioned previously. these shield rods form the upper end of the fuel assembly. Each assembly has 17 1-5/8-inch diameter solid rods which surround the central flow sample tube. Grid-type spacers are used at two locations along the length of this rod cluster in order to provide the stability necessary for operation as well as the rigidity required for fuel handling.

The fuel assembly handle is a stainless steel casting which has a flared upper end serving as the flow sample tube outlet as well as a point of attachment of the fuel handling grapple or extension rod. The flow sample tube is attached (seal welded) at this point to eliminate the possibility of bulk coolant mixing with the sample tube flow prior to exit from the fuel assembly handle. The shield rods are attached at the lower part of this casting.

2.2.2 FUEL AND BLANKET ASSEMBLIES (Continued)

2.2.2.4 Radial Blanket Assembly

The radial blanket assembly (Figure 2.2.2.4) is identical in many respects to the fuel assembly described in the preceding section. The only difference is in the fuel region, which will be described in this section.

Each assembly has 208 fuel rods, 0.500 inch in diameter and approximately 5 feet long. The rods are supported from the bottom of the assembly in the same manner as the core assembly fuel rods. They are spaced relative to one another and from the channel wall on a 0.565-inch triangular pitch by means of a 0.065-inch diameter wire wrap attached to each rod. The only unusual spacing problem is that caused by the indented corner of the fuel assembly channel. As shown on Figure 2.2.2.4, Section C-C, a 0.300-inch-thick angle section spacer is required at this location where the nine fuel rods were removed. This will give proper positioning and lateral support to the remaining fuel rods in the assembly.

The fuel rods have an active fuel length of 36 inches which is vertically centered relative to the 24-inch-high core region. Steel end plugs, approximately 15 inches long, are used to position properly the fuel section and support the fuel rods from the nose piece at the lower end of the radial blanket assembly. An 18-inch-long gas plenum and short end plug exist at the upper end of the fuel rod above which there is a void region (filled with coolant when installed in the reactor) approximately 3 feet long. This void region inside the channel is in line with (and in place of) the approximately 3-foot-long fission gas plenum of the core assembly fuel rods.

2.2 REACTOR PLANT (Continued)

2.2.3 FUEL AND BeO PIN CONSTRUCTION

All pins making up the core structure are 8 feet in length and use 0.25-inch OD, 15-mil wall thickness stainless steel cladding. In the case of both the fuel rods and the beryllia pins there is a 3-foot gas plenum provided at their upper ends. The design data is based on extensive stainless steel clad oxide fuel experience in water reactors and more recent fast oxide breeder development program results.[1,2,3]

Figure 2.2.3 illustrates the details of the construction of both BeO and fuel pins. Material specifications and methods of fabrication for the reference design are given below.

2.2.3.1 Materials Specifications

Clad Material

The material for the fuel cladding is to be fully annealed type 316 stainless steel seamless tubing with a 0.25-inch OD ±0.001 inch and a 0.015-inch wall thickness ±0.0005 inch. The finished tubing is to be supplied in 8.25 foot lengths, tested ultrasonically, and having the following minimum mechanical properties at room temperature:

Yield strength (0.2 percent offset)	30,000 psi
Ultimate strength (tensile)	75,000 psi
Elongation	35 percent

The top and bottom end plugs are to be fabricated of similar material. BeO rods will be clad with the same materials.

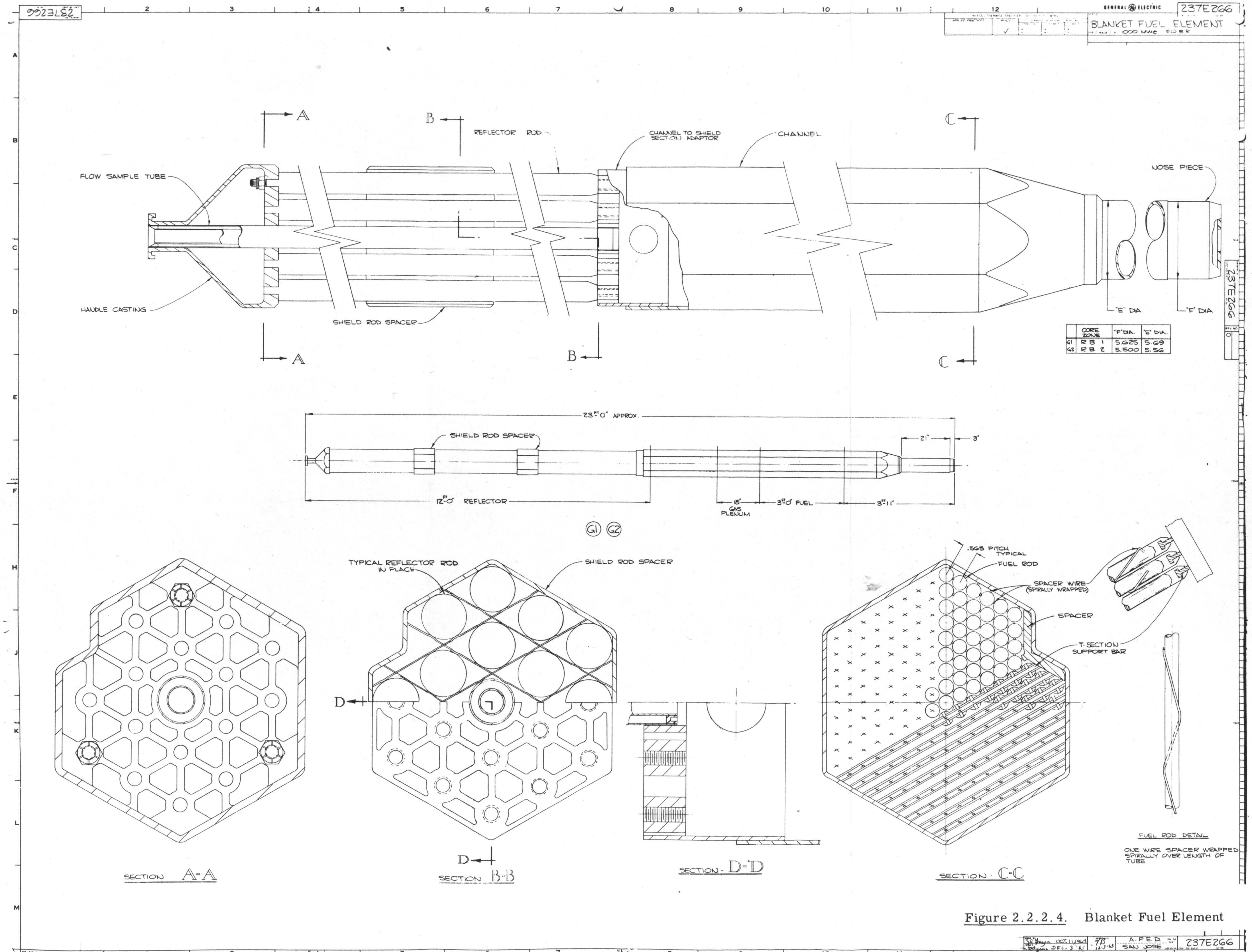
Radial blanket rods will be fabricated with materials to the same specifications having a 0.5-inch OD ±0.002 inch and a 0.020 ~~0.015~~-inch wall thickness ±0.001 inch.

Oxide Fuel and Blanket Materials

The mixed oxide fuel materials are to be prepared using the co-precipitation of ammonium diuranate, plutonium hydroxide route. The material is to be pressed and sintered to yield pellets no greater than 0.215 inch in diameter having a fuel content of 5.75 ±0.15 gms/inch of stacked length; this corresponds to a density of 9.52 gms/cc (85.76 percent theoretical) at maximum permitted diameter. No grinding of pellets is required.

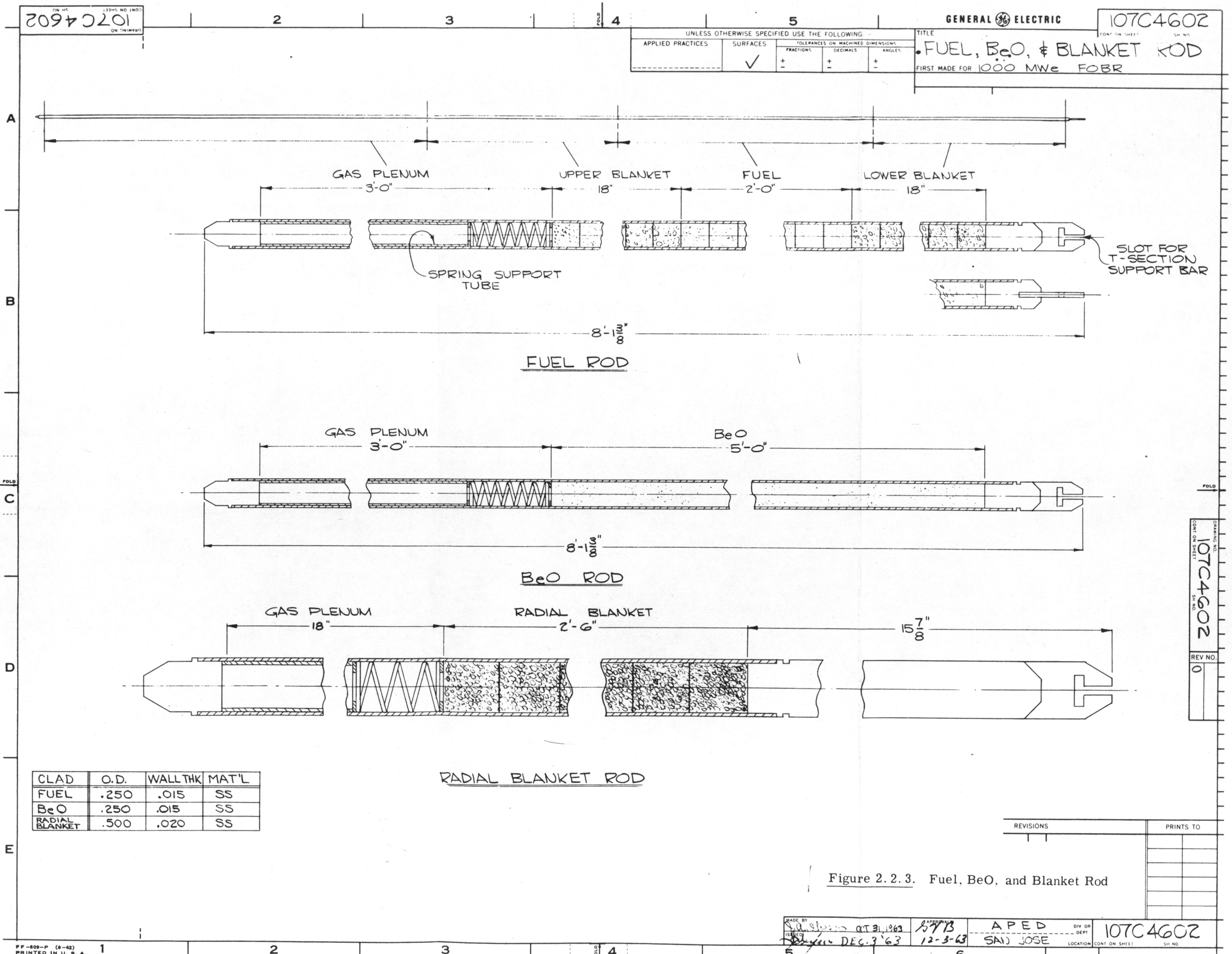
The pellets are to have an atom ratio of oxygen to heavy metal of 1.97 ±0.03.

The depleted uranium blanket material is to be fabricated using similar processes and to the same specifications.



	CORE ZONE	"F" DIA.	"E" DIA.
G1	R B 1	5.625	5.69
G2	R B 2	5.500	5.56

Figure 2.2.2.4. Blanket Fuel Element



CLAD	O.D.	WALL THK	MAT'L
FUEL	.250	.015	SS
BeO	.250	.015	SS
RADIAL BLANKET	.500	.020	SS

Figure 2.2.3. Fuel, BeO, and Blanket Rod

2.2.3 FUEL AND BeO CONSTRUCTION (Continued)

2.2.3.2 Fabrication

Loading of Fuel and Blanket Materials

The mixed uranium plutonium oxide fuel and depleted uranium oxide blanket pellets are to be loaded into the same length of fuel cladding using a semiautomatic loading technique. This technique will involve assembly of a 5-foot-long pellet stack (18 inches each of depleted and 2 feet of active fuel), gamma scanning of the assembly to ensure correct length and location of the active fuel, followed by its insertion into the clad from the lower, nonfission-gas reservoir, end of the clad until the spring and retainer ring, previously located within the clad, are reached.

Welding and Inspection of Fuel Pins

Welding is to be accomplished in two stages: initial welding of the upper end plug and location of the fuel retainer ring and its spring are carried out together with an initial (2,000-psi pressurization) helium leak test prior to fuel loading. Following the fuel loading the fuel assembly is flushed (pressurized) with helium and the second end plug is welded in a helium atmosphere. A second leak test by no pressurization test is carried out at this time.

A final radiographic and gamma scan inspection plus external diameter go-no go gaging and weighing are carried out before pins are shipped to the fuel assembly area.

Beryllia Pin Fabrication

Granular beryllia material, prepared by sintering and crushing UOX grade BeO, is to be compacted vibratorily into cladding tube lengths prepared in a manner identical to that used for fuel pins. A single sintered BeO pellet is to be loaded ahead of the vibratory-compacted material to prevent the latter passing through the spring and retainer ring and thence into the 3-foot gas reservoir also provided in the BeO pins.

Similar leak checking and final radiographic checking is planned to that proposed for the fuel.

2.2 REACTOR PLANT (Continued)

2.2.4 THERMAL DESIGN

The reactor is power flattened to maximize its thermal performance by use of three-zone core fuelling pattern; thus, peaking factors of 1.24 radially and 1.14 axially [see Figures 2.5.2.2(a) and (b) and Table 2.2.4] are achieved. The core power rating is nominally 2,200 MWt; a further ~200 MWt are generated in the axial blankets and the balance of the 2,500-MWt total power output is generated in the radial blanket assemblies [see Table 2.5.2.2(b)].

2.2.4.1 Heat Removal

The coolant flow through core and radial blanket regions is orificed to maintain a nominally-uniform outlet coolant temperature of 1100 F from all channels. Deviations from the design exit temperature will arise due to local peaking effects, fuel burnups, orificing errors, and transient conditions. A maximum possible outlet temperature of 1300 F is computed for the hottest assembly during a 10 percent over-power transient. The maximum coolant clad surface temperature under this circumstance is computed to be 1332 F. The maximum coolant-clad surface temperature computed, disregarding local peaking effects, is approximately 1150 F.

The design maximum coolant velocity is 13.6 ft/sec within the channel, and sufficient coolant mixing between parallel flow channels within the same fuel bundle is planned to prevent significant local temperature variations.

Coolant flow through the control rod guide tubes and thermal shields is similarly designed to yield a nominal exit temperature of 1100 F.

Natural convection flow through the core and primary circuit at design temperatures will remove up to 8 percent of full power from the core.

2.2.4.2 Fuel Temperatures

The maximum fuel rating in the core is 22.7 Kw/ft and will occur only in those sections of the hottest fuel pins in which maximum local peaking conditions are present. Furthermore, it will occur only under an assumed 10 percent transient overpower condition (worst operating incident - see Section 3.2).

The average linear power generation of fuel within the core is 10.4 Kw/ft. This results in an average fuel temperature of 2400 F for all fuel material within the core. The maximum fuel temperature in the hottest point of the hottest rod has been maintained by design of less than the estimated melting point of mixed UO_2 PuO_2 of approximately 4900 F (see Section 3.4.6.3). The temperature distributions along the length of the hottest rod are shown in Figure 2.2.4.2.(a) (see Appendix 6.2, Section 3.4.6 for more detailed design data and a discussion

2.2.4 THERMAL DESIGN

2.2.4.2 Fuel Temperatures (Continued)

of hot spot factors). The radial temperature distribution through the hottest rod at the hottest point is shown in Figure 2.2.4.2 (b).

The fuel has been designed to develop an appreciable central void in the peak fuel temperature region during its first rise in temperature to the full power condition. Recognition of the ability of maximum oxide fuel to redistribute from the hottest central region results in the ability to operate such fuels at a rating 25 percent above that computed for solid rod fuels having the same maximum or center temperature. The estimated central void shape for the hottest rod is illustrated in Figure 2.2.4.2(c). A section of such a rod following long-term irradiation is shown in Figure 3.4.6.2.

2.2.4.3 Clad Stresses

The maximum fission gas release from the hottest mixed oxide fuel pin is estimated to result in an internal pressure within the clad not exceeding 1000 psi. This would correspond to a maximum of 70 percent release of all fission products generated. It would result in a maximum clad stress of 8620 psi and will only occur at the end of fuel lifetime of a maximum exposure fuel pin which at that time would be operating at less than 20 percent below the maximum computed power rating of 22.7 Kw/ft. The maximum thermal stress within the 15-mil wall thickness clad operating at peak power conditions is computed to be 21,000 psi. Clad stresses from other sources are estimated to be small by comparison to the above-stated thermal and pressure stresses and thus do not influence, too significantly, the over-all performance of the clad. The maximum clad surface temperature is computed to be 1332 F and would result in an estimated short-term yield strength of approximately 20,000 psi for typical 300 series stainless steel (as shown in Figure 2 of Appendix 6.3). The maximum possible thermal plus pressure stress of 30,000 psi is thus computed to result in a limited plastic strain at temperature. This strain may be considered as a thermal prestress having a maximum value of 10,000 psi. Subsequent cycling of the fuel pin between zero and full power should not result in significant further stresses of the clad beyond its yield point. See Section 3.4.4 for a design discussion and Appendix 6.4 and 6.5 for numerical basis.

TABLE 2.2.4

PEAKING FACTORS

(a) Local Factors

Factors	Coolant Temperature Rise Across Core	Surface Heat Transfer Coefficient	Rod Power
Maldistribution of flow	1.10	1.00	1.00
Deviation from nominal dimension and density	1.025	1.025	1.025
Composition tolerance	1.02	1.00	1.02
Neutron flux peaking	1.05	1.00	1.05
Power measurement and control	1.05	1.00	1.05
Burnup in fuel rod	1.20	1.00	1.20
Heat transfer correlation	1.00	1.13	1.00
Overpower	1.10	1.00	1.10
Other unaccounted	1.01	1.00	1.00
	1.68	~~1.28~~ 1.15	1.54

(b) Gross Core Power Distribution

Radial	1.24
Axial	1.14
Total	1.41

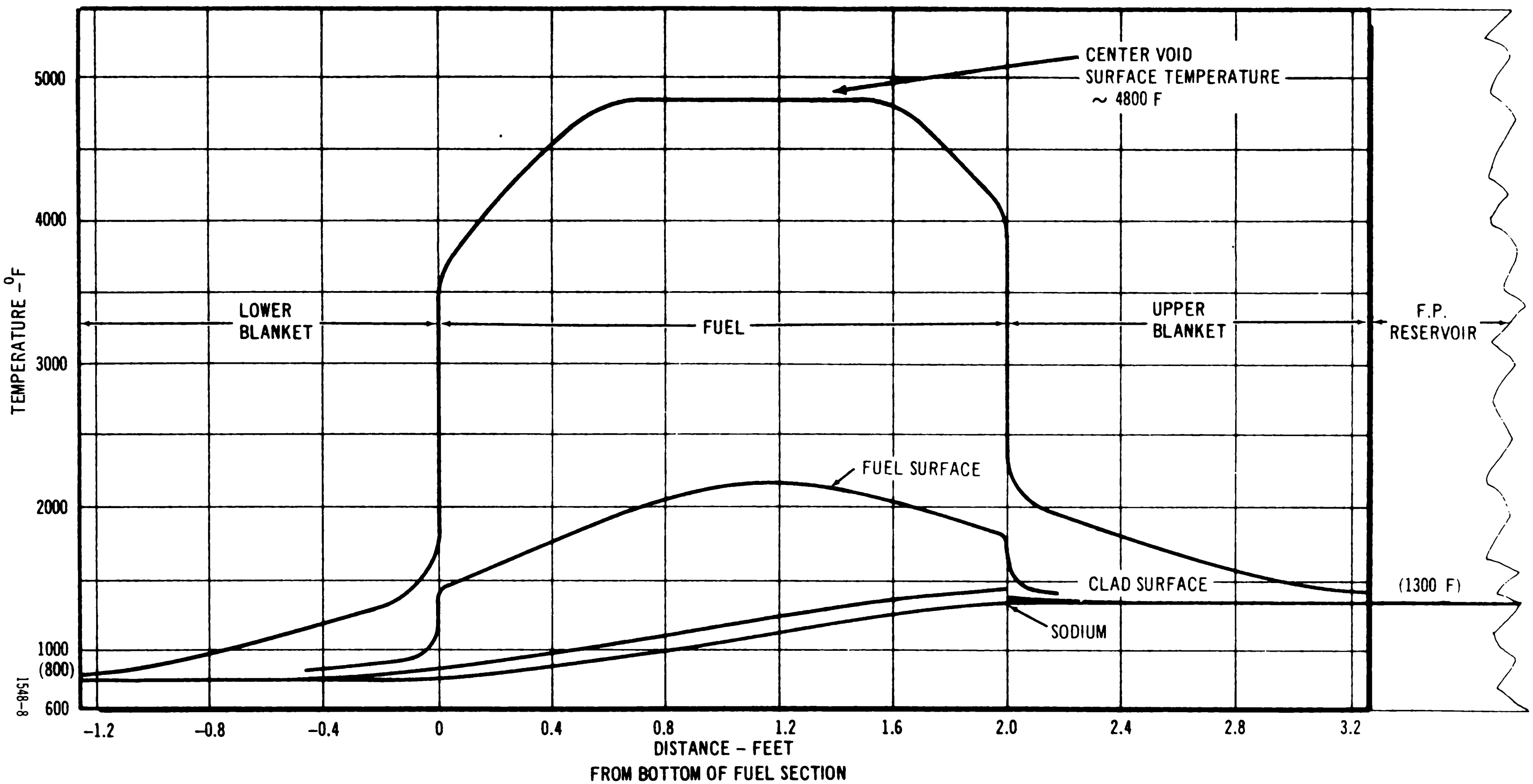


Figure 2.2.4.2(a). Longitudinal Temperature Distribution in Hottest Fuel Pin

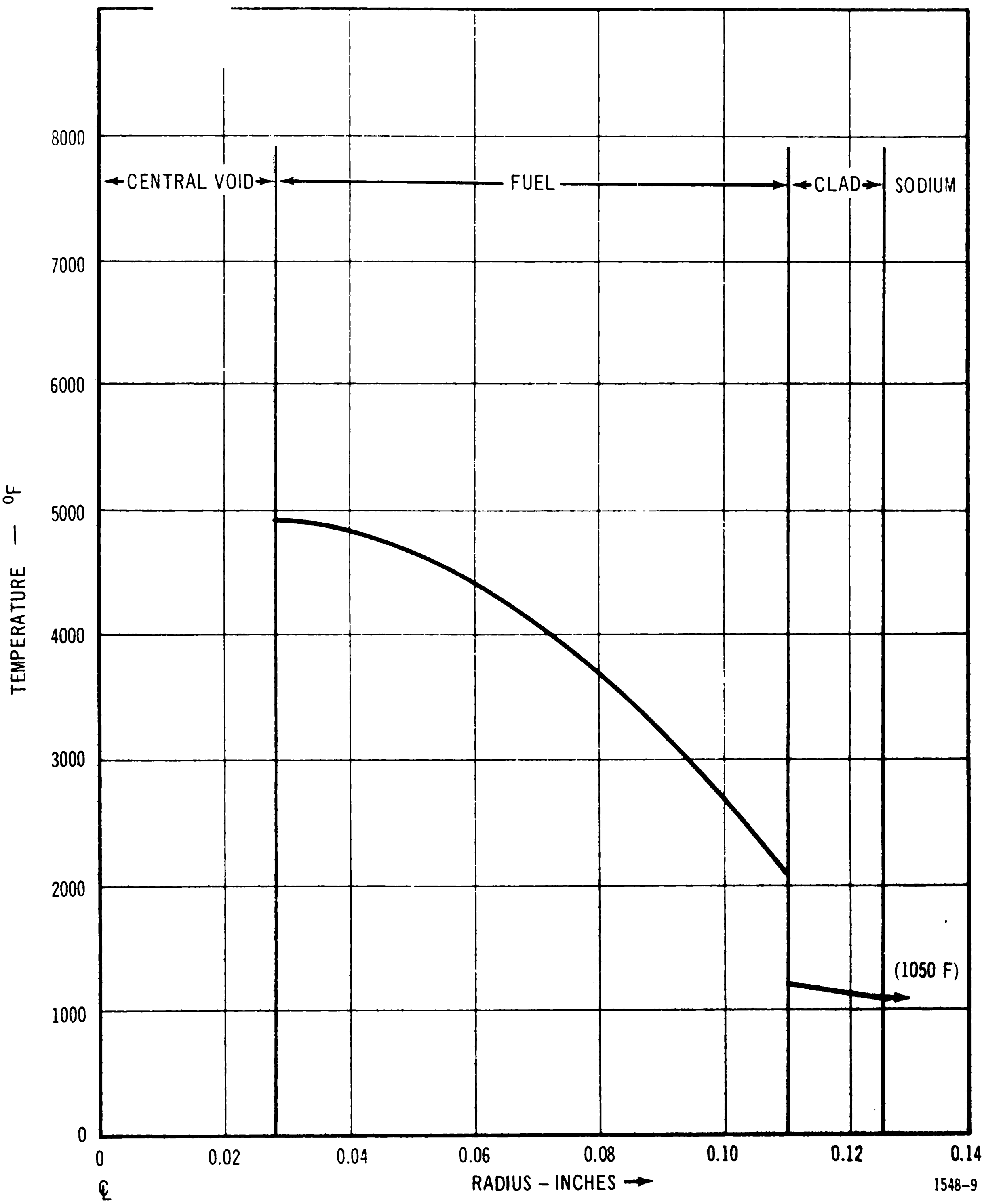


Figure 2.2.4.2(b). Radial Fuel Temperature Profiles

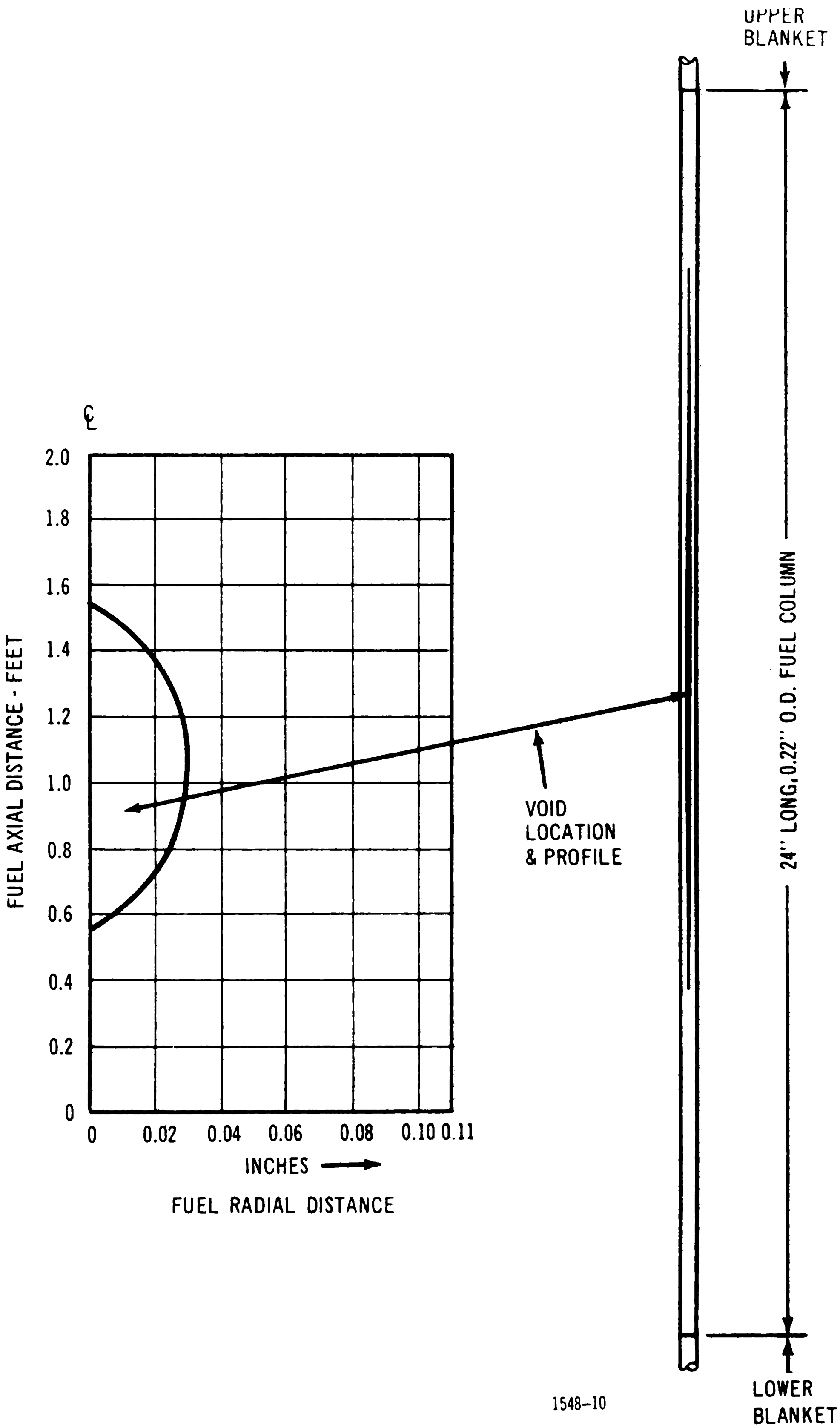


Figure 2.2.4.2(c). Estimated Void Shape in Hottest Fuel Pin

2.2 REACTOR PLANT (Continued)

2.2.5 CONTROL RODS

Each of the 85 control rods illustrated on Figure 2.2.1.2(a) consists of a 3-foot-long annular boron carbide poison section, a 2-foot-long plenum section,* and a 35-foot-long connecting rod. Rollers are provided above and below the poison and gas plenum region, as shown on Figure 2.2.5, Sections AA and DD, to position this section properly within the guide tube. Since the control rod drives are located an appreciable distance from the core, the connecting rod to the control element must also be guided carefully. This is accomplished by roller assemblies, located along the 35-foot-long connecting rod, which locate it properly within the guide tube.

2.2.5.1 Poison Section

The control element consists of vibratory-compacted B_4C in an annular stainless steel container. A disk and spring assembly is located in the region above the poison section. This performs the hold-down function to contain the poison material in the lower portion of the rod while providing a gas plenum to accommodate the helium formed from the n-alpha reaction. The control rod cladding is Type 316 stainless steel which has a 5-mil copper plate on the inside surfaces. This provision is made to prevent carburization of the cladding material if there is any free carbon present due to decarburization of the poison material. End plugs with the roller assemblies are welded to the control element thereby forming the sealed container. The thermal design of the poison section, to insure maximum B_4C temperatures kept below 1500 F, is given in Appendix 6.6.

2.2.5.2 Connecting Rod and Guide Tube

The connecting rod is made from a heavy-wall, 2-inch OD tube. There are individual roller assemblies spaced equally along the length of the connecting rod which moves within a 34-foot-long guide tube mounted in the core support grid. The relative azimuthal positioning of one roller assembly to another is 120 degrees, so the repeating pattern is three times the length of the equal axial spacing. The connecting rod terminates with a pickup head at the upper end where the connection is made to the control drive assembly.

2.2.5.3 Drive Mechanism

The reference control rod drive assumed is a modified EBR-II type.

The drives are connected to the control rods by grippers which are actuated by sliding sleeve cams in a positive opening or closing manner. A sensing plunger makes contact with the top of the control rod and indicates that a control rod has been gripped.

A rack-and-pinion drive is the prime mover of the control rod, through a control-rod actuator shaft. The upper end of each control-rod actuator shaft is attached to the piston of a pneumatic cylinder which contains nitrogen under pressure. This constantly exerts a downward force on

* Insufficient to accommodate total gas release by B_4C from continuous insertion in core for its full 12-month lifetime.

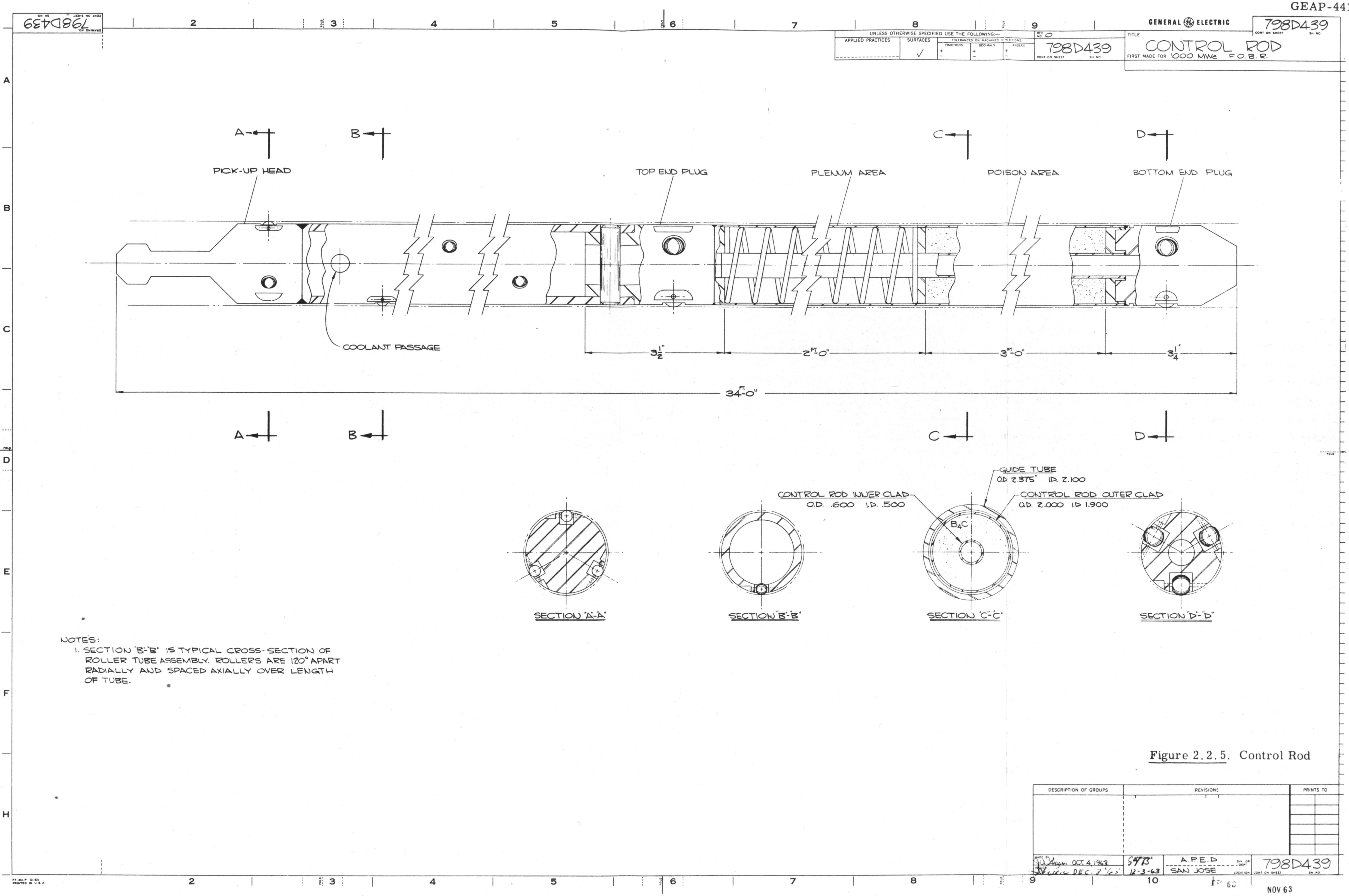


Figure 2.2.5. Control Rod

2.2.5 CONTROL RODS

2.2.5.3 Drive Mechanism (Continued)

the control rod actuator shaft and will drive each control rod into the core upon a scram signal at any position in the operating stroke. Deceleration of the scram stroke is achieved by hydraulic snubbers located near the end of the stroke.

The scram is followed by a fast rundown to insure final return of the rod into the core. Where the drive penetrates the reactor cover, a suitable seal is used that does not require rubbing or friction on the drive shaft.

2.2 REACTOR PLANT (Continued)

2.2.6 CORE INSTRUMENTATION

The core instrumentation consists of: 1. thermocouples mounted above each fuel and radial blanket assembly, and 2. ion chambers located below the lower axial blanket.

There is no permanently-installed, failed-fuel detection device. Some consideration has been given to incorporation of local sodium boiling detectors based on sonic effects. Such units may well indicate seriously-failed fuel.

2.2.6.1 Thermocouples

Thermocouples are positioned, in guide tubes under the vessel head such that with the vessel head in place there is a thermocouple located in the flow sample tube of each fuel assembly (see Section 2.2.2, "Fuel and Blanket Assemblies"). These guide tubes and the thermocouple leads are brought out through instrument nozzles on the vessel head. The combination of coolant inlet temperature and known flow orificing for each fuel assembly, plus the individual outlet temperature instrumentation, will enable one to monitor the power generation of each fuel assembly.

Instantaneous read-out of core exit temperature variations are not available from this system because of the distance the thermocouple is located from the active core. It does, however, provide local power generation information; this lags the operating condition change by only a few seconds. With the reactor at power, the transit time for the coolant from the core to pass through the flow sample tube to the outlet thermocouple location will be on the order of 3 seconds.

2.2.6.2 Ion Chambers

Ion chambers will be located within the core support grid structure which is directly beneath the lower axial blanket. There will be seven of these spaced uniformly across the grid structure.

These ion chambers are inserted through guide tubes which penetrate the operating room floor, pass through the reactor support structure and the vessel downcomer annulus, then turn upward and terminate within the core support grid structure. It will be possible to replace these from the operating room floor by withdrawing individual chambers into removal casks. The replacement can then be inserted through the guide tube into position below the core.

REFERENCES

(1) Gerhart, J. M., "The Post-Irradiation Examination of a PuO_2 Fast Reactor Fuel." GEAP-3833, November 1961.

(2) Field, J. H., "Experimental Studies of Transient Effects in Fast Reactor Fuels. Series I. UO_2 Irradiations." Fast Ceramic Reactor Development Program. GEAP-4130, November 15, 1962.

(3) O'Neill, G. L., "Fast Ceramic Reactor Development Program Experimental Studies of Sodium Logging in Fast Ceramic Reactor Fuels." GEAP-4283, October, 1963.

2.3 REACTOR SYSTEMS

2.3.1 PRIMARY TANK

2.3.1.1 Component Accommodation
2.3.1.2 Coolant Flow Path
2.3.1.3 Coolant Leakage
2.3.1.4 Tank Penetrations
2.3.1.5 Component Activation
2.3.1.6 Shielding

2.3.2 REACTOR VESSEL

2.3.2.1 Internal Structures
2.3.2.2 Top Closure
2.3.2.3 Coolant Flow Path

2.3.3 FUEL HANDLING SYSTEM

2.3.3.1 Refueling Equipment
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2.3.4 PRIMARY COOLANT SYSTEM

2.3.4.1 Primary Coolant Pumps
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2.3.4.4 Sodium Purification Unit
2.3.4.5 Component Removal Method
2.3.4.6 Primary Gas System
2.3.4.7 Inert (Nitrogen) Cover Gas System

2.3 REACTOR SYSTEMS

The reactor systems design for the reference is based on the submerged primary and visual refueling concepts (see Section 3.1 for discussion). Major systems components for the 1000-MW plant include:

A primary tank which accommodates all components of the primary circuit, maintains them at uniform temperature, and minimizes the possibility of coolant loss accidents.

A reactor vessel, mounted centrally within the primary tank, in which are the core, ~~and~~ blankets, and part of the ~~radiation~~ shielding.

Primary coolant circuits, six in number, the components of which are mounted around the perimeter of the primary tank and by means of which the reactor's heat is transferred to the intermediate (nonactive) sodium system to raise steam.

A fuel handling system for introducing new fuel and removal of spent fuel from the core, separation of its shield sections, and subsequent insertion into a sodium-filled cask for shipment to reprocessing facilities.

Figure 2.3 is a plan view showing the arrangement of the reactor systems with respect to the primary tank. This equipment is shown in elevation in Figure 2.1.

2.3 REACTOR SYSTEMS (Continued)

2.3.1 PRIMARY TANK

The tank is a cylindrical vessel 52 feet in diameter with a semi-eliptical bottom and a flat shield cover structure. It is ~~suggested~~ suspended from its upper perimeter within an inert-gas-filled cavity below the operating float, and is surrounded by concrete shielding. The walls of the tank are constructed of Type 304 stainless steel plate, 2 inches thick. The tank has a design pressure of 20 psia at its maximum normal operating temperature of 800 F, which is the inlet temperature of the coolant to the reactor core. It contains approximately 3 million pounds of sodium. The tank is insulated on its outer surface by corrugated stainless steel foil or equivalent.

2.3.1.1 Component Accommodation

The reactor core, vessel, primary pumps, intermediate heat exchangers, primary coolant purification equipment, and fuel handling equipment are suspended from the primary tank cover. The arrangement of this equipment is shown in the plan view of the primary loop, Figure 2.3.1.1 (a).

The design concept for the support structure for the primary equipment shown pictorially on Figure 2.3.1.1 (b) is formed by three plates welded to the centrally-located reactor vessel. The internal bracing of the structure is designed to form the ducting for the primary sodium flow. The lower and middle plates contain the pump casings and piping to the reactor inlet. The upper and middle plates contain the ducting with thermal baffles for the flow from the reactor outlet to the intermediate heat exchangers. Thus, all structural members are bathed in coolant at the reactor inlet temperature. The structure is supported from the sides of the primary tank.

2.3.1.2 Coolant Flow Path

The primary sodium circulating system consists of six loops. Check valves are used in the pump discharge line to permit isolation of a failed pump and continued operation at reduced power with the remaining loops. No other valves are used in the primary loop. Coolant flows from the pumps through the support structure to the reactor vessel. Discharge of coolant from the vessel is then through insulated ducts in the support structure to the intermediate exchanger. The return path for the sodium flowing from the intermediate heat exchangers to the pump inlet is via the tank, as noted on Figure 2.1. The flow discharge from the six heat exchangers augmented by the 2 percent flow from the pump seals in the upper region of the tank should cause sufficient mixing and turbulence to keep all structures at essentially uniform temperature. At low flow conditions, stratification may be avoided with the use of mixing baffles. Temperature stratification within the tank and its effect on structural stress and deflection is a problem that will require further study and most likely some development testing.

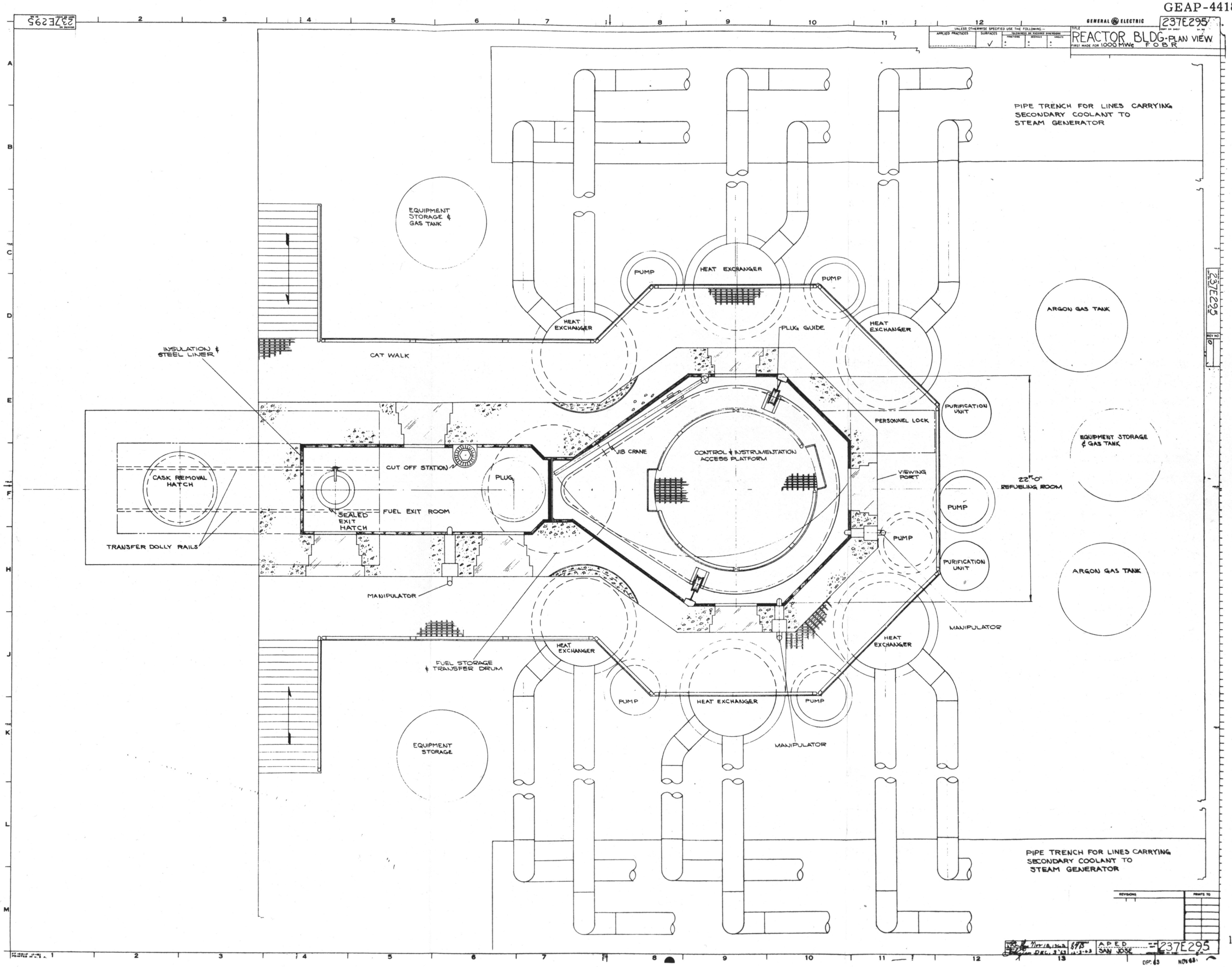

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Figure 2.3. Plan View of Reactor

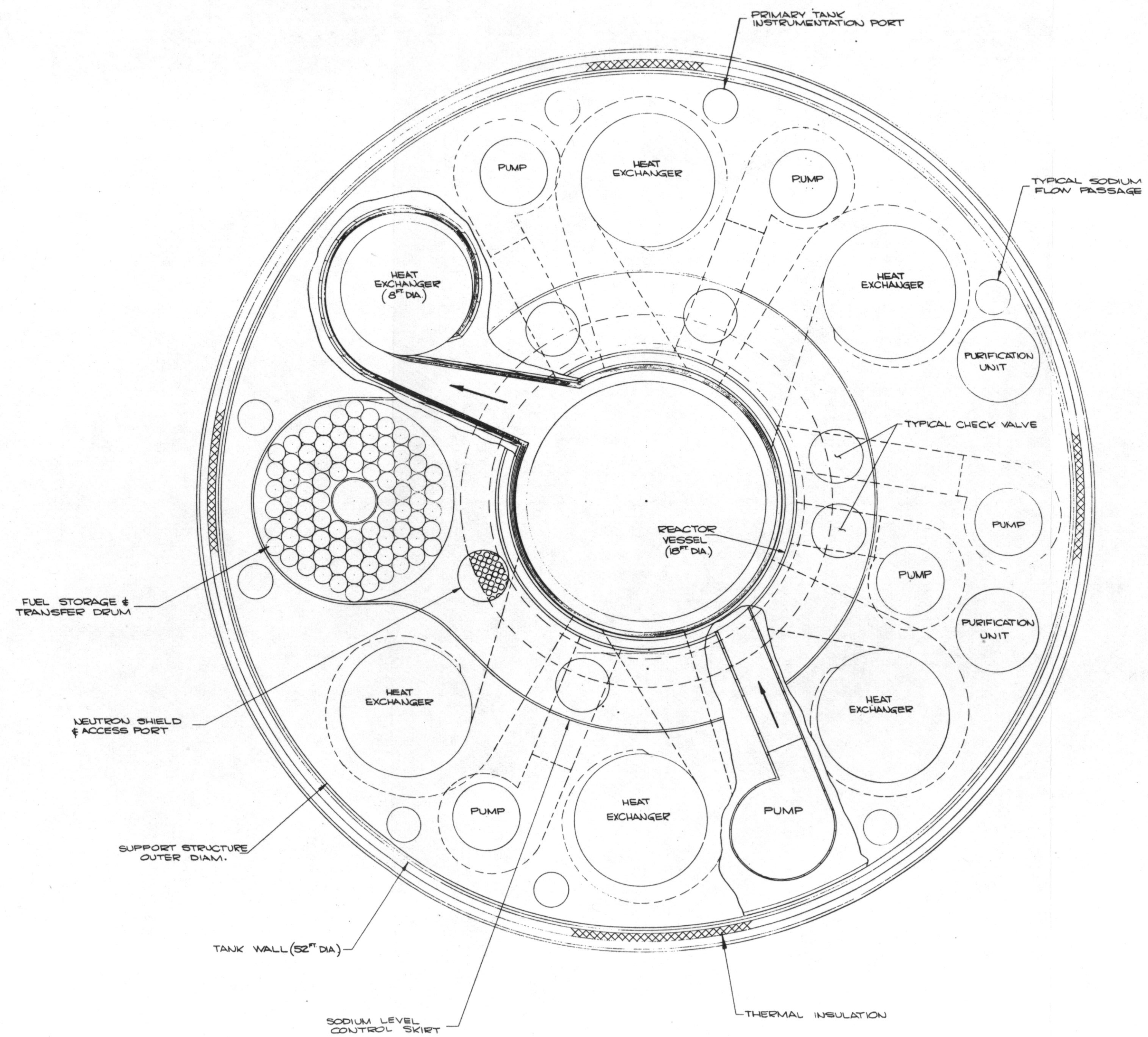

Figure 2.3.1.1(a). Plan View of Primary Loop

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2.3.1 PRIMARY TANK (Continued)

2.3.1.3 Coolant Leakage

No drain lines are attached directly to the tank, thereby minimizing the opportunity for a loss of coolant. While affecting plant performance, leakage within the vessel, ducting, heat exchangers, and related equipment, does not jeopardize cooling of the core. Provisions are made to install dip tubes for draining the tank, if necessary. The need to remove excess sodium from the tank may occur as a result of uncorrected leaks in the intermediate heat exchanger, which would allow secondary sodium to leak to the primary tank. In this event, the excess sodium can be removed with an overflow control.

2.3.1.4 Tank Penetrations

The tank cover has a penetration for the primary equipment which is made pressure tight by means of a gas seal. An inert cover gas occupies the void between the sodium level and the tank cover. The seal consists of a thin member diaphragm with flexible joints at the position of the equipment penetrations.

Where penetrations in the tank cover occur, a cylindrical extension with an offset for shielding purposes is brought up through the floor shield and sealed to the floor at its upper end. Direct access is provided for maintenance or replacement by removing the split ring shield sections. At the lower end of the cylindrical extension, a seal is made to the diaphragm through a bellows joint. The bellows separates the argon cover gas from the inert gas (nitrogen) in the cell. The operating temperatures of the bellows is 800 F, pressure differential across it is only a few pounds, and the working movement about 1 inch; therefore, the working loads are very low, and reliability is expected to be high.

Thermal gradients in the shield floor and structural members are kept low by taking a large temperature drop across the cooling gas barrier. The equipment plugs (for pumps, etc.) are designed to take the thermal gradient in the same manner as in the Fermi reactor, where sand and steel shot make up a granular shield.

Nitrogen gas seals over the shield plugs are located above the floor and are attached between the flange of the cylindrical extension and the floor. This seal is directly accessible (above the shield floor) and operates near room temperature.

2.3.1.5 Component Activation

Activation of the secondary sodium in the intermediate heat exchanger is minimized by surrounding the reactor vessel with a neutron shield of borated graphite, approximately 24 inches thick. The graphite is canned in stainless steel tubes which are arranged around the vessel in four staggered rows. The cans have an outside diameter of 6-1/2 inches and an over-all length of 12 feet. The cans are sealed and a gas plenum is provided above the graphite for the storage of released helium. The cans are mounted on a rotatable support frame to permit removal through an access hole in the main support structure.

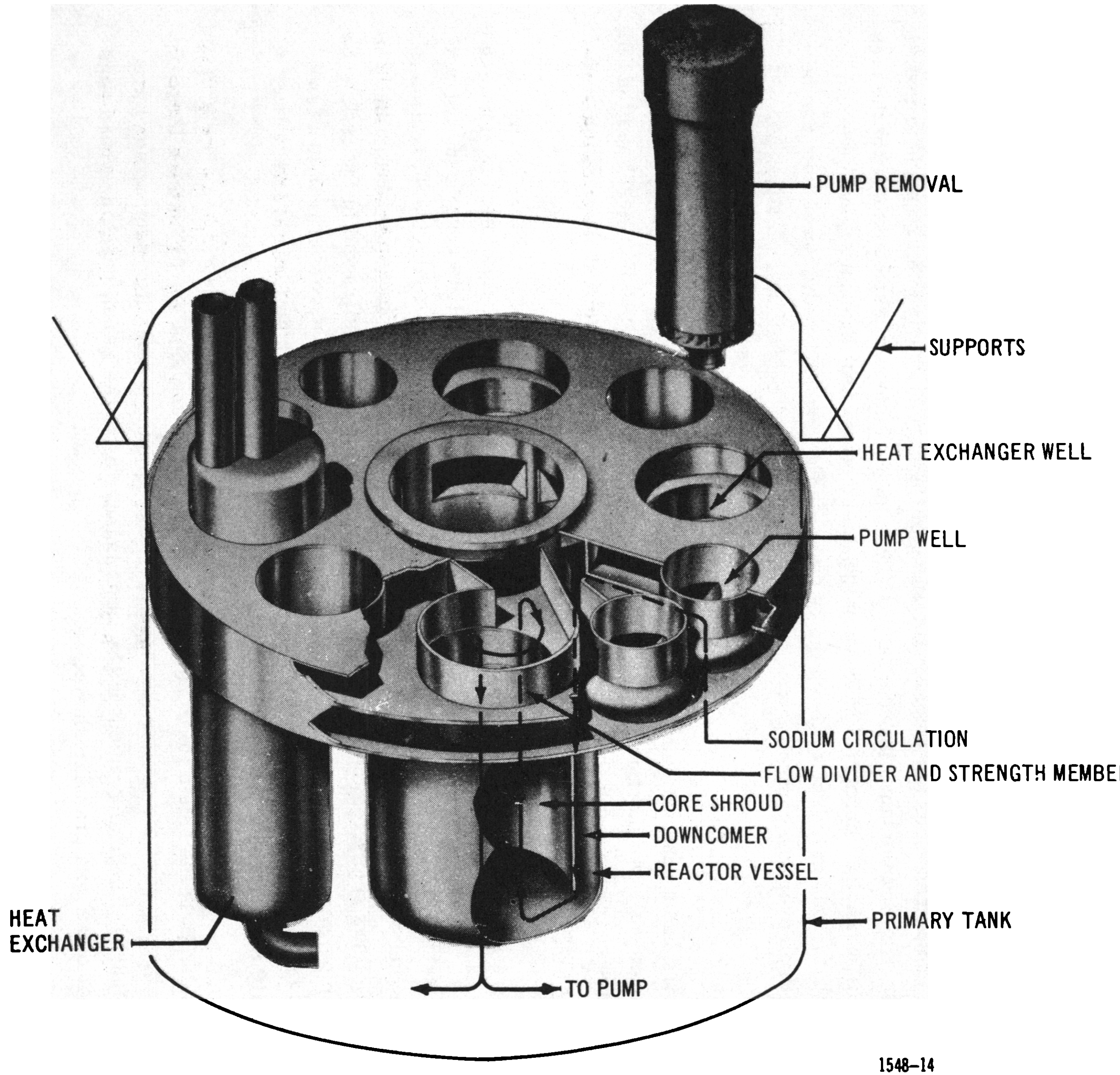


Figure 2.3.1.1(b). Pictorial View of Tank Concept

2.3.1 PRIMARY TANK (Continued)

2.3.1.6 Shielding

Personnel dose rate on the operating floor above the primary tank and during refueling and fuel handling is limited to 2.5 mrem/hr. This requires the following:

	Steel	Concrete
Operating Floor (at Full Power)	5 inches	3 feet (hgih density)
or	10 inches	6 feet (ordinary)
Refueling Room Walls (8 hours after shutdown)	12 inches	3.5 feet (high density)
Fuel transfer room walls (3 weeks fuel cooling time)	12 inches	2 feet (high density)

Dose rates at the outside of the radial concrete shield limited to 1 mrem/hr, requiring a radial concrete and steel shield of

Steel	Concrete
2.5 inches	9 feet (ordinary)

Secondary system Na^{24} activity yields a dose rate of no more than 2.5 mrem/hr at a point 12 inches from a 24-inch diameter unshielded pipe.

Incident gamma energy flux on concrete shield will be approximately 0.01 watts/cm^2.

Heat generation in the lower grid plate steel will be approximately 0.2 watts/cm^3 on the average. The life of the lower grid plate (using 10^{22} nvt of neutrons above 1 Mev criterion from thermal reactors interpreted on the basis of total energy deposition by elastic scattering) will be about 28 years. Other heavy structural members will not see conditions as severe as will the lower grid plate.

Activity of Na^{22} from the primary tank is a controlling factor as with regard to unshielded access to top plug opening region. An uncertainty of a factor of 10 places the unshielded dose rate from the primary tank Na^{22} between 20 and 200 rem/hr at a point 16 feet above the sodium surface.

Primary Na^{24} activity is 0.003 watts/cm^3, and is the controlling factor on radial and vertical steel and concrete personnel shielding around the primary sodium.

2.3.1 PRIMARY TANK

2.3.1.6 Shielding (Continued)

Radial neutron shield consisting of 24 inches of 20 volume percent stainless steel, 20 volume percent sodium, 0.6 volume percent B_4C, and 59.4 volume percent graphite has not been specifically analyzed. However, based on the analysis carried out for a 60 volume percent B_4C shield and comparing to EBR-II calculations (ANL-6614) the above shield composition should be adequate. The average heat generation in the B_4C-Graphite material is estimated to be 0.02 watts/cm^3.

2.3 REACTOR SYSTEMS (Continued)

2.3.2 REACTOR VESSEL

The reactor vessel, shown on Figure 2.3.2, has a diameter of 18 feet and an over-all height of 33 feet. The vessel is constructed of welded stainless steel plate (Type 304) 2 inches thick and has a weight of approximately 130 tons. The vessel design pressure is 100 psia at 800 F.

The vessel is supported from its upper end by the support structure described in Section 2.3.1.1. The vessel is free to expand downward and is surrounded both internally and externally by primary sodium coolant at the core inlet temperature of 800 F. The vessel is, therefore, not expected to be subject to significant thermal stresses.

2.3.2.1 Internal Structures

Approximately 6 inches of stainless steel shielding are provided around the vessel walls to protect the vessel material from gamma and neutron heating and radiation damage. The weight of the radial and bottom shielding is approximately 80 and 35 tons, respectively. The radial shield is extended above the core region to serve as a flow divider, minimize the heat loss to the inlet coolant, and protect the vessel wall from the outlet coolant temperature.

The core is supported by means of a cylindrical structure 17 feet in diameter fabricated from 2-inch plate. This cylinder also separates the inlet flow from the core outlet flow. The core support grid is attached to the bottom of this cylinder. The core support grid consists of two plates, each 1-inch thick, which are fabricated into a load supporting structure by means of an internal grid. The grid has a depth of 24 inches and by means of its depth, is used to provide lateral stability to the fuel assemblies and control rod guide tubes.

A core shroud separates the core and radial blanket from the thermal shields. The shroud, which is supported by the grid structure, is fabricated from 1/4-inch-thick stainless steel and is 17 feet high.

2.3.2.2 Top Closure

The reactor vessel cover (see Figure 2.3.2) is designed to be handled as an integral unit with the top shield plug (see Figure 2.3.3.1). Access to the core and blanket is provided by vertical removal of the shield plug and vessel cover assembly. The vessel cover has shielding plates for both thermal and radiation protection. The cover assembly also includes the fuel assembly hold-down structure and temperature sensors for the coolant flowing from the assemblies. The cover is eliptical and is constructed of Type 304 stainless steel.

2.3.2 REACTOR VESSEL (Continued)

2.3.2.3 Coolant Flow Path

The sodium coolant flow within the vessel (94.5 million lb/hr) is directed downward from the inlet (pump discharge) in the region outside the radial thermal shields to the lower plenum and core support grid. The core support grid houses the fuel assembly nose pieces and therefore the flow control orifices. Thus, coolant distribution between the various regions of the core and blanket occurs in the lower plenum. Coolant then flows upward through the core and blanket assemblies, control rod guide tubes, and thermal shield cooling passages, finally discharging inside the upper portion of the insulated core shroud structure which separates the inlet from the outlet coolant.

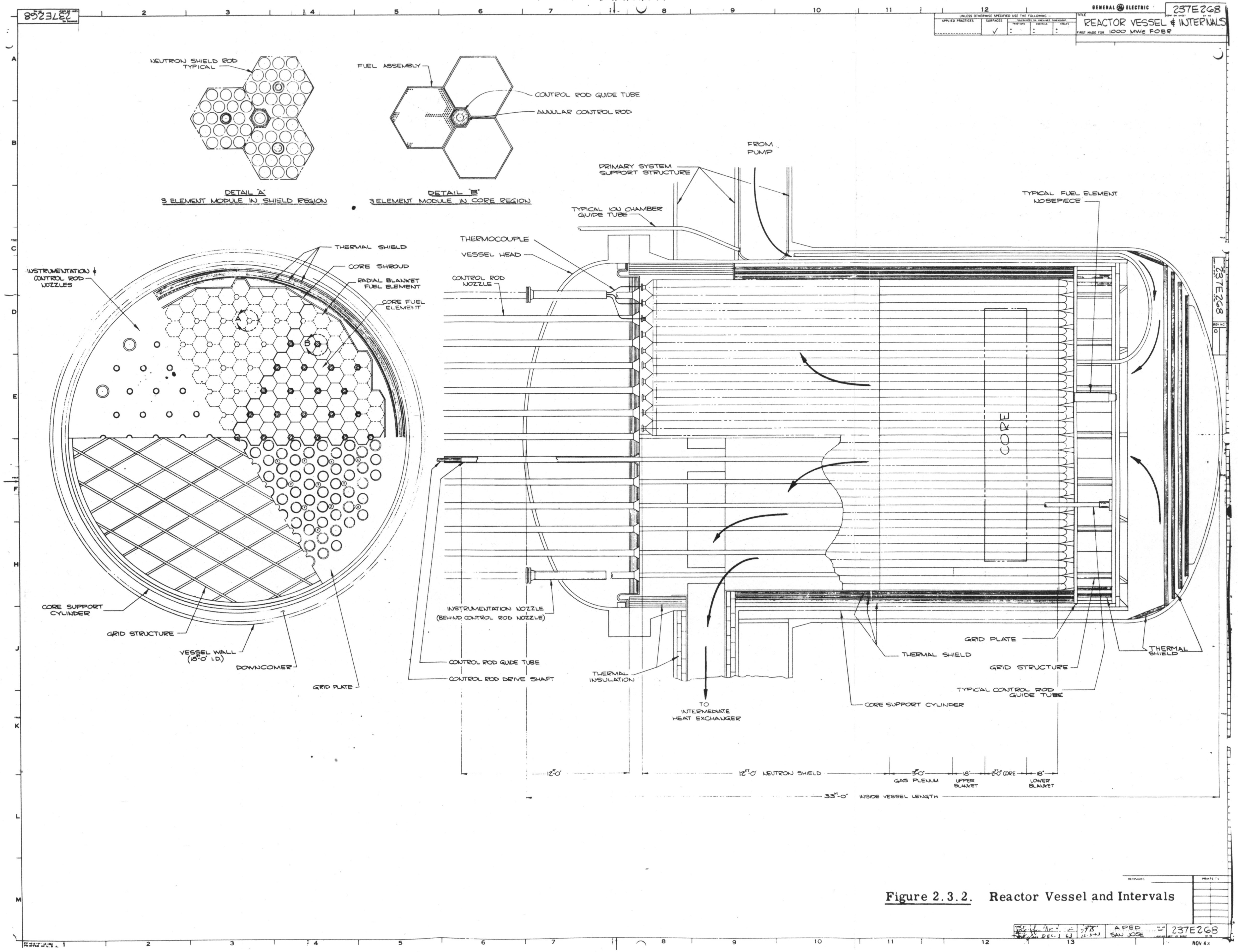


Figure 2.3.2. Reactor Vessel and Intervals

2.3 REACTOR SYSTEMS (Continued)

2.3.3 FUEL HANDLING SYSTEM

Direct observation of all fuel handling operations is planned. A large, argon-filled shielded cell is provided to permit vertical removal of the vessel head, control, drives and associated top shield as a single structure. Means are provided for adjusting the sodium level downward to permit observation of the fuel assembly handles and attachment of extension pieces, and then upward to provide a sodium-filled canal for transfer of the fuel from the core to the storage drum.

The fuel is submerged in sodium at all stages of the refueling operation, a sodium-filled, finned pot being employed during transfer and shipping stages of the operation. [It is planned to ship spent fuel from the power plant to the reprocessing facility in a sodium-filled cask, thus cooling-period requirements (4 months specified in study ground rules) become a function of shipping cask weight and reprocessing limitations only.]

The sequence of operations and sodium level changes for refueling is summarized in Figure 2.3.3. The detailing of these operations and of the time estimates for their implementations are given in Section 2.7. The required equipment is described below:

2.3.3.1 Refueling Equipment

The refueling equipment for the proposed reactor design concept is comprised of the following major items:

1. A shielded cell having sufficient head-room to provide overhead storage for control drive mechanisms and to accommodate the upper end of the fuel assembly and the fuel assembly extensions during refueling.

2. Mechanical handling equipment for the refueling operation including the cover hoist, refueling jib crane, manipulators, and accessories.

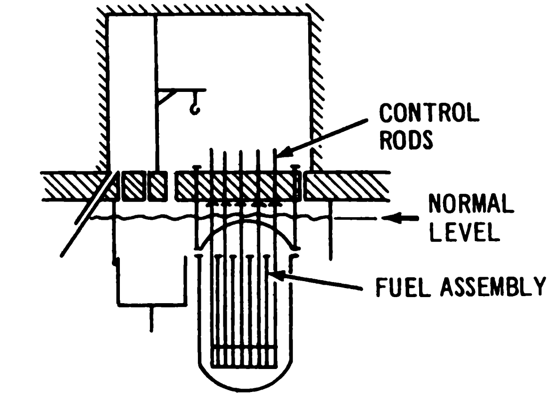
3. Fuel transfer and storage drum and associated fuel transfer cans.

4. A shielded fuel exit room and associated handling equipment.

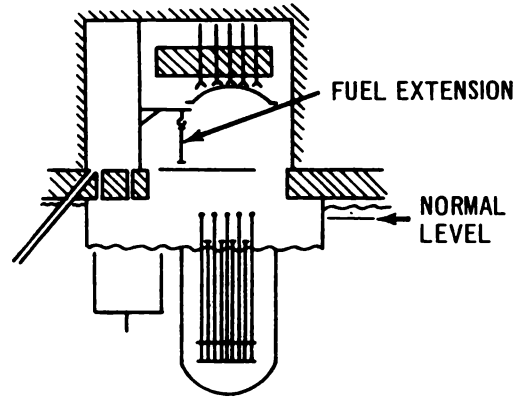
5. A fuel disassembly and shipping facility.

The arrangement of these facilities is shown on Figure 2.3.3.1.

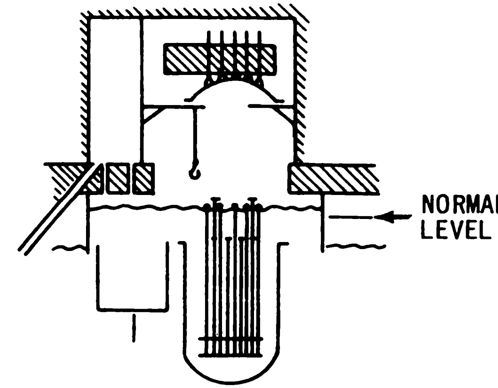
The equipment is designed to permit replacing of the 55 subassemblies (approximately one-fourth of the total core subassemblies) at each of the biannual plant shutdowns. Facilities are provided to store up to 84 fuel assemblies, if necessary.



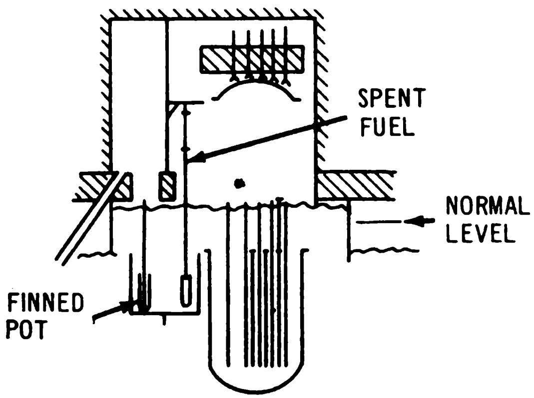
A. REACTOR SHUTDOWN
UNCOUPLE CONTROL DRIVES AND VESSEL HEAD BOLTS



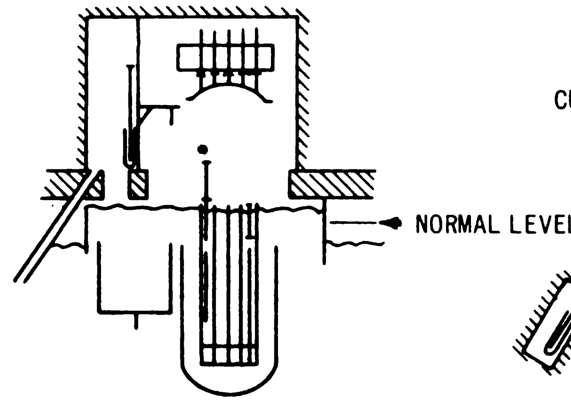
B. HEAD REMOVAL
LOWER SODIUM LEVEL AND RAISE HEAD AND IDENTIFY ASSEMBLIES FOR REPLACEMENT.



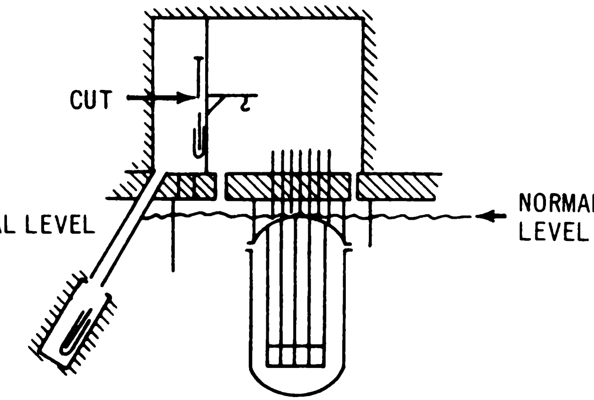
C. PREPARE TO MOVE FUEL
ATTACH FUEL EXTENSIONS, RAISE SODIUM LEVEL TO FORM TRANSFER CANAL.



D. TRANSFER SPENT FUEL
COUPLE LIFTING WINCH WITH AID OF MANIPULATOR AND MOVE FUEL TO FINNED POT IN TRANSFER DRUM - LEAVE LOCATION MARKER* REMOVE EXTENSION



E. INSERT NEW FUEL
FILL NEWLY VACATED SPACE IN CORE USING EXPOSED CONTROL GUIDE TUBES AS GUIDES FOR INITIAL PART OF INSERTION. TRANSFER SPENT FUEL TO CELL.



F. SHIP FUEL
REMOVE SHEILD PIECE AND DISCHARGE FUEL IN Na FILLED POT TO SHIPPING CASK.

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Figure 2.3.3. Refueling Sequence

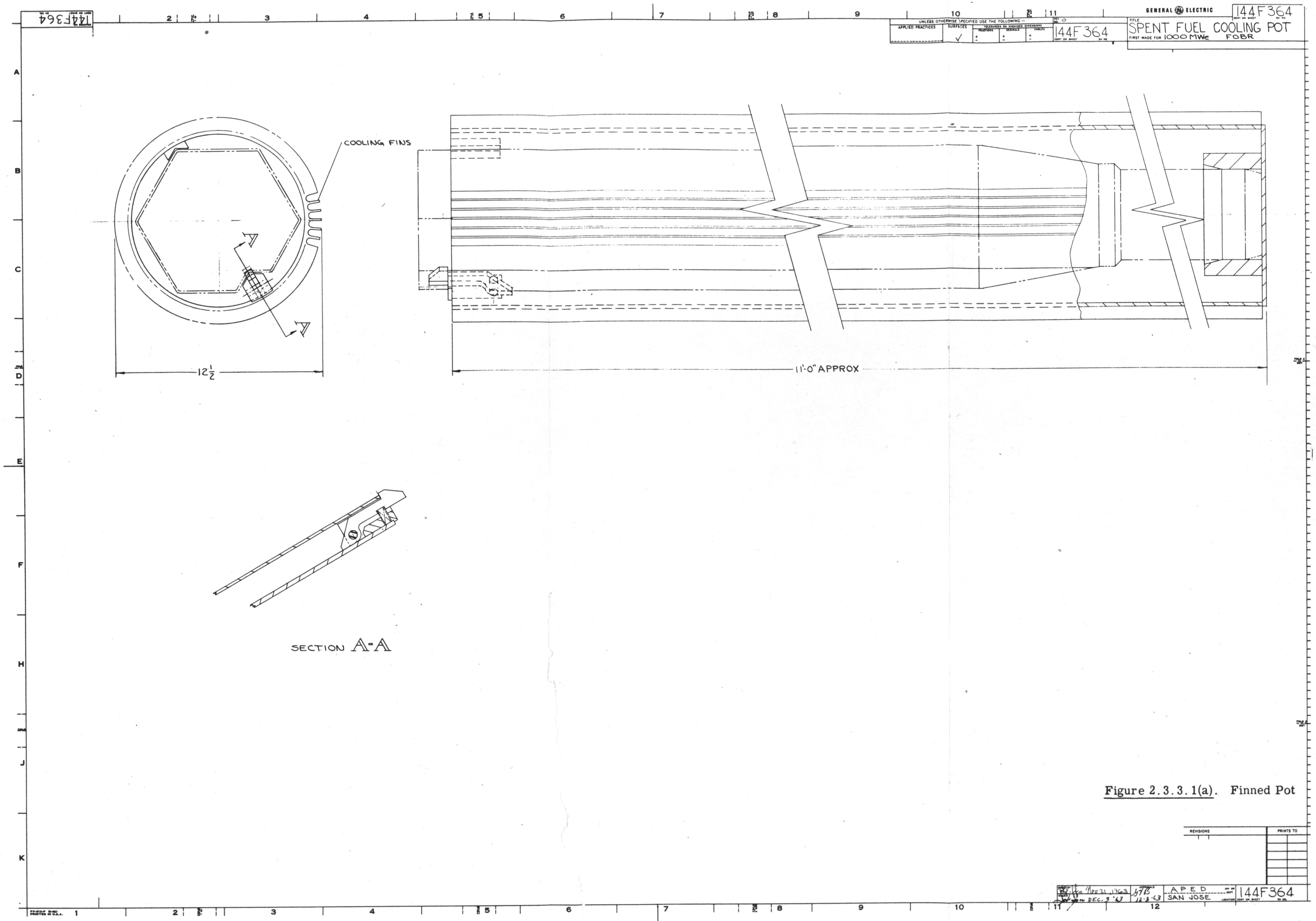

Figure 2.3.3.1(a). Finned Pot

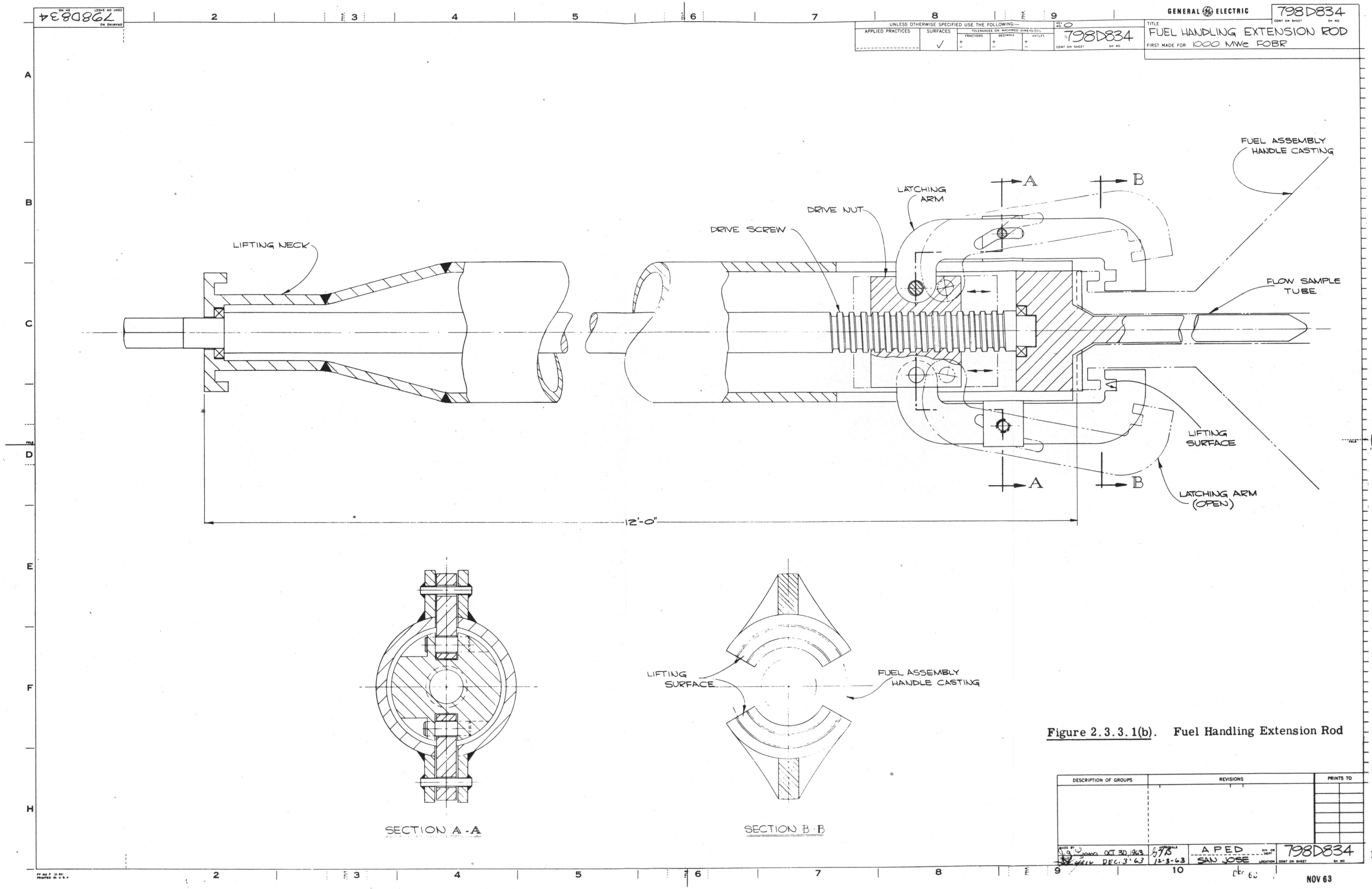

Figure 2.3.3.1(b). Fuel Handling Extension Rod

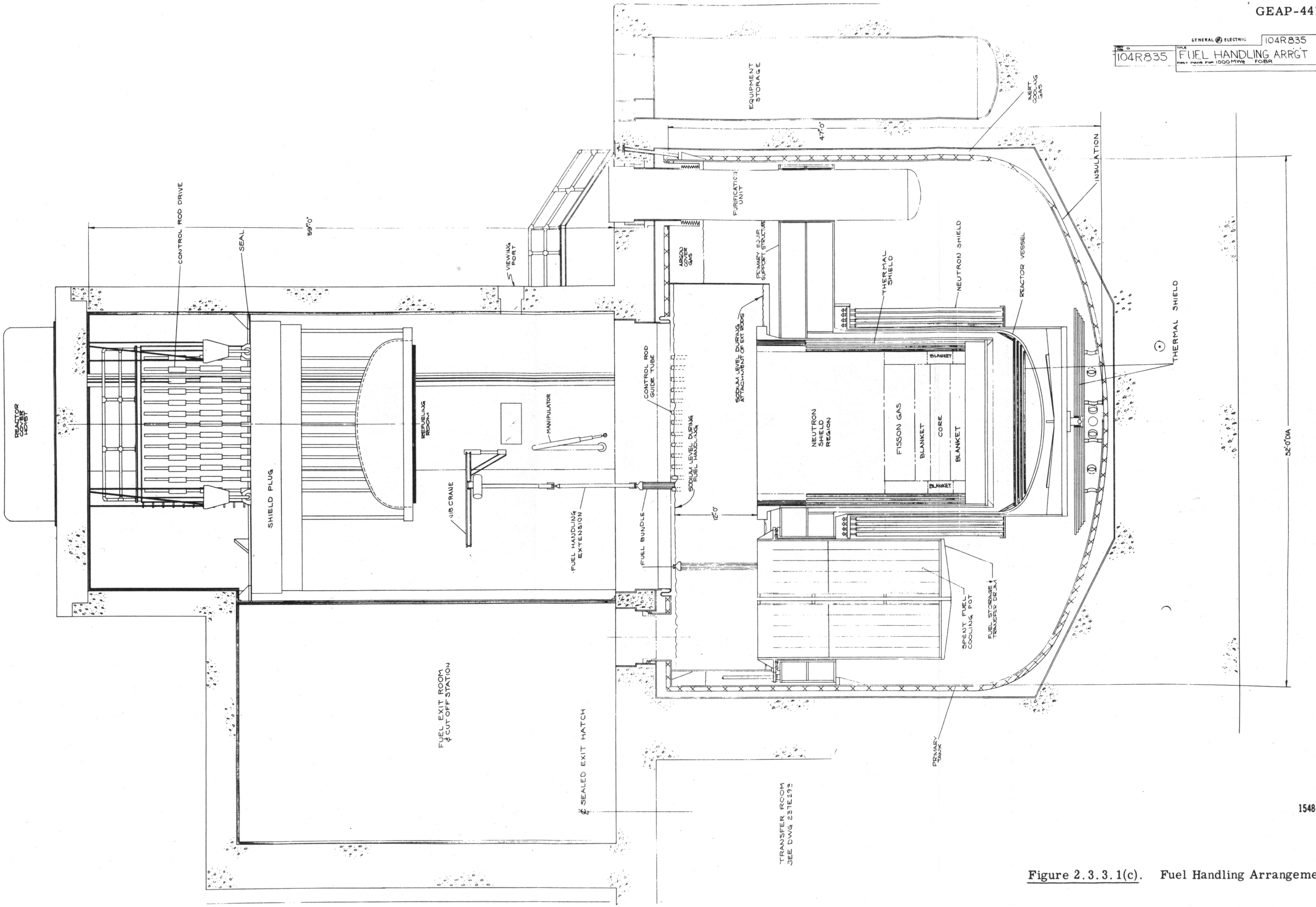

Figure 2.3.3.1(c). Fuel Handling Arrangement

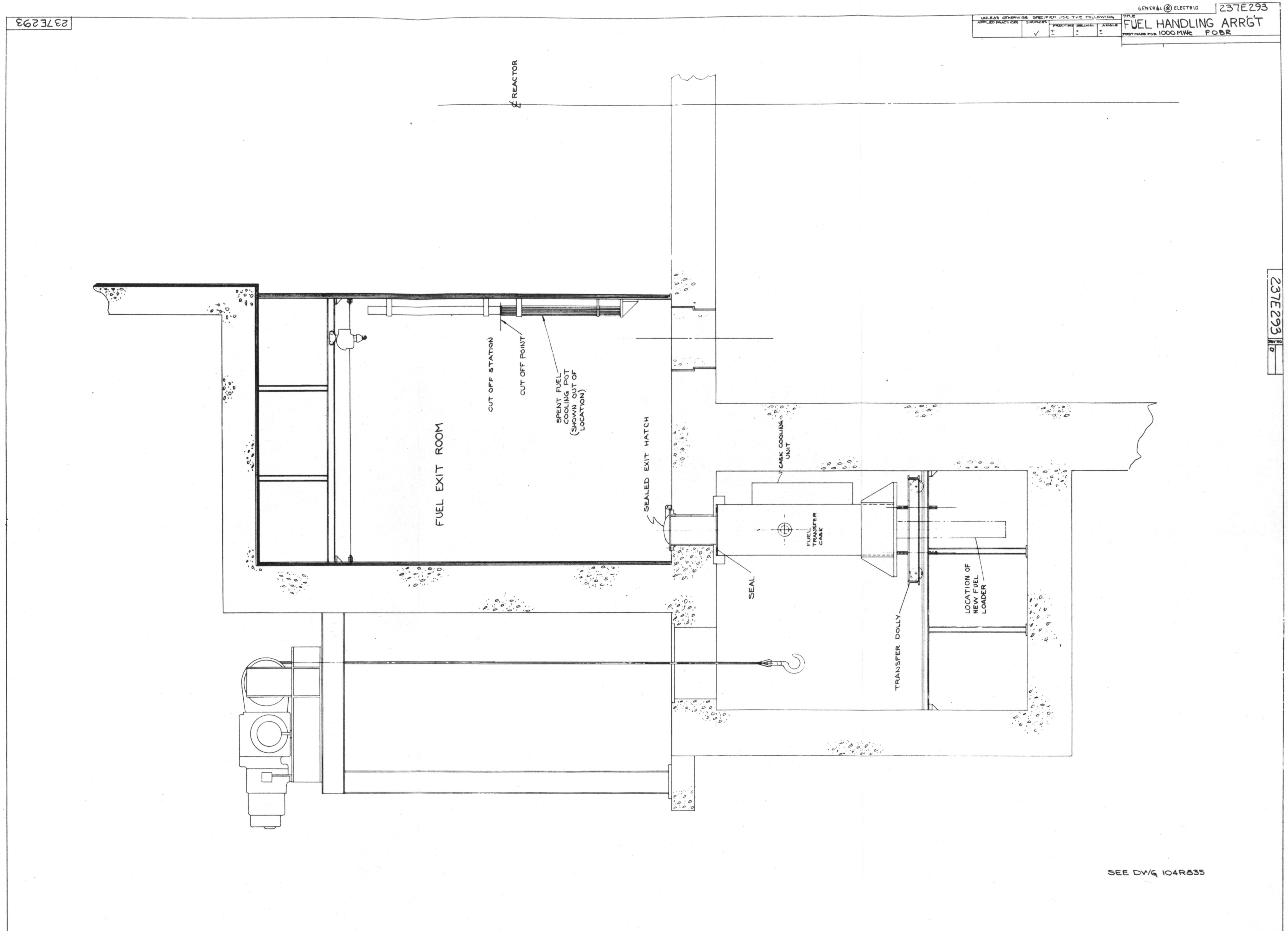

Figure 2.3.3.1(c) Sheet 2. Fuel Handling Arrangement

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2.3.3 FUEL HANDLING SYSTEM

2.3.3.2 Cell Description

The refueling cell is a gas-tight shielded structure capable of use as a hot cell. The cell consists primarily of a hermetically-sealed 1/2-inch-thick steel shell bolted to the floor and surrounded by high-density concrete blocks. The cell shape is octagonal; it measures 22 feet across flats and is 59 feet high inside. Its internal arrangement permits a walk way around the perimeter of the top shield plug on which the control drives are mounted. A platform above the control drives facilitates access to the individual drives through removable floor sections.

Thermal insulation between the steel liner and the blocks affords protection to the concrete against high internal temperature. The walls are about 3 feet thick with a 4-foot-thick ceiling. The ceiling is penetrated by the drive cover hoist cables and auxiliary emergency access plugs. The hoist is a drum type of 240 tons capacity, and it is mounted above the shield for accessibility. It is sealed by a metal cover to maintain inert atmosphere integrity within the cell.

Penetrations for equipment operating through the cell walls are made with sealed plugs.

Four lead glass windows are mounted in the walls. An easily-removed cover glass on the inside protects the windows from shock crazing induced by radiation exposure.

The majority of the floor space encompassed by the refueling cell is a removable shield plug approximately 20 feet in diameter, which can be raised up to the ceiling by an overhead hoist. In this location it seats against a seal preventing ingress of sodium vapor to the control drives, cable connections, etc. The jib crane housed in a recess in the cell wall is provided to perform the refueling lift operations. An extendable cable guide mounted on the wall guides the jib crane cable into the desired position. All operations within the refueling cell are observable by the operators through the use of periscopes and windows.

General purpose manual manipulators are located at strategic spots in the cell to perform versatile emergency functions, if they should become necessary. In addition, the cell contains an illumination system, a gas purging system, personnel access lock, a removable block section for large equipment transfer, heating and cooling systems to control the temperature, a communication system, and radiation monitoring devices. Additional plugs are available on all walls to allow insertion of special purpose handling aids or devices to assist in maintenance or repairs. All power-driven mechanisms in the cell have emergency manual take-over drives.

During reactor operation the refueling room is maintained sealed and at ambient temperature. Access to the refueling cell through a personnel lock is planned as one of the initial steps in the refueling operation to remove the vessel head bolts. The cell interior subsequently may be raised in temperature to ~400 F to minimize evaporation of sodium from the opened reactor cavity and condensation on surroundings. All equipment within the cell in thus designed to

2.3.3 FUEL HANDLING SYSTEM

2.3.3.2 Cell Description (Continued)

withstand ~ 400 F temperatures. The interior of the structure comes in direct communciation with the reactor coolant and potentially with fission product gases or alpha from any failed fuel present; thus, it is regarded as a hot area and is designed to prevent spread of contamination to other areas.

2.3.3.3 Fuel Transfer and Storage Equipment

The fuel transfer and storage drum is a 13-foot diameter rotating drum about 15 feet high that has provisions to store up to 84 fuel subassemblies in individual finned pots (see Figure 2.3.3.3). The pots are designed to transfer the fuel decay heat by convection to a heat dump system on the shipping cask and to insure that fuel is kept submerged in sodium at all times. The drum can be preloaded with the required number of fresh fuel assemblies contained within finned pots, and rotated to bring finned pots into position for the refueling operation. The drum rotation support bearing is located under the fuel transfer cell and is accessible for maintenance at reactor shutdown. The drive is by a ring gear and pinion on the drum periphery with an extension drive through the floor and shielding wall for direct operation and maintenance.

A sodium level control skirt surrounds the vessel top flange with its upper end attached to the shield plug lining and its lower end dipping into the sodium below the top level of the fuel rod handles (see Figure 2.3.3.1). This skirt allows a differential sodium level setting between the region over the reactor and the remainder of the main tank sodium. An extension of this skirt forms the transfer canal and thus permits movement of fuel between the storage drum and the region above the reactor core which submerged in sodium. Spent fuel may be left in the storage drum for up to a 4- or 5-month decay period if required.

A fuel transfer cell is an additional shielded and heated structure, which may be filled with inert gas, and provides a means of transferring the spent fuel from the storage drum to the shipping cask. It also has facilities permitting loading of the drum with fresh fuel, and for separation of the fuel and shielding extension sections of the fuel assembly for shipping purposes.

An overhead hoist is installed to permit the spent fuel bundle to be brought up into the transfer cell with its finned pot attached to its lower end. A hoist cable guide is located a few feet below the hoist for pulling the fuel assembly over to the disassembly station where separation of the fuel and shielding sections takes place. After disassembly, the fuel section, still inside its finned pot, may be lowered through an access port and directly into the shipping cask below. The shipping cask can be brought into a position where it can make a seal with the access port.

Interlocks are provided so that the cask must be sealed against the access port before it can be opened.

2.3 REACTOR SYSTEMS (Continued)

2.3.4 PRIMARY SODIUM COOLANT SYSTEM

The primary coolant flow of 94.5 million lb/hr of sodium is circulated to the reactor at 800 F, it flows up through the core and blanket regions at a maximum velocity of 13.6 ft/sec and discharges through six outlet ducts at a mean temperature of 1100 F. The six outlet ducts at the top end of the reactor vessel lead to six intermediate heat exchangers operating in parallel. Coolant discharge from the exchangers at approximately 800 F flows to the inlets to the six centrifugal pumps also operating in parallel, thus completing the circuit.

Pressures aroumd the primary coolant circuit are estimated (excluding static heads) at:

Pump discharge	55 psia
Core inlet plenum	50 psia
Core discharge	25 psia
Intermediate Heat Exchanger inlet	23 psia
Intermediate Heat Exchanger discharge and pump inlet	15 psia

2.3.4.1 Primary Coolant Pumps

The coolant is circulated throughout the primary system by vertically-mounted single-stage centrifugal pumps. Each of these pumps, illustrated on Figure 2.3.4.1 is driven by a variable-speed motor. The required flow capacity of each pump is approximately 37,500 gpm at 40 psig. The pumps are designed such that the shaft, impeller, discharge diffuser, and supporting members can be removed as an assembly attached to the shield plug (see Section 2.7.3, Reactor Maintenance ~~Primary Loop Component Removal Method~~). The length of this removable assembly including the shield plug is approximately 23 feet.

The pump casing, inlet diffuser, and discharge piping are attached to the primary system support structure. Any leakage from the pump seals returns directly to the primary tank.

2.3.4.2 Check Valves

A swing check valve is located in the discharge piping from each pump. This enables shutdown and isolation of any number of the primary pumps without interference to coolant circulation through the primary system supplied by the remaining operative pumps. The valve body is part of the piping structure whereas the valve access cover, check valve disk, and disk seat ring can be removed and replaced by access through the refueling room.

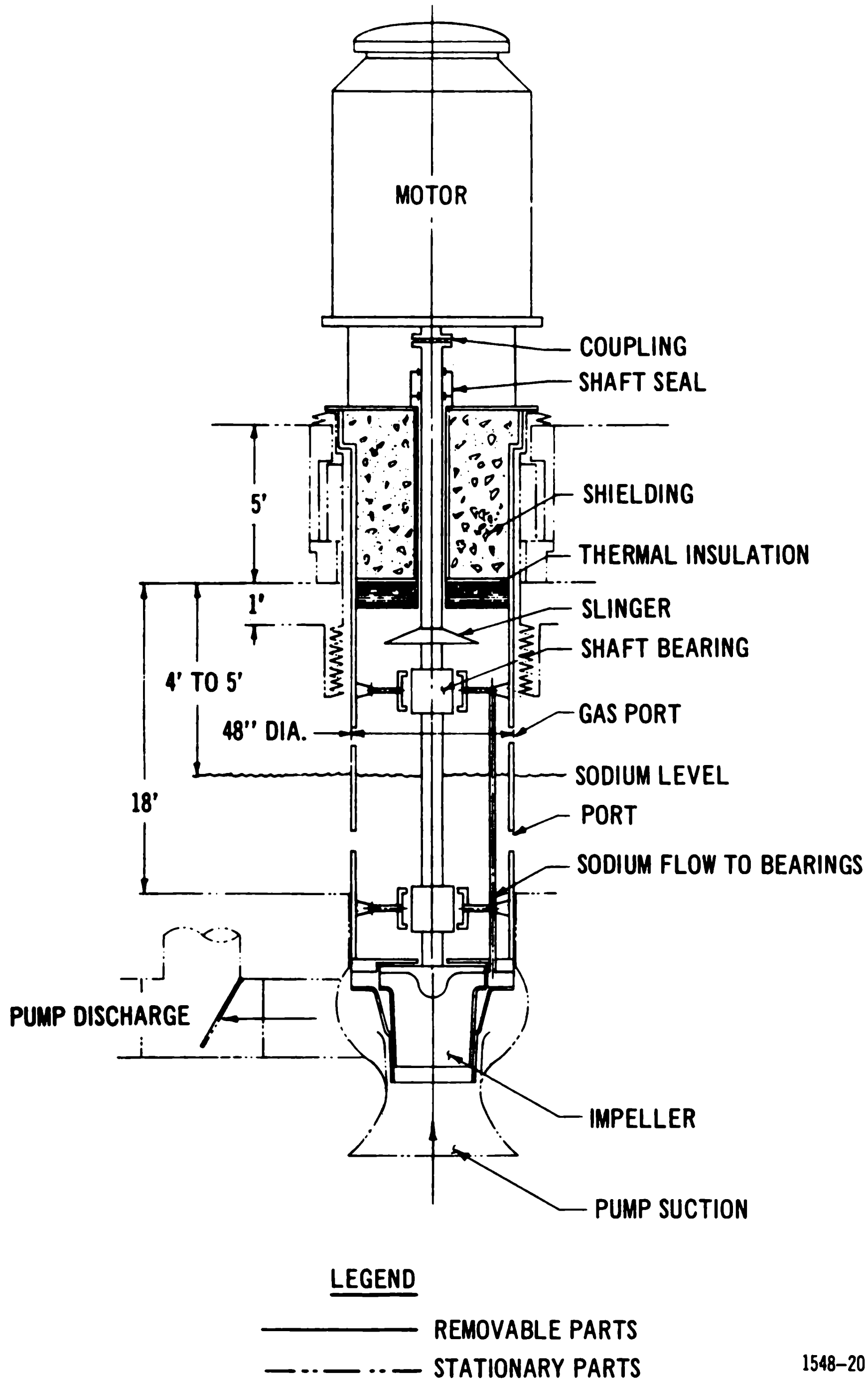


Figure 2.3.4.1. Primary Pump

2.3.4 PRIMARY SODIUM COOLANT SYSTEM (Continued)

2.3.4.3 Intermediate Heat Exchanger

The intermediate heat exchangers are a counterflow U-tube design. They are assembled, as illustrated on Figure 2.3.4.3, such that the tube sheets are located above the coolant level. Primary coolant flows on the shell side. The secondary sodium within the tubes will operate at a pressure not less than 5 psi above that of the primary sodium, thus ensuring that leakage (if any) will always be into the primary (active) circuit.

The shell and primary coolant connections are attached to the primary support structure but the tube bundle can be removed with the shield plug after both secondary coolant lines have been removed (see Section 2.3.4.5, Component Removal Method). The removable assembly, including the shield plug, is approximately 8-1/2 feet in diameter by 45 feet long.

TABLE 2.3.4.3

INTERMEDIATE HEAT EXCHANGER SPECIFICATIONS

Shell OD	8-1/2 feet
Over-all unit length	45 feet
Tube diameter	0.75 inch
wall thickness	0.042 inch
length	54 feet
Total number of tubes	1725
Total surface area	18,320 ft^2
Effective surface area	15,300 ft^2
Estimate over-all heat transfer coefficient	1500 $Btu/ft^2 hr\,^\circ F$

2.3.4.4 Sodium Purification Unit

The sodium purification unit is illustrated on Figure 2.3.4.4. It consists of a mechanical sodium pump, a heat economizer, a cold trap, and a plugging indicator. This equipment removes the reaction products resulting from leakage of air or water into the sodium system. Overflow devices are used in place of valves so when the pump is not operating, there is no communication of impurities in the cold trap with the purified tank sodium. This combination of components results in a complete unit which can be replaced in the same manner as the primary pumps and heat exchangers.

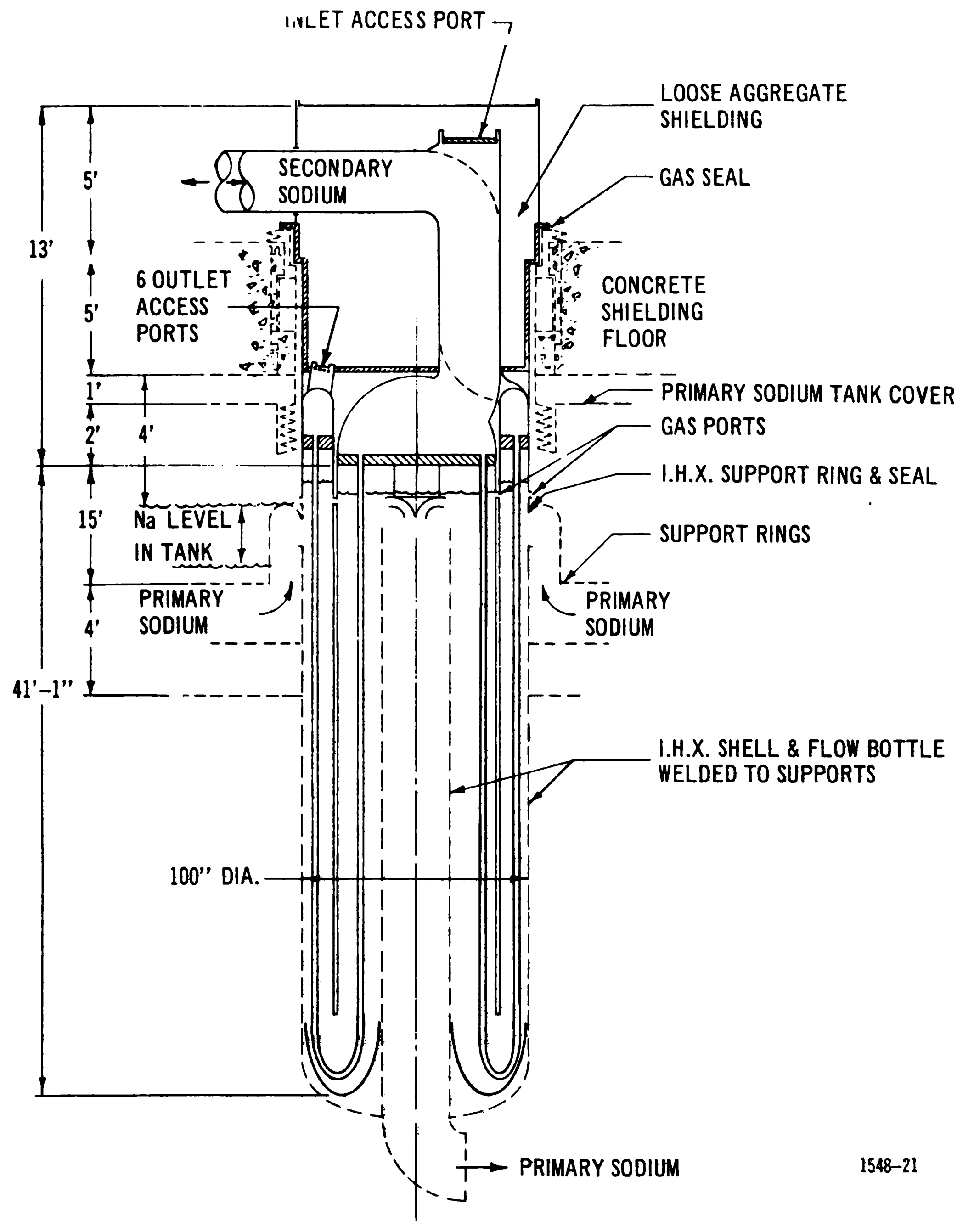


Figure 2.3.4.3. Primary Heat Exchanger

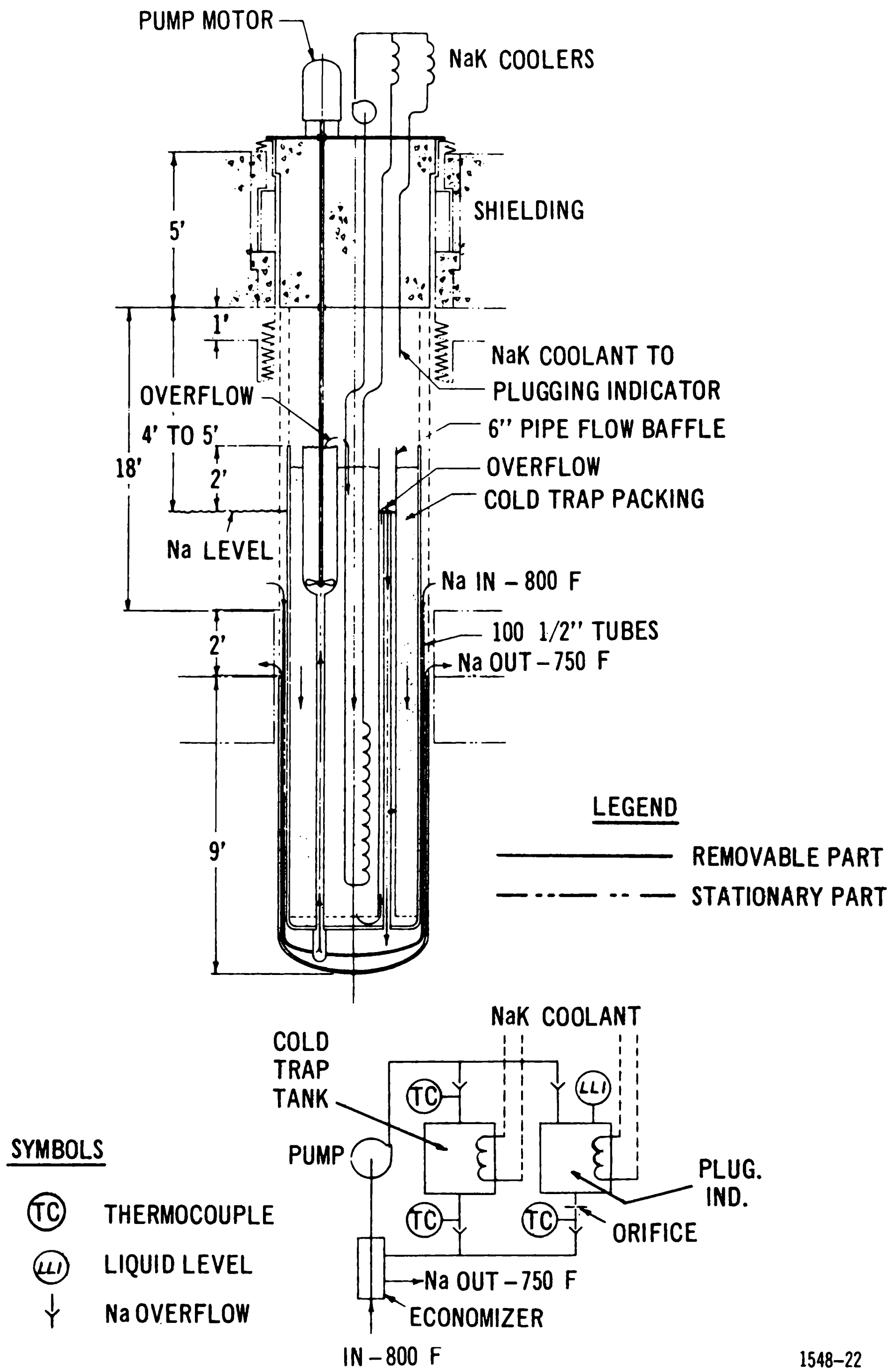


Figure 2.3.4.4. Sodium Purification Unit

2.3.4 PRIMARY SODIUM COOLANT SYSTEM (Continued)

2.3.4.5 Component Removal Method

Equipment design and component layout in the single-tank design dictates close integration of shielding with the components. Component removal is accomplished by pulling the shield plug that has the component attached, maintaining an inert atmosphere over the tank sodium using a bagging technique, and placing the used component in an available storage pit. The replacement part is then bagged into position using the reverse procedure (see Section 2.7 for further details).

2.3.4.6 Primary Inert Gas System

The inert-gas system is shown in Figure 2.3.4.6. The argon gas over the primary tank is vented to gas expansion and equipment storage tanks resulting in a net pressure change of less than 3 psi for a sodium temperature change (and expansion) from 300 F to operating condition. This eliminates the need for close pressure control and results in minimum gas consumption while utilizing a noncirculatory system. The major functions of the system are to: 1.) provide inert atmosphere for the primary sodium, and 2.) control the sodium level over the reactor during refueling (see Section 2.3.3, Refueling Scheme).

2.3.4.7 Inert (Nitrogen) Cover Gas System

Nitrogen (1 - 3 percent oxygen) is circulated in the space between the Primary Tank insulation and the concrete shield (see Figure 2.1, Reactor Building Elevation). It has three purposes:

1. Provide an intermediate barrier between the radioactive system and the personnel area so that any leakage (air, sodium, or inert gas) can be detected.

2. Provide the heat removal system for the concrete shielding which is heated by Na^{24} gammas and by thermal losses from the tank.

3. Provide fire prevention in the event of a primary sodium leak, but normally the nitrogen does not contact either primary or secondary sodium.

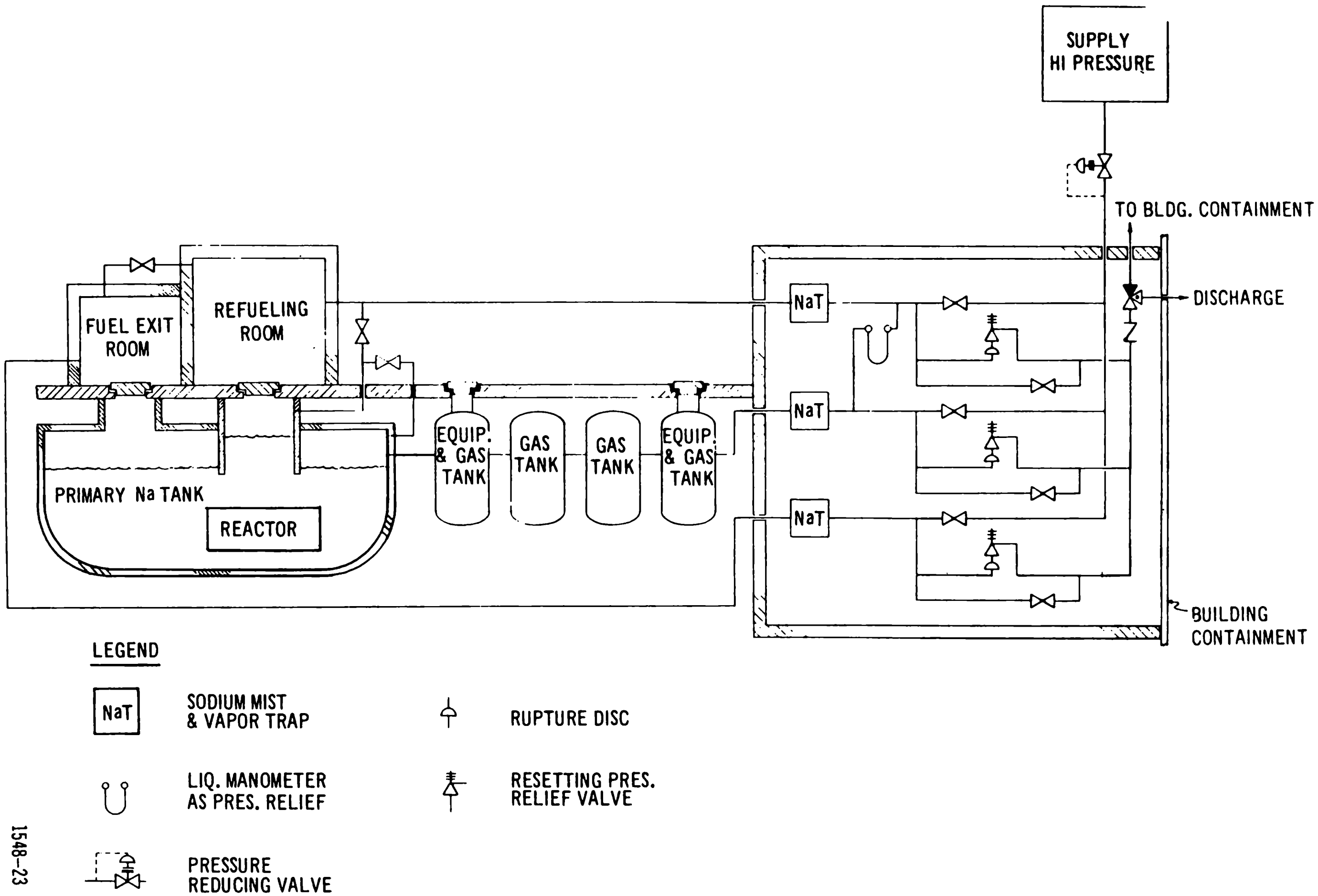


Figure 2.3.4.6. Argon Gas Flow Diagram

2.4 STEAM GENERATION AND POWER PLANT

2.4.1 INTERMEDIATE CIRCUIT AND STEAM PLANT

2.4.1.1 Steam Generator Units

2.4.1.2 Reheater Units

2.4.2 PLANT HEAT BALANCE

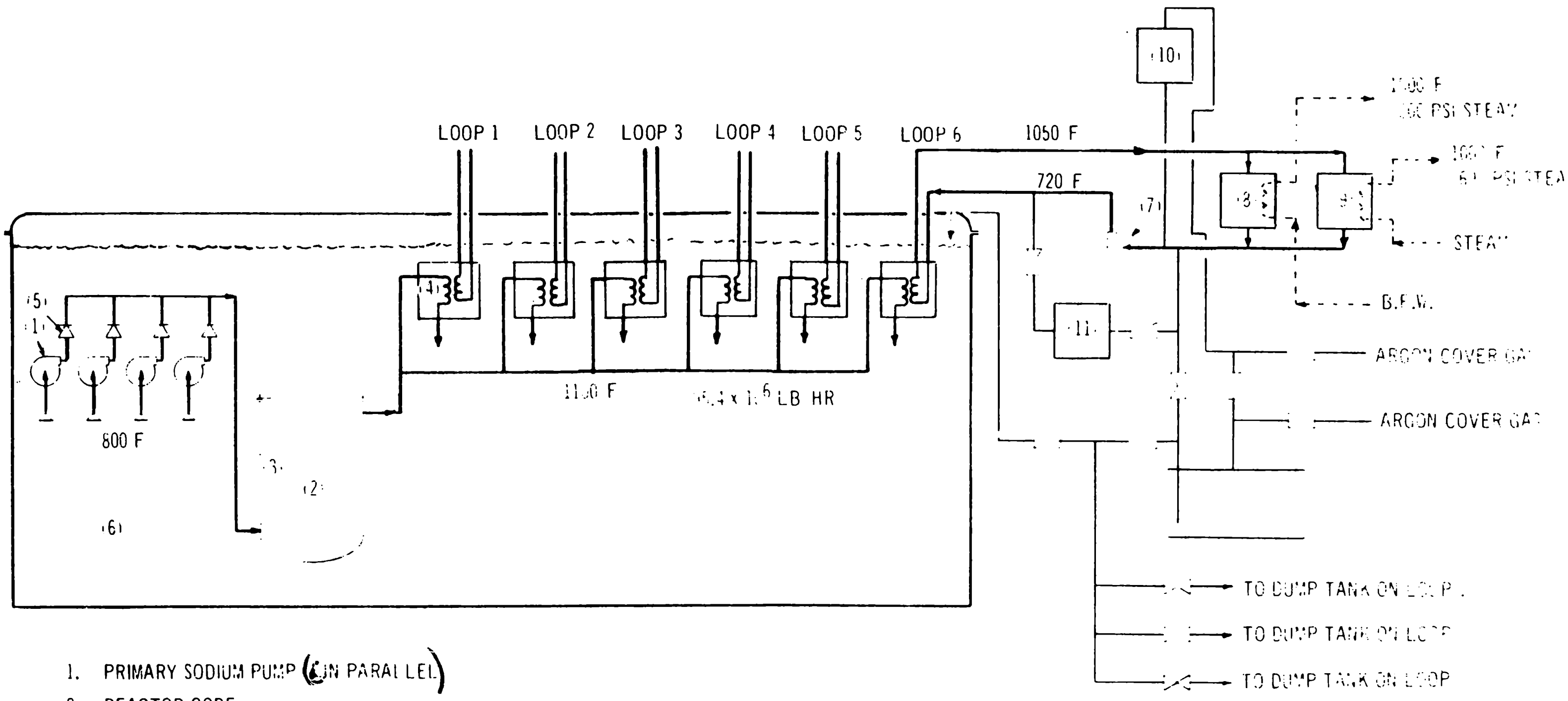
2.4.3 PLANT ARRANGEMENT

2.4 STEAM GENERATION AND POWER PLANT

Nonactive sodium from the intermediate heat exchangers within the primary tank is piped to the steam generation plant area. This area, as with the reactor and systems area, is designed with all sodium equipment located below the operating flow submerged in an inert (nitrogen) atmosphere.

Steam is raised at 3500 psia, 1000 F in six parallel units and flows to a single cross-compound reheat turbine unit.

The design data presented in this section for the steam generation and power plant areas of the reference plant is preliminary and is intended solely to give an indication of the plant capability, scale of equipment, and possible arrangement until such time as more detailed information becomes available.



1. PRIMARY SODIUM PUMP (4 IN PARALLEL)
2. REACTOR CORE
3. REACTOR BLANKET
4. INTERMEDIATE HEAT EXCHANGER (6 IN PARALLEL)
5. PRIMARY SODIUM CHECK VALVE
6. PRIMARY SODIUM DUMP TANK WITH SUBMERGED PURIFICATION UNIT

~~FOUR~~ SIX SECONDARY SODIUM LOOPS – INDEPENDANT EXCEPT FOR CROSSOVER ON THE DRAINS TO THE DUMP TANK

7. SECONDARY SODIUM PUMPS
8. ONCE THRU STEAM GENERATOR
9. STEAM REHEATER
10. SODIUM LEVEL EXPANSION TANK & HEAD TANK
11. SODIUM REPURIFICATION UNIT
12. SECONDARY SODIUM DUMP TANK

MAIN PIPING
SECONDARY PIPING
STEAM LINE

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Figure 2.4. Sodium Flow Diagram

2.4 STEAM GENERATION AND POWER PLANT (Continued)

2.4.1 INTERMEDIATE CIRCUIT AND STEAM PLANT

The flow circuit for the intermediate sodium system and steam raising plant is illustrated in Figure 2.4. Sodium is heated in the six parallel intermediate heat exchangers to a temperature of 1050 F and then flows in part (83 percent) to the steam raising plant where steam at 3500 psi, 1000 F is generated; the remainder flows directly to two reheater units also located in the steam turbine generator area. Sodium is returned at a temperature of approximately 720 F from the two reheaters and six steam raising units (all in parallel) to the centrifugal pumps for the secondary circuit, and thence to the intermediate heat exchangers. Figure 2.4.1 illustrates the temperature distributions existing within the intermediate heat exchangers, steam generators and reheat units at the full-power operating condition.

All loops are cross-connected and designed to operate with appreciable intermixing between parallel flow steams.

The intermediate sodium loop pressures are estimated to be approximately (static heads excluded):

Pump discharge	40 psia
Intermediate exchanger inlet	35 psia
Intermediate exchanger exit	38 psia
Steam generator inlet	26 psia
Steam generator exit	23 psia
Reheater inlet	24 psi
Reheater exit	20 psi
Pump inlet	15 psi

2.4.1.1 Steam Generator Unit

The reference steam generator design is a single-wall tube, once-through design based on concepts presently being studied under other AEC contracts. It is anticipated that a maximum sodium temperature of 1050 F and a steam temperature of 1000 F will permit the design of a unit using Croloy 2-1/4 throughout. An operating pressure of 3700 psia is estimated to require tubes 3/4-inch OD having a wall thickness ranging between 0.1 and 0.15 inch from feedwater inlet to superheated steam exit locations. A maximum temperature difference of 330 F between inlet and exit sodium and of 220 F between sodium coolant and steam at any point in the exchanger at design condition is intended to minimize potential thermal stress problems within the system.

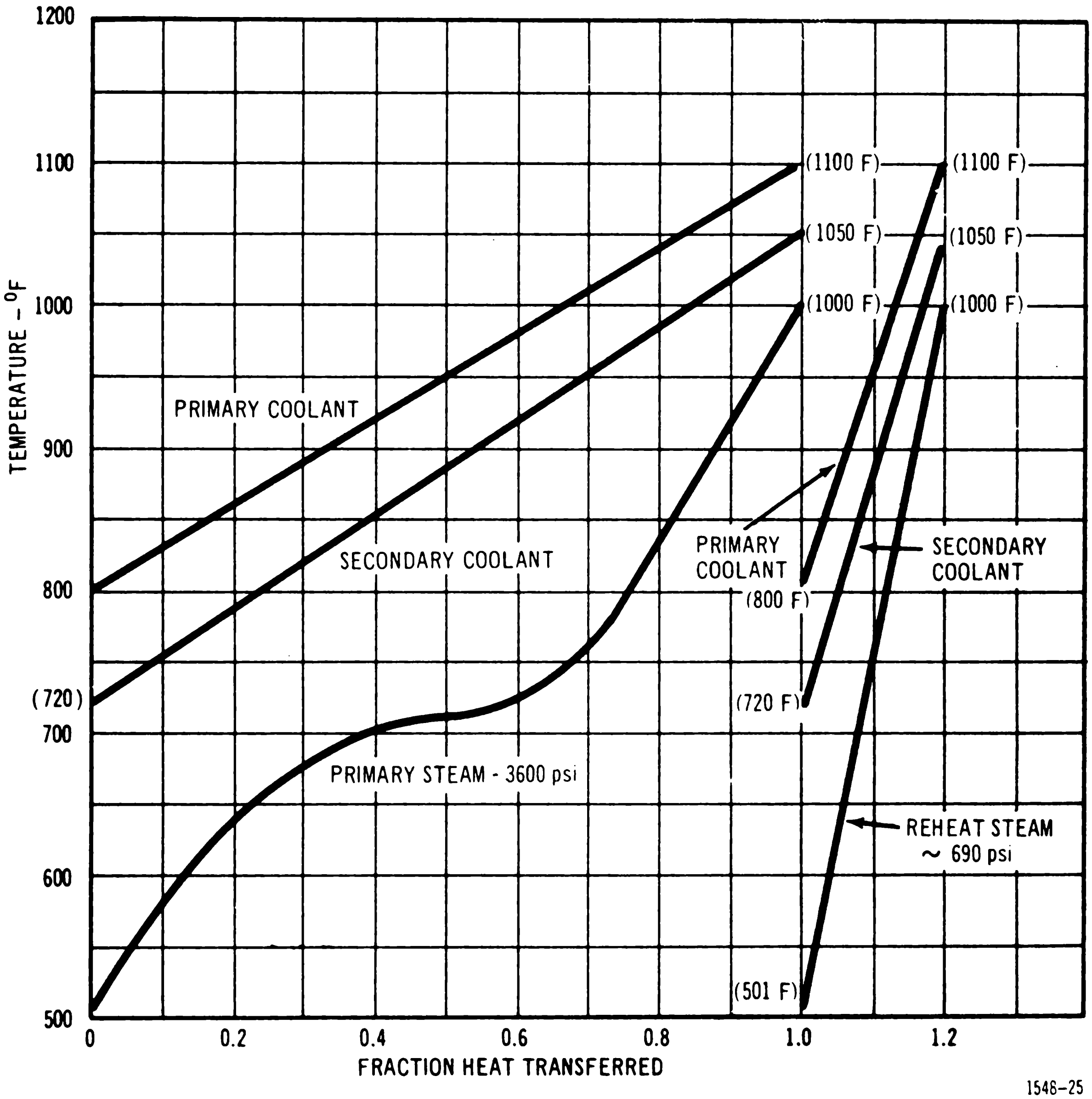

Figure 2.4.1. Thermal Diagram for Heat Exchanger Equipment

2.4.1 INTERMEDIATE CIRCUIT AND STEAM PLANT (Continued)

2.4.1.1 Steam Generator Unit (Continued)

Figure 2.4.1.1 shows the full-power (2500-MW) operating conditions for the plant. Part-load operation requires use of a bypass valve and reduced flow in primary and secondary loops (see Section 2.6). In this way, constant steam conditions and reactor coolant inlet temperature can be maintained over the full operating power range.

2.4.1.2 Reheater Units

The reference reheater units (two in number) are single-walled tube design constructed largely of Type 316 stainless steel and similar in concept to designs presently under development under other AEC contracts. The units have steam flow on the tube side of the exchanger.

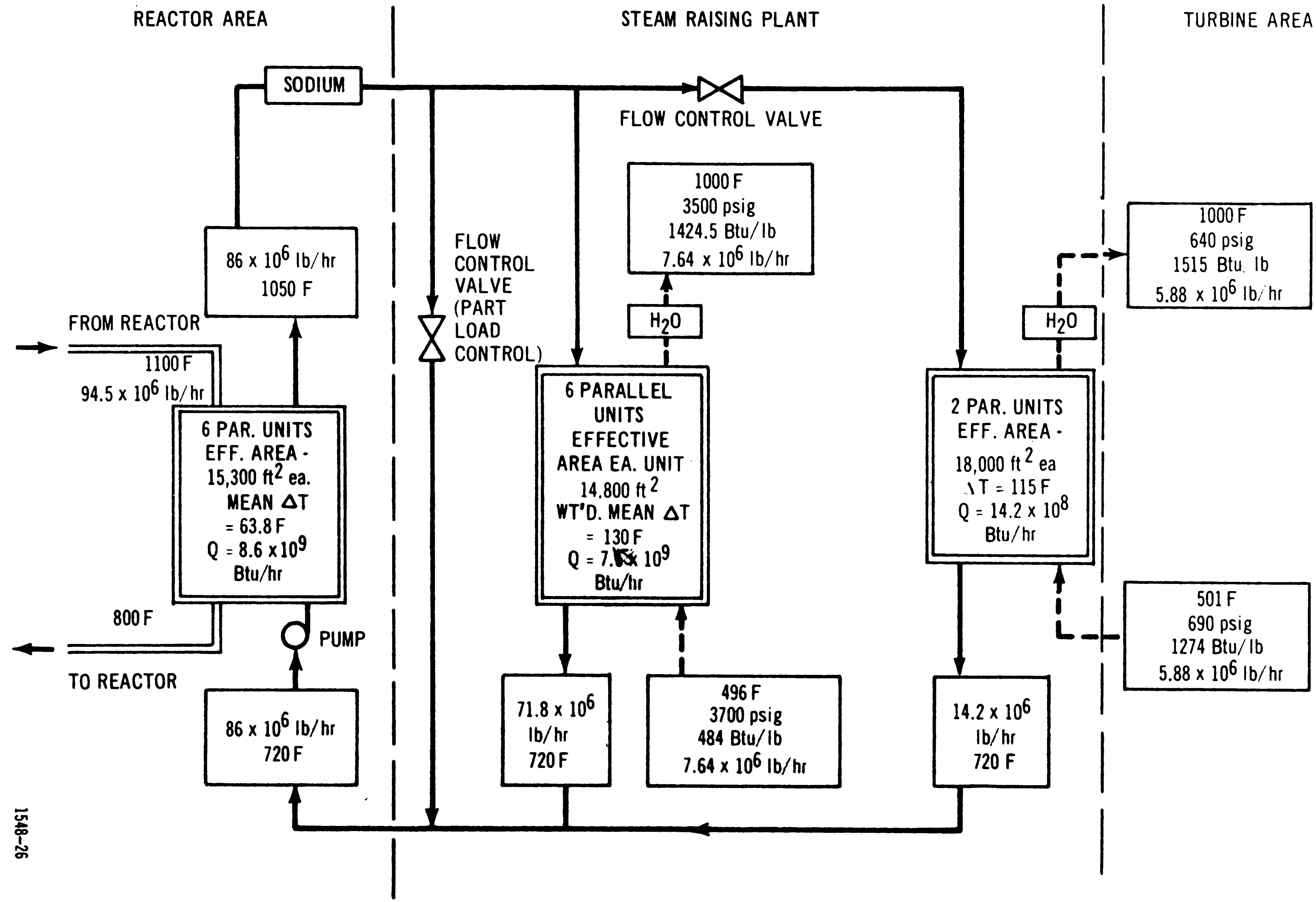

Figure 2.4.1.1. Intermediate Coolant and Steam Flow Circuits

STEAM GENERATION AND POWER PLANT (Continued)

2.4.2 PLANT HEAT BALANCE

Figure 2.4.2 illustrates a turbine heat balance for a 3500-psi 1000°, 1000° tandem compound unit exhausting at 1-1/2 inches mercury, generally similar* to that proposed for the reference plant. Assuming the 7570 Btu/Kwhr turbine heat rate, the thermal output of 2500 MWt will yield a gross station output of $\frac{2500 \times 3413 \times 10^3}{7570} = 1127$ MW. Station power requirements are estimated at:

Sodium Pumping Power	Primary	6.8 MW
	Secondary	4.2 MW
	Total	11.0 MW
Plant Auxiliaries Load (approximate)		15.0 MW
Total Station Load		26.0 MW

The estimated net station output may thus approach 1100 MWe corresponding to a net efficiency of 44 percent.

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*The turbine heat balance net heat rate, 7570 (Figure ~~2.6.3)~~ 2.4.2), is for a unit having a slightly-lower steam flow than that of the reference plant (7.0 vs. 7.6 × 10^6 lb/hr). Furthermore, it includes separate turbine driven generators to provide up to 30 MW of power at variable frequency for coolant circulator variable-speed motor drives, a feature probably not required for a sodium-cooled reactor.

2.4 STEAM GENERATION AND POWER PLANT (Continued)

2.4.3 PLANT ARRANGEMENT

An over-all arrangement for the reference plant is shown in Figure 2.4.3(a). An elevation of the steam raising and power plant areas can be seen in Figure 2.4.3(b); elevations of the reactor area are shown separately on Figures 2.1 and 2.3.3.1(c) and a plan view on Figure 2.3.

The reference plant layout provides low pressure containment of the reactor area through use of a low-leakage reactor building maintained a few inches water gage below ambient pressure by the ventilation system which exhausts via chemically-treated scrubbers and filters to the stack.

The computed energy release from the maximum hypothetical accident (see Section 2.6) will, due to its small magnitude of approximately 200 pounds of TNT, be contained within the primary tank and associated shield structure without major disruption. Major sodium fires are prevented by use of an inert atmosphere in all areas below the operating floor level where the primary sodium systems are located.

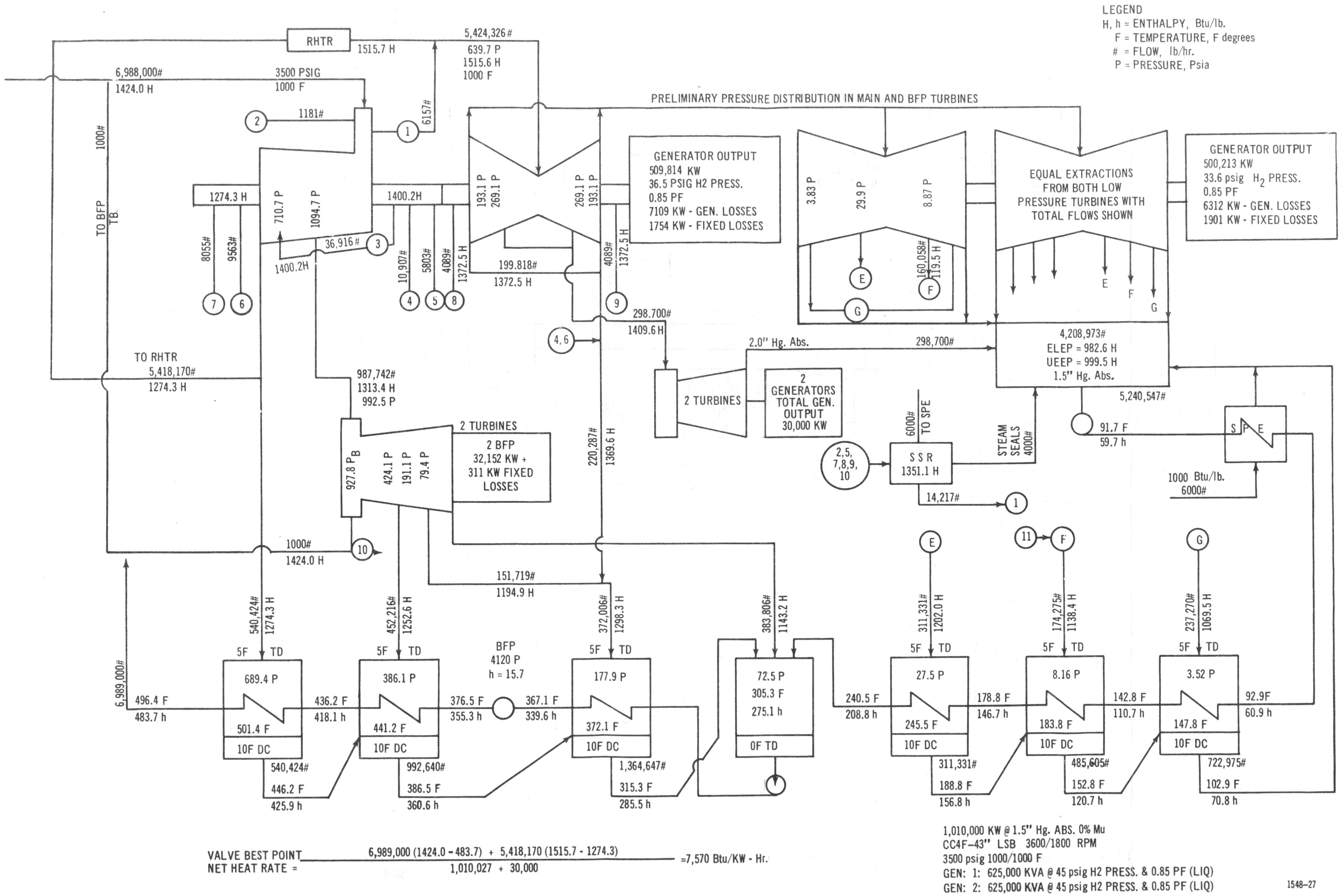

Figure 2.4.2. Representative Turbine Heat Balance

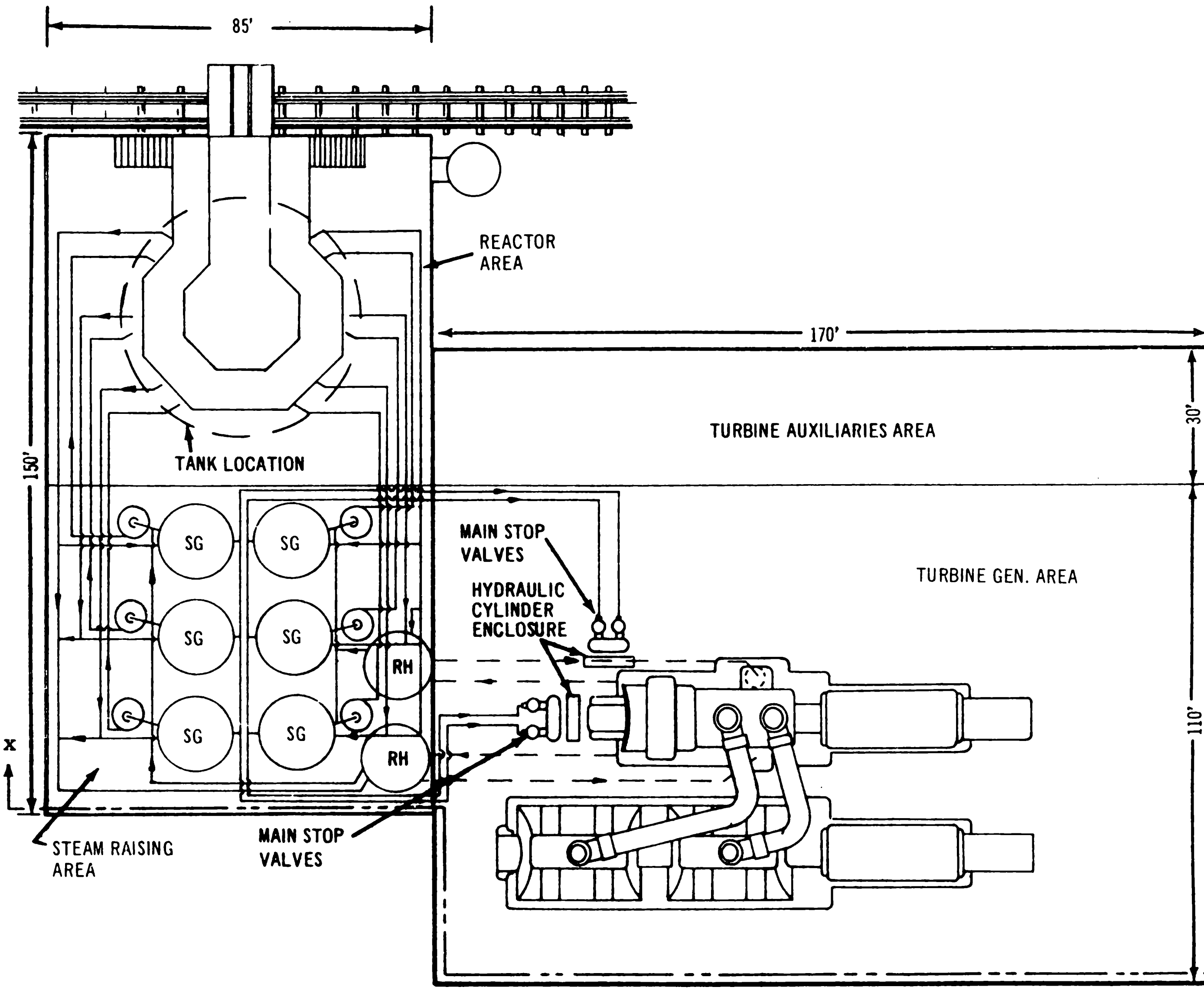

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Figure 2.4.3(a). Plant Layout

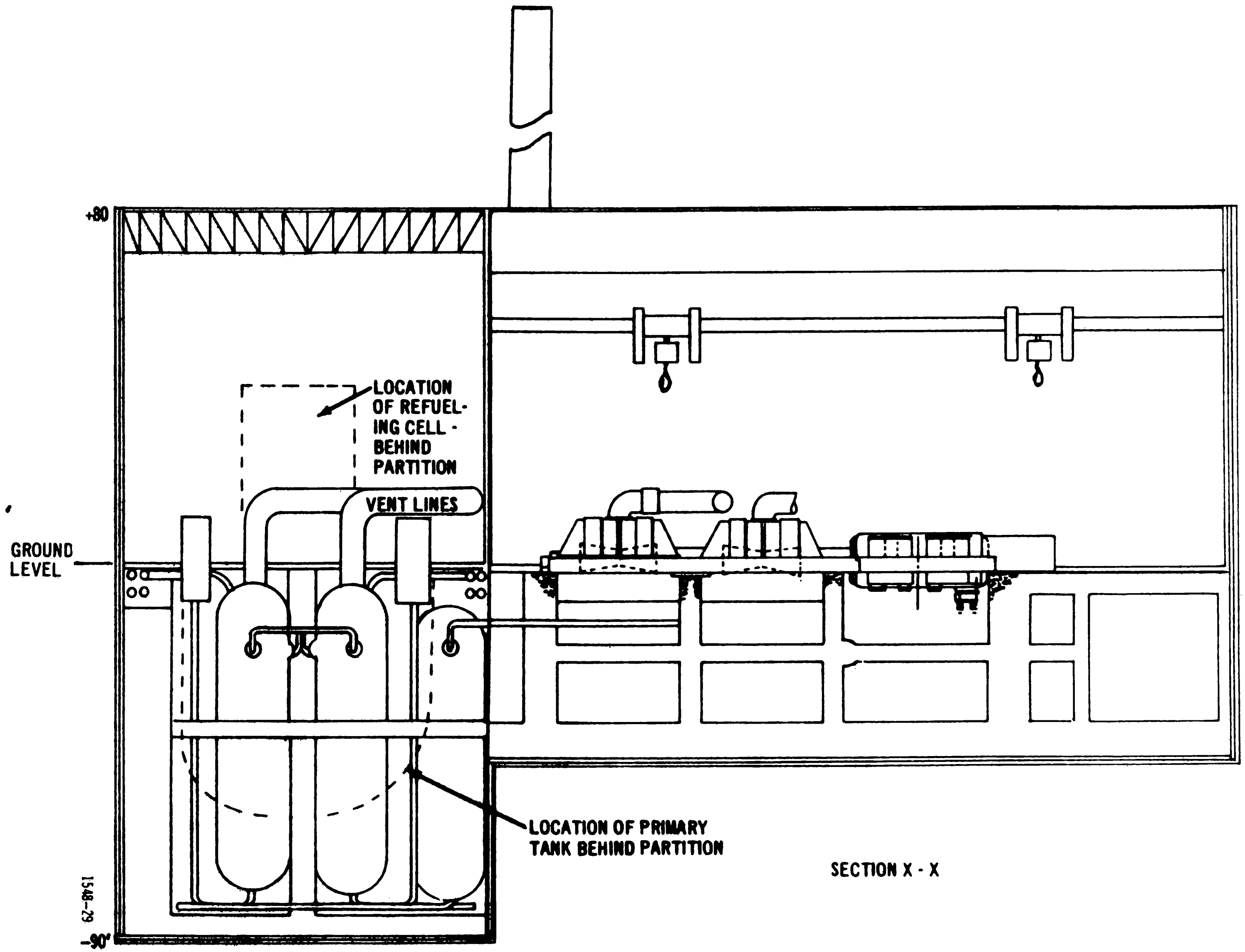


Figure 2.4.3(b). Plant Elevation

2.5 REACTOR PHYSICS DATA

2.5.1 FUEL REQUIREMENTS AND REFUELING PATTERN

2.5.1.1 Regional Fuel Compositions
2.5.1.2 Neutron Balance Data and Breeding Ratio
2.5.1.3 Fuel Lifetime

2.5.2 POWER AND FLUX DISTRIBUTIONS

2.5.2.1 Spectral Flux and Power Distributions
2.5.2.2 Spatial Power Distributions

2.5.3 CONTROL REQUIREMENTS AND ASSEMBLY WORTHS

2.5.3.1 Total Requirements
2.5.3.2 Type and Distribution of Rods in Core
2.5.3.3 Maximum Rod and Fuel Assembly Worths

2.5.4 REACTIVITY COEFFICIENTS AND KINETICS PARAMETERS

2.5.4.1 Doppler Effect
2.5.4.2 Reactivity Effects Due to Sodium Loss
2.5.4.3 Other Reactivity Effects
2.5.4.4 Summary of Temperatures and Power Coefficients
2.5.4.5 Neutron Lifetime and Effective Delayed Neutron Fraction

2.5.5 STARTUP AND TRANSITION TO EQUILIBRIUM CYCLE

2.5 REACTOR PHYSICS DATA

The reactor design as described in Section 2.3 has a carefully selected soft neutron spectrum achieved through inclusion within the core of ~~8.3~~ 7.7 volume percent BeO; 50 percent of the fissions are produced by neutrons of energy below 125 Kev, and 25 percent by neutrons below 10 Kev. The spectrum yields a large negative Doppler temperature coefficient, $-0.01\ T\frac{dk}{dT}$, considered at this time to be desirable from safety considerations (see Section 3.2).

The presence of BeO within the core also reduces the influence of sodium on the neutron spectrum, (both are light scattering materials), thereby appreciably lessening the maximum reactivity increment due to loss of sodium in the central portion of the core. This effect, combined with the selection of a pancake shaped core, limits the maximum reactivity increment due to central sodium voidage to 1.36 percent; total loss of sodium from core and blankets produces essentially zero reactivity; and sodium loss in the top portion of the core and its associated axial blanket produces negative reactivity. The sodium power coefficient is negligible.

The softened spectrum of the reference design reduced breeding ratio compared to some of the more advanced designs shown later. Nevertheless, the breeding ratio (1.25) exceeds the minimum value specified as an objective of the study.

Radial power flattening is achieved by varying the fissile plutonium content of the supplied fuel. A three-zone core is utilized for this purpose. Axial power flattening is inherent in the pancaked core with appreciable fissile plutonium buildup in the axial blankets. The over-all peak-to-average core power density, excluding local peaking factors, is $1.23 \times 1.14 = 1.41$.

Average Pu (239 + 241) content in the fresh fuel supplied to the core at the beginning of an equilibrium fuel cycle is 18.0 atom percent. The average number of core fuel assemblies replaced at the end of each refueling cycle is about 22 percent. This percentage varies from zone to zone (in accordance with the radial power distribution) to yield about 100,000 MWD/T burnup for the discharged assemblies of each zone.

Fuel assembly dimensions are based primarily on mechanical considerations. The reactivity worth of the central fuel assembly (60 cents) is well below one dollar ($\beta_{eff} = 0.0033$). The control rod size is also based primarily on mechanical considerations, with the spacing and number of rods selected to accommodate the 4.8 percent reactivity swing from the cold condition at the beginning of a refueling cycle to the full power operating condition at the end of the refueling cycle, plus 1.5 percent safety and shutdown margin. Maximum control rod worth for the central rod in a highly reactive configuration is also well below one dollar (~48 cents).

The accuracy of the physics calculations is limited by the present status of fast reactor cross-section data and, probably to a lesser extent, by inadequacy of current computational techniques. These deficiencies of data and methods are discussed in Section 3.6, which describes the computational methods used to obtain the physics data given in this section.

2.5 REACTOR PHYSICS DATA (Continued)

The results summarized above are presented in greater detail in Sections 2.5.1 through 2.5.4, starting with a description of the equilibrium fuel cycle, including isotopic fuel compositions for fresh and discharged fuel and at various stages in the fuel cycle, neutron balance data, and breeding ratio. This is followed by a discussion of the power distributions of the equilibrium cycle, including local power-peaking effects due to control rods and differences in the irradiation levels of fuel assemblies within the same core zone. Spectral power, flux, and adjoint flux distributions are also presented. Control requirements and reactivity worths of control rods and fuel assemblies are discussed following the data on power distributions. The reactor physics data concludes with a discussion of the reactivity changes due to the Doppler effect, sodium voidage, and other reactivity coefficients and kinetics parameters. Section 2.5.5 then follows with a qualitative discussion of the startup reactor and its transition to an equilibrium cycle.

2.5 REACTOR PHYSICS DATA (Continued)

2.5.1 FUEL REQUIREMENTS AND REFUELING PATTERN

A three-zone enrichment* core design for the 1000 MWe fast breeder was specified to flatten the power distribution within the core. The design results in a requirement to refuel approximately 22 percent of the core following 6.8 months of operation on the equilibrium cycle (~5 months full power operation).

The core and blanket arrangement of the system is shown in Figure 2.5.1(a). The scheduled fractional fuel replacements at each shutdown are also shown on the figure. A detailed schematic of the upper right quadrant of the system, which was used as the computational model (see Section 3.6), is shown in Figure 2.5.1(b).

The symbolism employed in Figure 2.5.1(b) will be used frequently in this section and in Section 3.6. Hence, it is desirable to tabulate the symbols together with their description.

Z1	Interior one-third volume of the core
Z2	Middle one-third volume of the core
Z3	Outer one-third volume of the core
RB1	Inner annulus of radial blanket
RB2	Outer annulus of radial blanket
AB1, n	First axial blanket region above core zone n (n = 1, 2, 3)
AB2, n	Second axial blanket region
AB3, n	Third axial blanket region
AB4, n	Fourth axial blanket region
AB5, n	Fifth axial blanket region
AB6, n	Sixth axial blanket region (furthest from core)
CRB	Corner radial blanket (extending above and below the ~~core)~~ RB1 and RB2 regions),

*The term "enrichment" is used here as an abbreviation to denote the content of fissionable isotopes in the fuel.

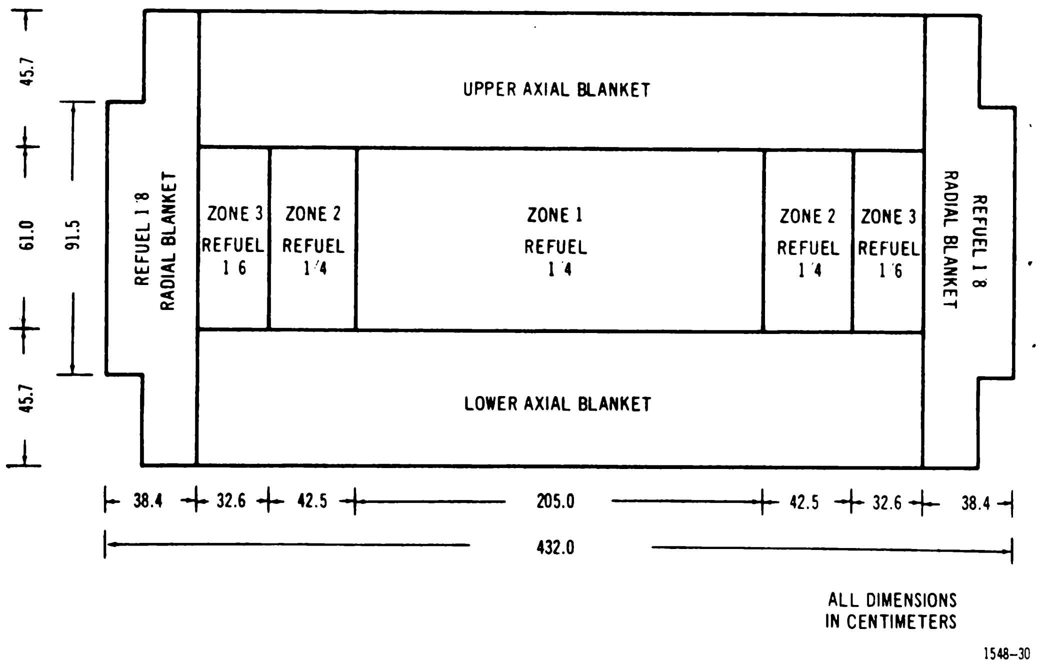


Figure 2.5.1(a). Assumed Core and Blanket Arrangement

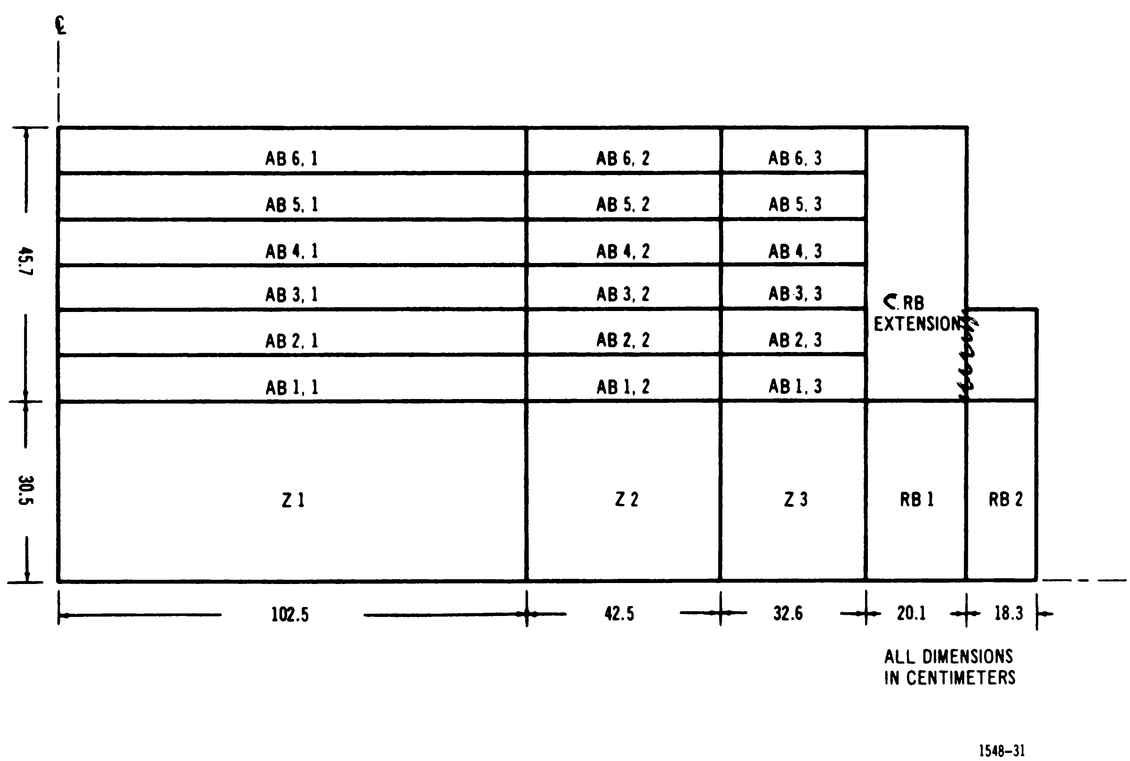


Figure 2.5.1(b). Blanket Region Locations (for Computation Purposes)

2.5 REACTOR PHYSICS DATA (Continued)

2.5.1 FUEL REQUIREMENTS AND REFUELING PATTERN (Continued)

The reason for subdividing the axial blanket into six thin (3-inch thick) regions is to provide detailed information on the generation of fissile material within the axial blanket and to give a more accurate description of neutronic processes near the core-blanket interface.

The axial blanket fuel assemblies are to be replaced on the same schedule as their associated core fuel assemblies and form an integral part of them. Two zones of radial blanket fuel assemblies are planned. For the present fuel cycle studies, blanket fuel assemblies in both zones were replaced after 4.6 years. (The outer zone might remain in place for the lifetime of the reactor in an optimized fuel cycle.)

2.5.1.1 Regional Fuel Compositions

The fuel isotopic compositions for the various regions of the core and blankets as they are to be supplied by the manufacturer and presented in Table 2.5.1.1(a). The discharged isotopic compositions for the three core zones, the portion of the axial blanket above and below zone 1, and the portion of radial blanket opposite the core are presented in Table 2.5.1.1(b). Tables 2.5.1.1(c), (d), and (e) list the average regional fuel compositions at the beginning, middle, and end of the 6.8 month burnup interval, respectively, for these core and blanket regions.

Weights of the fissile isotopes supplied and discharged for each region of core and blanket, including the corner extensions of the radial blanket, are listed in Figure 2.5.1.2. The system is assumed to be on an equilibrium fuel cycle.

The relative compositions of the three core zones, Table 2.5.1.1(a), were evolved from an initial study of a three-zone out-in fuel shuffling scheme that showed a good power distribution. This evolution is discussed fully in Section 3.6. The isotopic compositions in Tables 2.5.1.1(a) through (e) are consistent with the following assumptions:

1. Unit cell designs for all the core zones are the same and are as specified in Table 2.1.

2. All core fuel is irradiated at a fixed location to approximately 100,000 MWD/T.*

3. Core fuel is recycled using blanket plutonium as makeup.

4. The fresh blanket fuel is nearly depleted uranium (U-238 with 0.3 atom percent U-235).

5. The load factor is 70 percent.

*For a 6.8-month (208-day) refueling interval, 0.7 load factor, and rated core power of 2117 thermal megawatts averaged over the interval, the scheduled fractional fuel replacements listed on Figure 2.5.1(a) give 98,300 MWD/T average core fuel burnup.

TABLE 2.5.1.1(a)

FUEL ISOTOPIC COMPOSITIONS SUPPLIED (ATOM PERCENT)

Core Zones*	Pu-239	Pu-240	Pu-241	Pu-242	Fission Product Pairs	U-235	U-236	U-238
Z1	14.11	6.95	1.56	0.84	0	0	0	76.54
Z2	16.21	7.81	1.75	0.94	0	0	0	73.29
Z3	18.27	8.77	1.97	1.06	0	0	0	69.93
Blanket Regions*								
RB1	0	0	0	0	0	0.3	0	99.7
RB2								
AB1, n**								
AB2, n								
AB3, n								
AB4, n								
AB5, n								
AB6, n								
CRB	0	0	0	0	0	0.3	0	99.7

*See Figure 2.5.1(b) for a description of the symbolism used to specify the various zones and regions of the reactor.

**For n=1, 2, and 3.

TABLE 2.5.1.1(b)

FUEL ISOTOPIC COMPOSITIONS DISCHARGED (ATOM PERCENT)

Core Zones*	Pu-239	Pu-240	Pu-241	Pu-242	Fission Product Pairs	U-235	U-236	U-238
Z1	11.05	6.75	1.55	0.84	11.40	0	0	68.41
Z2	12.05	7.54	1.74	0.94	11.50	0	0	66.23
Z3	13.30	8.43	1.96	1.06	11.30	0	0	63.95
Blanket Regions*								
RB1**	3.597	0.166	0.006	0	1.012	0.197	0.027	94.995
RB2**	1.274	0.022	~0	0	0.155	0.263	0.010	98.276
AB1,1*	5.844	0.618	0.041	0	2.408	0.124	0.048	90.917
AB2,1	5.225	0.511	0.031	0	1.684	0.139	0.047	92.363
AB3,1	4.382	0.358	0.018	0	1.079	0.161	0.042	93.960
AB4,1	3.473	0.222	0.009	0	0.644	0.187	0.036	95.429
AB5,1	2.624	0.127	0.004	0	0.367	0.212	0.029	96.637
AB6,1	1.933	0.071	0.002	0	0.207	0.232	0.023	97.532

*Axial blanket compositions are listed only for the region above core zone 1. Listed radial blanket compositions are for the portion of the radial blanket opposite the core. Figure 2.5.1.2 shows supplied and discharged weights of fissile isotopes for all blanket regions, including the corner portion of the radial blanket.

**Assuming a radial blanket residence time of 4.6 years.

TABLE 2.5.1.1(c)

AVERAGE FUEL ISOTOPIC COMPOSITIONS AT BEGINNING OF REFUELING CYCLE* (ATOM PERCENT)

Core Zones	Pu-239	Pu-240	Pu-241	Pu-242	Fission Product Pairs	U-235	U-236	U-238
Z1	12.83	6.96	1.58	0.85	4.27	0	0	73.51
Z2	14.46	7.83	1.77	0.95	4.31	0	0	70.68
Z3	15.96	8.79	1.99	1.07	4.71	0	0	67.48
Blanket Regions								
RB1	1.976	0.047	0.001	0	0.345	0.244	0.016	97.371
RB2	0.636	0.006	~0	0	0.051	0.282	0.005	99.020
AB1, 1**	2.574	0.159	0.007	0	0.656	0.222	0.023	96.359
AB2, 1	2.240	0.130	0.005	0	0.439	0.230	0.022	96.934
AB3, 1	1.806	0.087	0.003	0	0.272	0.242	0.019	97.571
AB4, 1	1.371	0.052	0.001	0	0.158	0.254	0.015	98.149
AB5, 1	0.993	0.028	0.001	0	0.089	0.266	0.012	98.611
AB6, 1	0.707	0.015	~0	0	0.050	0.275	0.009	98.944

*For an equilibrium cycle with an interval of 6.8 months between refueling. See equilibrium fuel cycle assumptions discussed in Section 2.5.1.1. This isotopic composition gives k_{eff} = 1.045 with no boron present.

**Axial blanket compositions are listed only for the region above core zone 1. Listed radial blanket compositions are for the portion of the radial blanket opposite the core. Figure 2.5.1.2 shows supplied and discharged weights of fissile isotopes for all blanket regions, including the corner portion of the radial blanket.

TABLE 2.5.1.1(d)

AVERAGE FUEL ISOTOPIC COMPOSITIONS AT
MID BURNUP OF THE REFUELING CYCLE (ATOM PERCENT)

Core Zones	Pu-239	Pu-240	Pu-241	Pu-242	Fission Product Pairs	U-235	U-236	U-238
Z1	12.44	6.96	1.58	0.85	5.70	0	0	72.47
Z2	13.92	7.82	1.77	0.95	5.75	0	0	69.79
Z3	15.53	8.78	1.99	1.07	5.65	0	0	66.98
Blanket Regions								
RB1	2.169	0.057	0.002	0	0.408	0.237	0.017	97.114
RB2	0.707	0.007	~0	0	0.060	0.279	0.006	98.941
AB1, 1*	3.301	0.227	0.012	0	0.929	0.199	0.029	95.303
AB2, 1	2.886	0.186	0.008	0	0.630	0.209	0.028	96.053
AB3, 1	2.344	0.126	0.005	0	0.394	0.224	0.024	96.883
AB4, 1	1.794	0.076	0.002	0	0.231	0.240	0.020	97.637
AB5, 1	1.311	0.042	0.001	0	0.130	0.255	0.015	98.246
AB6, 1	0.941	0.023	~0	0	0.074	0.266	0.012	98.684

*Axial blanket compositions are listed only for the region above core zone 1. Listed radial blanket compositions are for the portion of the radial blanket opposite the core. Figure 2.5.1.2 shows supplied and discharged weights of fissile isotopes for all blanket regions, including the corner portion of the radial blanket.

TABLE 2.5.1.1(e)

AVERAGE FUEL ISOTOPIC COMPOSITIONS AT END OF THE REFUELING CYCLE (ATOM PERCENT)*

Core Zones	Pu-239	Pu-240	Pu-241	Pu-242	Fission Product Pairs	U-235	U-236	U-238
Z1	12.07	6.94	1.58	0.85	7.12	0	0	71.44
Z2	13.41	7.79	1.77	0.95	7.19	0	0	68.89
Z3	15.12	8.76	1.99	1.07	6.57	0	0	66.47
Blanket Regions								
RB1	2.380	0.069	~0.001	0	0.478	0.232	0.019	96.821
RB2	0.786	0.008	~0	0	0.072	0.277	0.007	98.85
AB1, 1**	4.007	0.313	0.018	0	1.258	0.178	0.035	94.191
AB2, 1	3.522	0.257	0.013	0	0.860	0.190	0.033	96.125
AB3, 1	2.882	0.177	0.008	0	0.542	0.207	0.029	96.155
AB4, 1	2.225	0.107	0.004	0	0.319	0.226	0.024	97.095
AB5, 1	1.639	0.060	0.002	0	0.180	0.244	0.019	97.856
AB6, 1	1.183	0.033	0.001	0	0.102	0.258	0.015	98.408

*Corresponds to $k_{eff} = 1.012$

**Axial blanket compositions are listed only for the region above core zone 1. Listed radial blanket compositions are for the portion of the radial blanket opposite the core. Figure 2.5.1.2 shows supplied and discharged weights of fissile isotopes for all blanket regions, including the corner portion of the radial blanket.

2.5.1 FUEL REQUIREMENTS AND REFUELING PATTERN

2.5.1.1 Regional Fuel Compositions (Continued)

The change in reactivity for the 6.8-month burnup interval represented by the compositions given in Tables 2.5.1.1(c) and (e) was computed to be 3.3 percent, including the reactivity penalty for nonuniform fuel burnup. The end-of-cycle fuel compositions, Table 2.5.1.1(e), give a computed reactivity $k_{eff} = 1.012$.

Although the tabulated fuel compositions were computed specifically for the variable composition concept using a one-dimensional multigroup diffusion calculation (see Section 3.6), the results may also be considered indicative of the characteristics which are attainable using the previously mentioned single composition out-in fuel shuffling scheme.

2.5.1.2 Neutron Balance Data and Breeding Ratio

Table 2.5.1.2(a) is a tabulation of the neutron balance data for the mid-burnup condition of the equilibrium cycle. The column labeled Axial Blanket 1 contains events summed over the n strips [see Figure 2.5.1(b)], for the axial blankets adjacent to core zone 1. The columns labeled Axial Blanket 2 and 3 contain similar quantities for the axial blankets adjacent to core zones 2 and 3, respectively. The symbols C, F, L_R, and L_Z stand for nonfission captures, fissions, radial leakage, and axial leakage, respectively. The superscripts denote the isotope considered as follows:

01	Fission Product Pairs
04	Beryllium
05	Natural Boron
08	Oxygen
11	Sodium
126	Stainless Steel
25	U-235
28	U-238
49	Pu-239
40	Pu-240
41	Pu-241
42	Pu-242

The capture events in Beryllium do not include the Be(n, 2n) reaction which captures an additional 1.2 neutrons, but provides a source of 2.4 neutrons. The Be(n, α) reaction is included in the balance table. Li-6 buildup as a byproduct of the (n, α) reaction is negligibly small in this spectrum for the fuel (and hence BeO) lifetime in the core.

TABLE 2.5.1.2(a)

NEUTRON BALANCE DATA FOR THE MID-BURNUP CONDITION OF THE EQUILIBRIUM CYCLE

Events	Core Z1	Core Z2	Core Z3	Radial Blanket	Axial Blanket 1	Axial Blanket 2	Axial Blanket 3	Corner Radial Blankets	Total Events
Neutron Source	31.86	32.12	20.97	3.03	4.78	4.12	2.51	0.61	100.00
C^{01}	0.99	0.86	0.48	0.03	0.04	0.03	0.02	0.01	2.46
C^{04}	0.13	0.12	0.08	0	0.02	0.02	0.01	0.01	0.39
C^{05}	0.64	0.57	0.33	0	0	0	0	0	1.54
C^{08}	0.06	0.05	0.03	0.01	0.01	0.01	0.01	~0	0.18
C^{11}	0.32	0.28	0.16	0.05	0.17	0.14	0.08	0.06	1.26
C^{126}	0.72	0.65	0.39	0.16	0.34	0.30	0.17	0.11	2.84
C^{25}	0	0	0	0.03	0.04	0.04	0.03	0.01	0.15
C^{28}	7.30	6.24	3.48	4.40	7.29	6.25	3.77	1.62	40.35
C^{49}	2.37	2.29	1.45	0.17	0.42	0.36	0.22	0.14	7.42
C^{40}	1.72	1.68	1.08	0.01	0.05	0.04	0.03	0.01	4.62
C^{41}	0.26	0.25	0.16	~0	~0	~0	~0	~0	0.67
C^{42}	0.18	0.17	0.11	~0	~0	~0	~0	~0	0.46
Total Captures	14.69	13.16	7.75	4.86	8.38	7.19	4.34	1.97	62.34
F^{25}	0	0	0	0.08	0.12	0.11	0.06	0.02	0.39
F^{28}	0.89	0.84	0.50	0.37	0.32	0.27	0.17	0.04	3.40
F^{49}	7.75	7.83	5.13	0.60	1.23	1.06	0.64	0.15	24.39
F^{40}	0.57	0.62	0.42	~0	~0	~0	~0	~0	1.61
F^{41}	1.51	1.52	1.00	~0	~0	~0	~0	~0	4.03
Total Fissions	10.79	10.88	7.10	1.05	1.67	1.44	0.87	0.21	34.01
L_R	--	--	--	0.70	--	--	--	0.07	0.77
L_Z	--	--	--	--	1.10	1.10	0.60	0.08	2.88
Total Leakage					1.10	1.10	0.60	0.15	3.65
								Total Loss	100.00

2.5.1 FUEL REQUIREMENTS AND REFUELING PATTERN

2.5.1.2 Neutron Balance Data and Breeding Ratio (Continued)

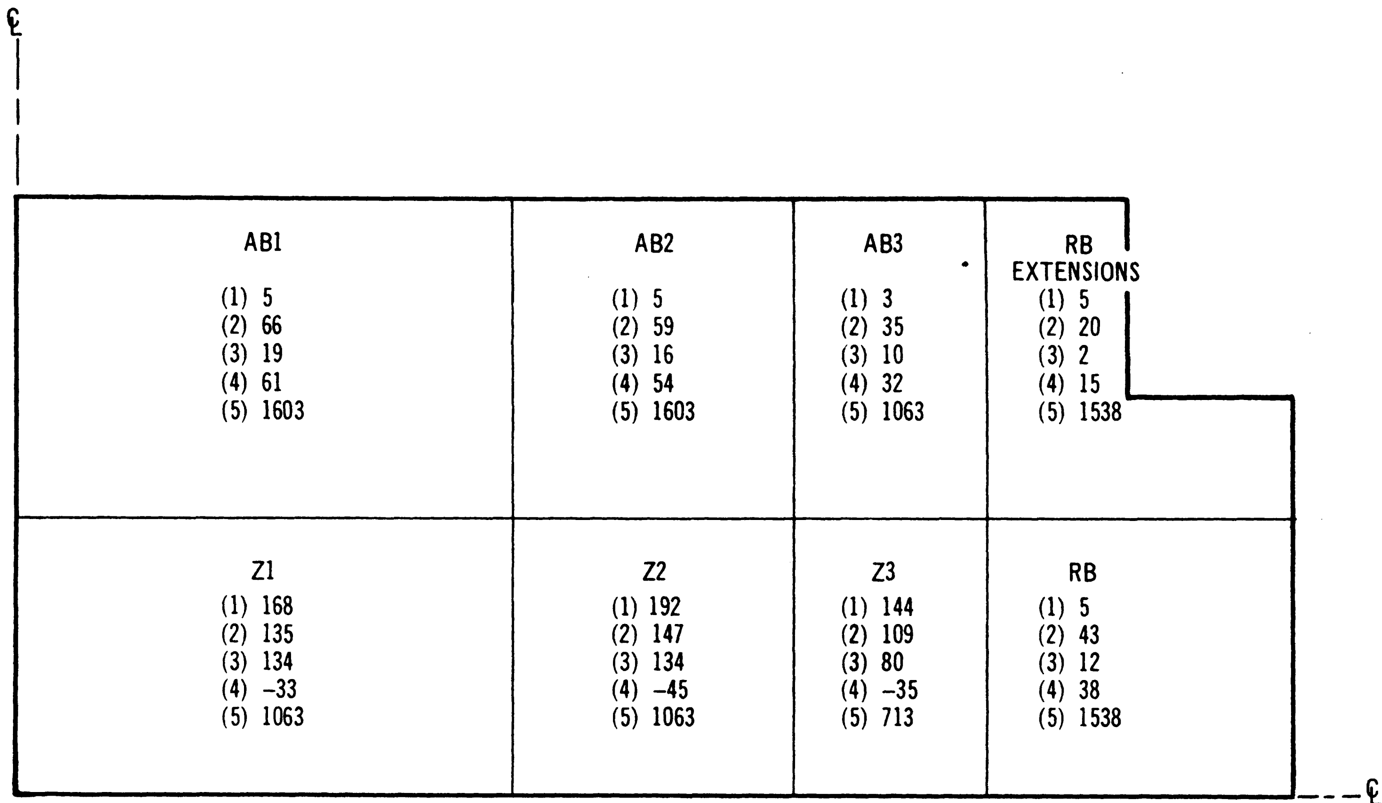
The weights of the fissile isotopes as they are supplied and discharged in each zone of the core and axial blankets and in the radial blanket (including the corner blanket), are shown in Figure 2.5.1.2. Also shown in this figure are the weights of the fissile isotopes destroyed in each zone and region mentioned above. The weights of fissile materials supplied, discharged, and destroyed listed on the figure are for a single refueling cycle of 6.8 months.

Contributions of the total breeding ratio by the three core zones, axial blankets, and the radial blankets (including the corner blanket), and the total breeding ratio as per the AEC definition discussed in Section 3.6.2.4 are listed in Table 2.5.1.2(b). The breeding ratios were computed from the fissile weight values given in Figure 2.5.1.2. The corrected breeding ratio, 1.24, is 0.02 higher than that computed simply from the fissile weight balance of Figure 2.5.1.2. This takes into account the 0.012 excess reactivity at the end of the refueling cycle [Table 2.5.1.1(e)]. Also included in Table 2.5.1.2(b) are the instantaneous core conversion ratios for the three core zones, defined as the rate of fissile production in a given zone to the rate of fissile destruction in that zone. The breeding ratio of 1.24 ~~1.257~~ for the reference reactor fuel cycle is computed to yield a fissile material doubling time of approximately 15.5 years including allowances for reprocessing losses (see Appendix 6.10).

2.5.1.3 Fuel Lifetime

The mean lifetime of the core fuel is 943 days. This value is based on a total core fuel inventory (input U+Pu oxide) of 14,550 kg, an average exposure of approximately 100,000 MWD/T within each core zone (megawatt days per 2000 pounds of input U+Pu), and a 70 percent load factor. Fuel lifetimes in core zones 1, 2, and 3 are, respectively, 250, 850, and 1128 days for the refuelling schedule discussed in Section 2.5.1.

The axial blanket lifetimes equal those of their associated core fuel assemblies for the various zones. The radial blanket lifetime is 4.6 years.



(1) FISSILE kg SUPPLIED PER BATCH.*
(2) FISSILE kg DISCHARGED PER BATCH.
(3) FISSILE kg DESTROYED PER BATCH.
(4) FISSILE kg EXCESS PER BATCH.
(5) TOTAL kg FUEL PER BATCH.

* ONE BATCH CORRESPONDS TO 1/4 EACH OF ZONES 1 AND 2 AND THEIR ASSOCIATED AXIAL BLANKETS, 1/6 OF CORE ZONE 3 AND ITS ASSOCIATED AXIAL BLANKET AND 1/8 OF THE RADIAL BLANKET. WEIGHTS ARE GIVEN TO THE NEAREST KILOGRAM.

Figure 2.5.1.2. Fuel Supplied, Discharged and Destroyed

TABLE 2.5.1.2(b)

REGIONAL BREEDING RATIO CONTRIBUTIONS
AND CORE CONVERSION RATIOS

	Contribution to Breeding Ratio Above Unity	Core Conversion** Ratio
Core Zone 1 (Z1)	-0.08	0.79
Core Zone 2 (Z2)	-0.11	0.70
Core Zone 3 (Z3)	-0.09	0.61
Axial Blanket Above and Below Z1	0.15	--
Axial Blanket Above and Below Z2	0.14	--
Axial Blanket Above and Below Z3	0.08	--
Radial Blanket Adjacent to Core (RB1 + RB2)	0.09	--
Radial Blanket Extensions	0.04	--
TOTAL	0.22	

Total Breeding Ratio (TBR) 1.22

Corrected TBR = 1.24*

*Includes an increment of 0.02 to account for the 0.012 excess reactivity remaining at the end of the refueling cycle.

**Includes increments of 0.03 in Zones 1 and 2 and 0.02 in Zone 3 to account for the above 0.012 excess reactivity.

2.5.2 POWER ... 'LUX DISTRIBUTIONS

2.5.2.1 Spectral Flux and Power Distributions

Figure 2.5.2.1(a) is a histogram of the relative power per unit lethargy in zone 1 of the core as a function of lethargy (logarithmic energy scale). The energy group numbers are listed on the histogram (see Section 3.6.1.3).

The plot in Figure 2.5.2.1(a) is for the mid-burnup condition in the equilibrium system. The spectrum at different times over the 6.8-month burnup interval is not significantly different from that shown in the figure. A brief tabulation of spectral indexes not readily derived from the figure is given below. The percent of the fissions due to neutrons whose energies are less than 9Kev is given in the second column and the value marking the energy at which equal numbers of higher and lower energy neutrons cause fissions is given in the last column.

Core Zone	Percent Fissions Below 9 Kev	Mean Fission Energy (Kev)
Z1	27	110
Z2	25	128
Z3	23	139

There is a slight hardening of the fission spectrum proceeding outward radially from the core center. This reflects the increasing plutonium content in the different zones.

Figure 2.5.2.1(b) is a histogram of the relative neutron flux per unit lethargy averaged over zone 1 of the core. The flux spectrum does not vary significantly with burnup. A slight hardening of the flux spectrum is obtained in going from zone 1 to 2 to 3.

Figure 2.5.2.1(c) is a histogram of the relative adjoint fluxes at the center of the core.

The effect on the flux spectrum (and power spectrum) of the sodium resonance at 2.85 Kev may be observed at a lethargy of 8 in the histogram. This resonance produces a large dip in the spectra for energy group 16, and a large rise in group 17 which is just below the resonance energy. A peak in the power spectrum in group 3 is due to the large yield of U-238 fissions in that energy group.

The adjoint flux, which is a measure of the neutron importance at the given energy, drops off sharply below the U-238 fission threshold and continues to decrease gradually to about 1000 ev, reflecting the increasing alpha values (σ_c/σ_f) with decreasing energy and also the increasing absorption rate in fertile and parasitic materials relative to the fission rate.

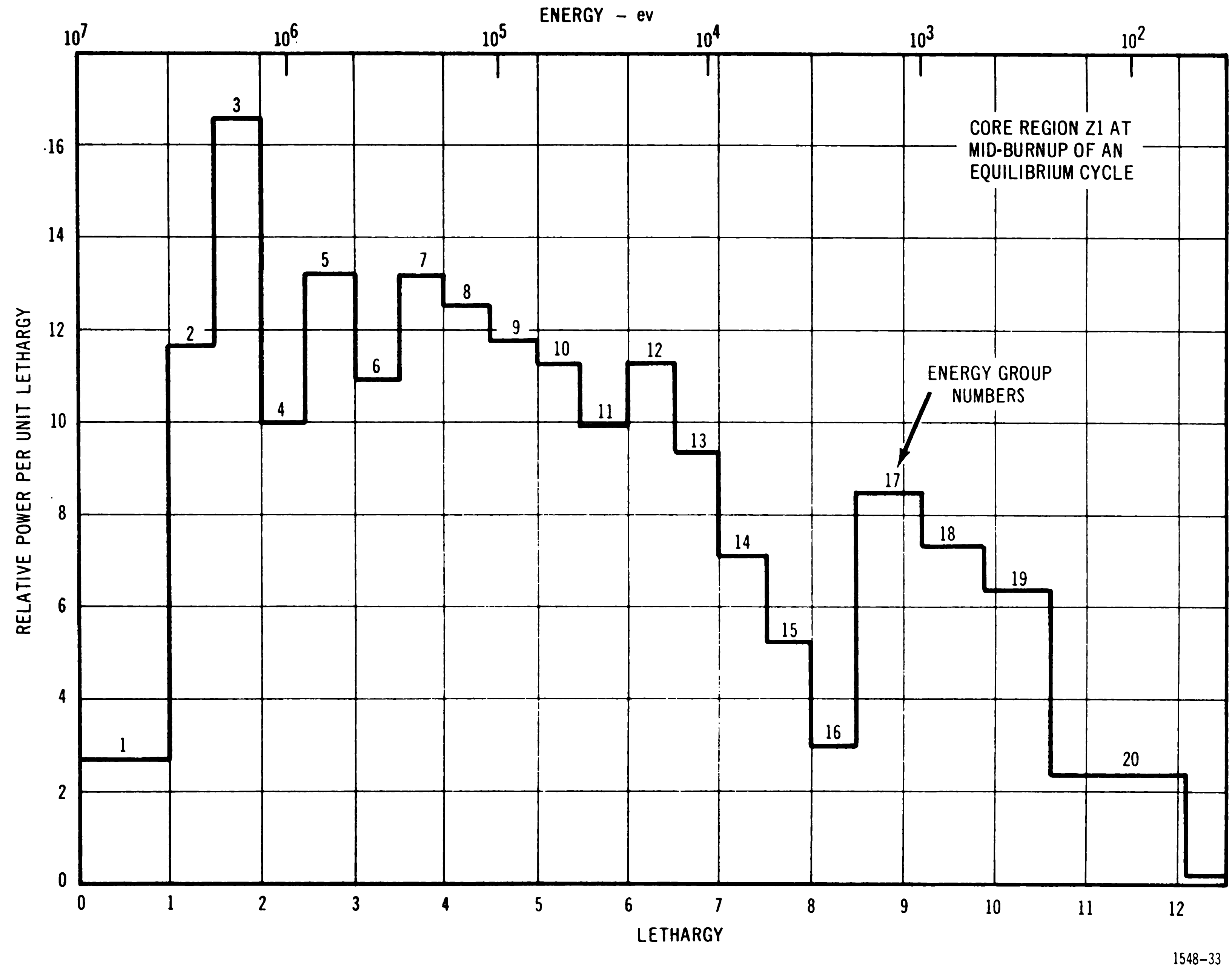


Figure 2.5.2.1(a). Power Spectrum

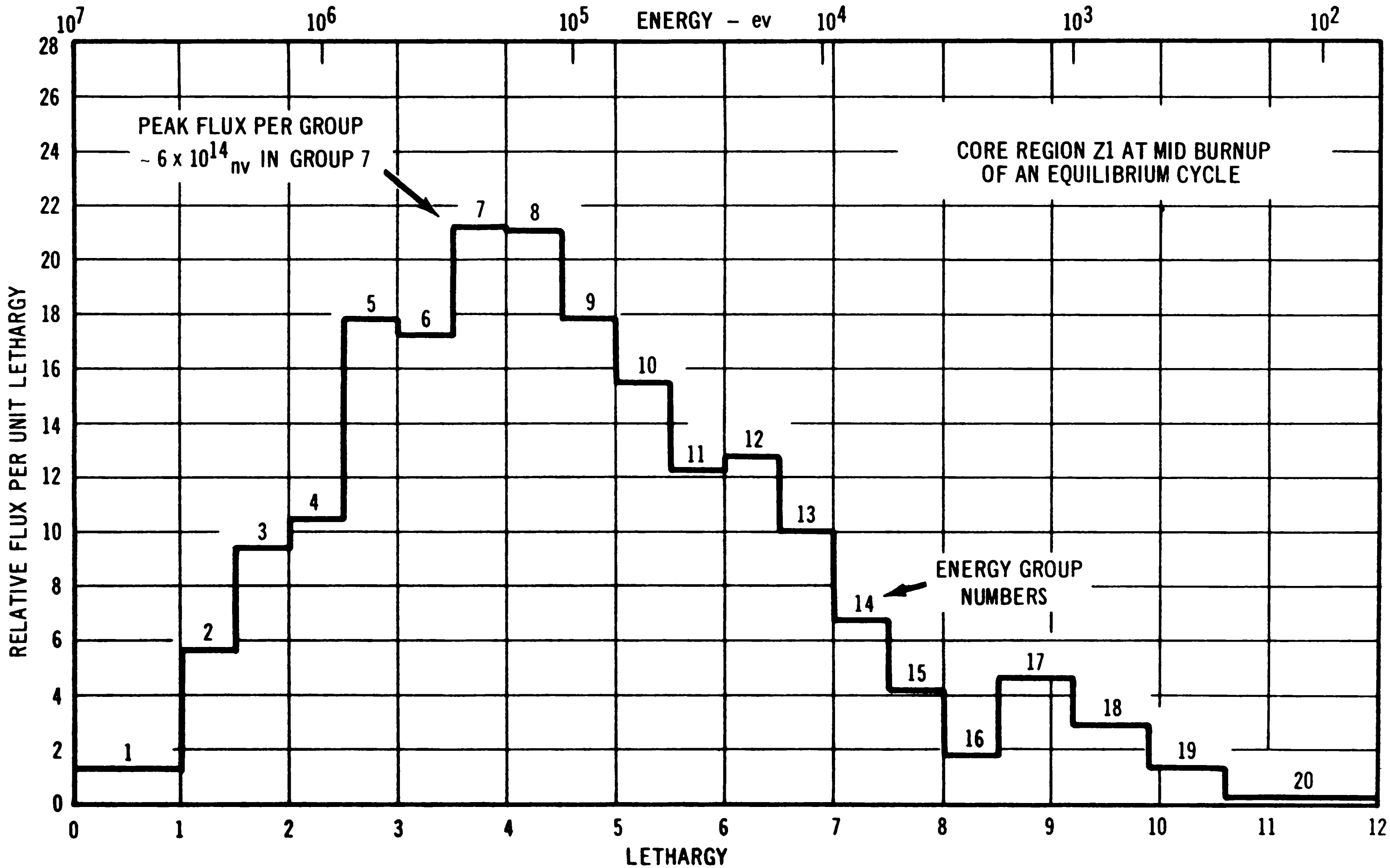


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Figure 2.5.2.1(b). Flux Spectrum

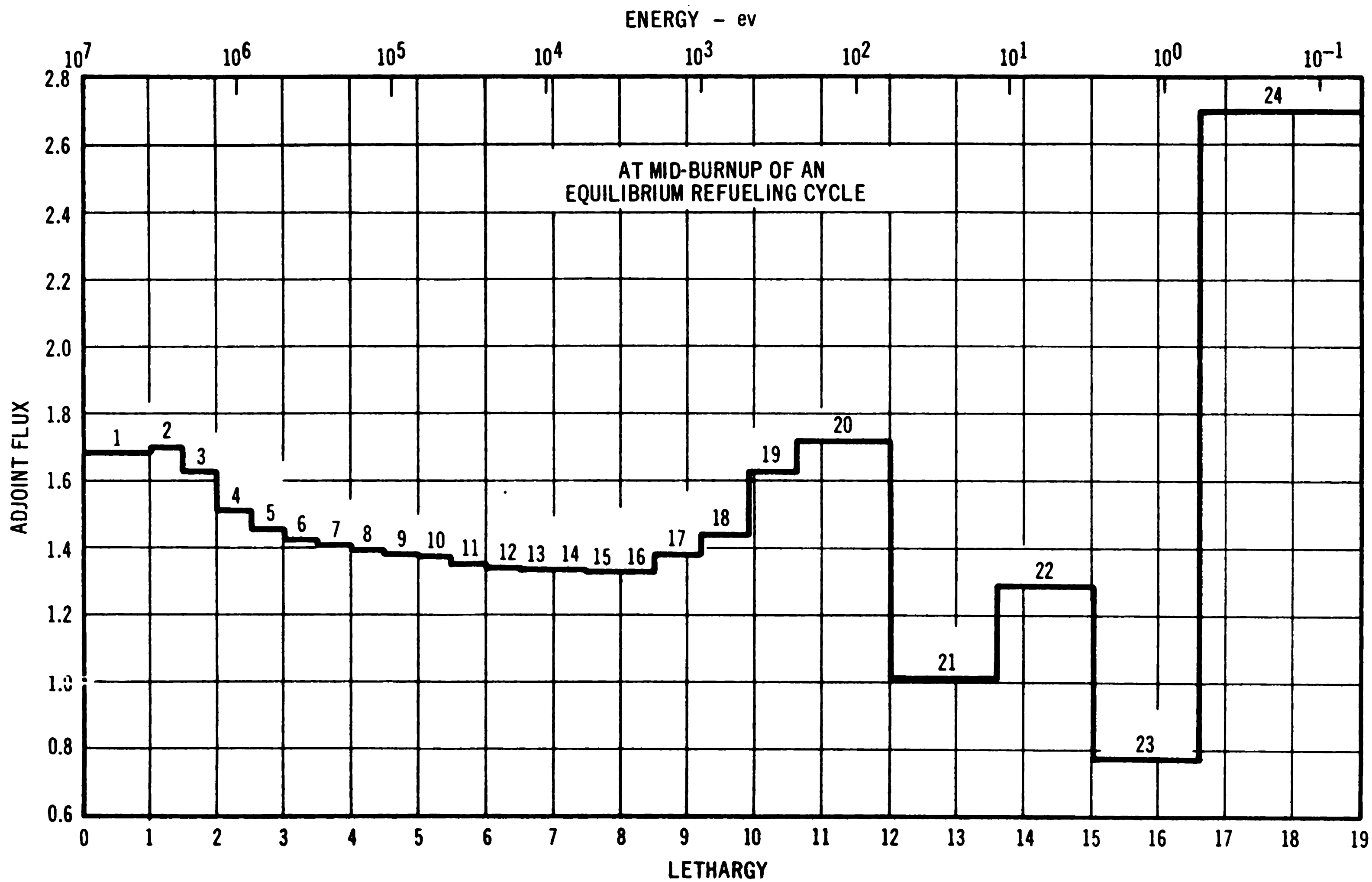

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Figure 2.5.2.1(c). Adjoint Flux Spectrum

2.5.2 POWER AND FLUX DISTRIBUTIONS (Continued)

2.5.2.2 Spatial Power Distributions

The radial and axial power distributions computed using the one-dimensional 24-group diffusion calculations described in Section 3.6 are illustrated in Figures 2.5.2.2(a) and (b). These power distributions have been found to be applicable to within 3 percent over the entire core region. This was established by a two-dimensional, four-group "PDQ" computation which is discussed in Section 3.6.2.1. Power distributions at both the beginning and the end of the refueling cycle are plotted in Figures 2.5.2.2(a) and (b).

The various core power peaking factors are listed in Table 2.5.2.2(a). These include radial and axial peak power densities in each zone relative to the average core power density, and local peaking factors due to the slight flux depression near control rods and due to the presence of fuel bundles at different burnup levels within the same zone. It is seen that the maximum over-all peak-to-average core power density is 1.56 and occurs in zone 2. The power distribution in a unit cell of fuel assemblies surrounding a single control rod is shown in Figure 2.5.2.2(c).

The thermal power generated in each core zone and in the blankets at the beginning and end of a refueling cycle are listed in Table 2.5.2.2(b). The average power distribution given in the table was computed using the PDQ results mentioned previously. The distributions at the beginning and end of the refueling interval were synthesized from the "PDQ distribution" and the results of one-dimensional MISY calculations discussed in Section 3.6.2.1.

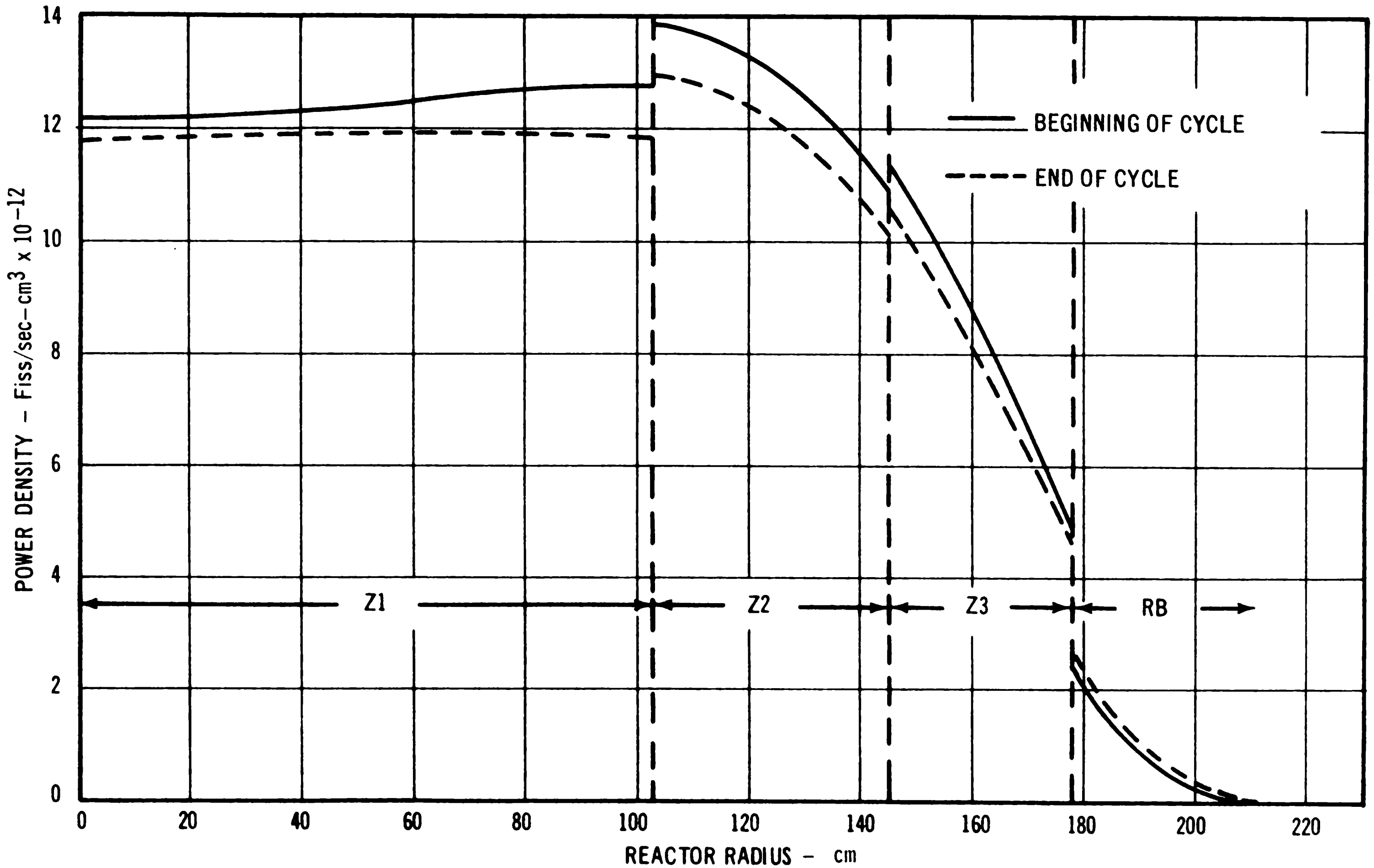

Figure 2.5.2.2(a). Radial Power Distribution

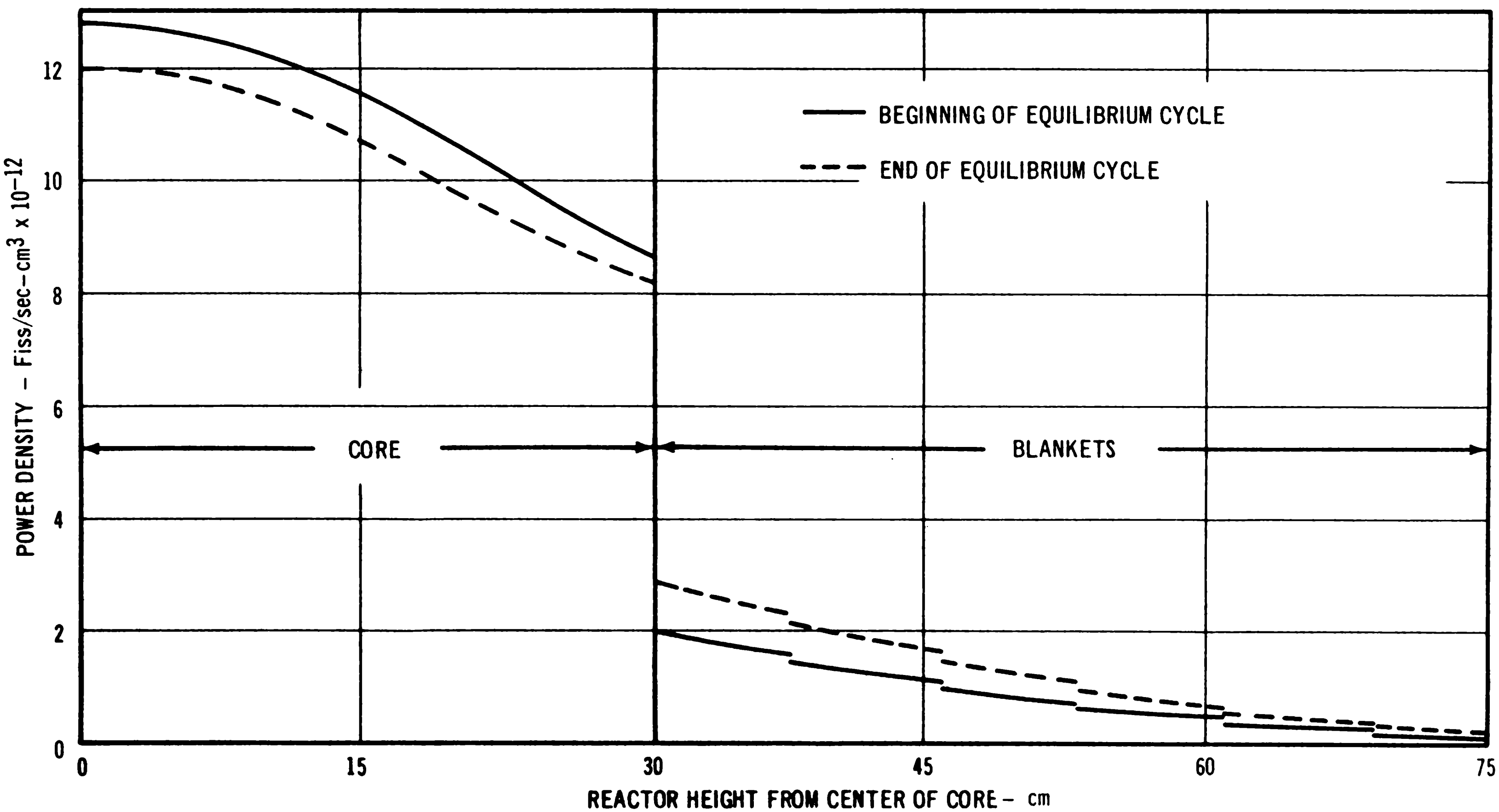

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Figure 2.5.2.2(b). Axial Power Distribution

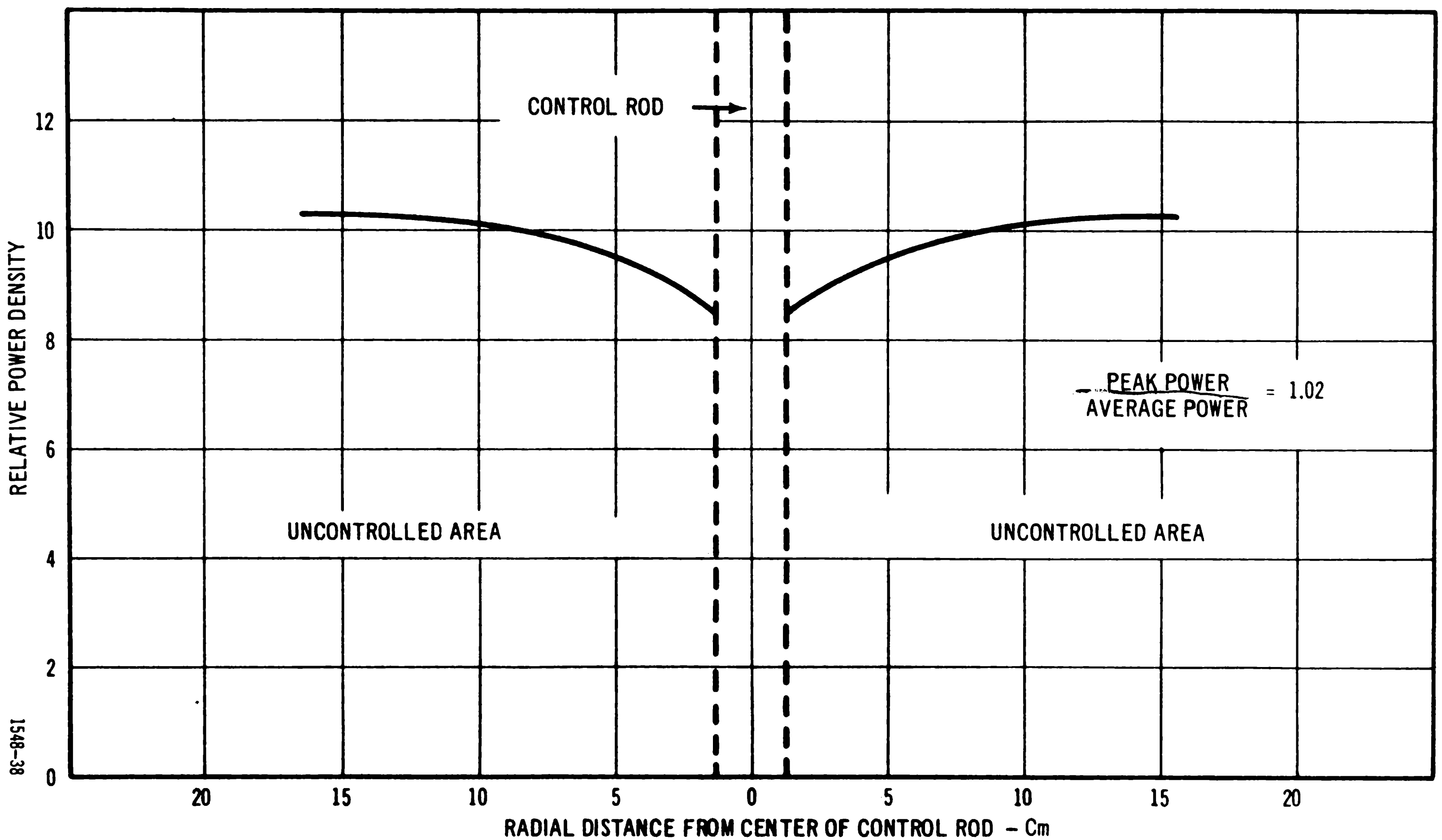

Figure 2.5.2.2(c). Power Distribution Around Control Rod

TABLE 2.5.2.2(a)

CORE POWER PEAKING FACTORS

		Beginning of Refueling Cycle	End of Refueling Cycle
Radial Peak in	Zone 1 to Average for Core*	1.11	1.11
	Zone 2	1.23	1.22
	Zone 3	1.06	1.06
Axial Peak in	Zone 1 to Average	1.14	1.14
	Zone 2	1.14	1.14
	Zone 3	1.14	1.14
Inter-Fuel Element Peaking in	Zone 1	1.08	1.08
(due to adjacent fuel	Zone 2	1.09	1.09
elements being at different irradiations)	Zone 3	1.12	1.12
Control Rod Peaking Factor**		1.02	1.00

TABLE 2.5.2.2(b)

REGIONAL DISTRIBUTION OF POWER GENERATION

	Power at Beginning of Cycle MWt	Power at End of Cycle Mwt	Power Averaged Over the Interval MWt
Core Zone 1	816	770	793
Core Zone 2	828	772	800
Core Zone 3	538	510	524
Radial Blanket Adjacent to Core	71	83	76
Axial Blanket Above Zone 1	96	143	121
Axial Blanket Above Zone 2	85	128	107
Axial Blanket Above Zone 3	51	76	63
Corner Radial Blanket Extensions	15	18	16
Total	2500	2500	2500

*Defined as the ratio of the maximum power density in the zone to the average power density of the entire core.

**Peak-to-average radial power density in a unit cell containing a single control rod.

2.5.3 CONTROL REQUIREMENTS AND ASSEMBLY WORTHS

2.5.3.1 Total Requirements

The major reactivity requirements for control rods are as follows:

Burnup (Equilibrium Refueling Cycle)	-3.3 percent
Cold to Operating at "Zero" Power*	-0.6 percent
Zero to Full Power	-0.9 percent
Reserve Control for Safety and Shutdown Margin	-1.5 percent
Total	-6.3 percent

The 3.3 percent reactivity for burnup is consistent with the changes in isotopic fuel composition during a refueling cycle as given in Tables 2.5.1.1(c) and (e). This includes a 0.6 percent reactivity penalty for nonuniform axial burnup since the listed isotopic fuel compositions are average values for each zone. Reactivity changes from cold to operation at zero power and zero to full power (2394 F average fuel temperature at 950 F average coolant temperature) are based on the reactivity coefficients discussed in Section 2.5.4.

The ~~1.5~~ percent reactivity for safety and shutdown margin is adequate to override the reactivity addition in the event of a fuel slump accident in which all of the oxide fuel compacts to its theoretical density within the clad. The reactivity addition due to such a total fuel slump accident was computed to be 1.0 ~~1.5~~ percent.

The full range of core reactivity from the cold to the hot full power condition is made up of the following components:

Fuel Doppler Effect	-1.14 percent
Core and Blanket Radial Expansion**	-0.32 percent
Clad Axial Expansion	-0.01 percent
Fuel Axial Expansion	-0.04 percent
Sodium Thermal Expansion	+0.01 percent
Sodium Displacement by Fuel Radial Expansion	~0

*Reactor brought from 400 F to 800 F at low power.

**Added volume is filled by sodium.

2.5.3 CONTROL REQUIREMENTS AND ASSEMBLY WORTHS (Continued)

2.5.3.2 Type and Distribution of Rods in Core

Control of the reactor is to be accomplished by top-entry boron poison rods having no followers. The rods have been sized as 2-inch OD, 0.5-inch ID boron carbide (natural boron) annuli clad in stainless steel and cooled on all surfaces by sodium. It has been found that a total of 85 rods of this design spaced equally on a 15-inch triangular pitch will provide the necessary negative reactivity for the core.* Furthermore, under normal operating conditions, the average rod worth will be 0.075 percent in reactivity (about 23 cents). In a highly-reactive control configuration, rod ejection will result in less than a 1 dollar reactivity step (see Section 2.5.3.3).

The control rod drives have been specified to achieve 50 percent insertion (30-inch travel) within 200 milliseconds of receipt of scram signal. This requirement is based on design considerations described in Section 3.2.3 which suggest that this scram capability, coupled with the computed Doppler $T \frac{dk}{dT}$ for the reactor of -0.01, will permit recovery with over-all system integrity from a total reactivity insertion of 4 dollars inserted at a rate of 10 dollars/sec. [The maximum reactivity increment due to a symmetric central total sodium voidage from four-ninths of the core volume has been computed as 3.5 dollars (see Section 2.5.4.2).]

Control requirements for reactor startup and approach to power call for reactivity insertions at the rate of possibly 0.33 cents/sec in the subcritical portion of startup and a maximum of 2 cents/-sec in the critical condition.(1)

2.5.3.3 Maximum Rod and Fuel Assembly Worth

Two primary evaluations made in connection with assessment of maximum rod worth are:

1. The rod ejection criteria in which the assessed maximum reactivity step resulting from sudden removal of a rod from a postulated high rod worth control pattern has been computed at less than 1 dollar.

2. The stuck rod criteria in which the reactivity of all zonal configurations of the core with all credible fuel loadings have been assessed to ensure the reactor can be held subcritical by insertion of all available rods in the presence of a single stuck (withdrawn) rod.

Control Rod Ejection Accident - A severe control-rod-ejection accident has been computed in which the central control rod is suddenly removed from a cold core. The core is in the freshly-fueled, cold critical condition, and the central control rod is surrounded by an annulus of

*A 6.4 percent Δk rod worth was computed for all control rods unirradiated. This is reduced to 6.2 percent for the equilibrium cycle if control rods are replaced on the same schedule as surrounding fuel assemblies within each control rod cell. See Section 3.6.2.6 for more details.

2.5.3 CONTROL REQUIREMENTS AND ASSEMBLY WORTHS (Continued)

2.5.3.3 Maximum Rod and Fuel Assembly Worth (Continued)

uncontrolled fuel. The volume of uncontrolled fuel represents an amount normally controlled by six control rods. The remainder of the core volume is poisoned just enough to maintain criticality. The situation is illustrated graphically in Figure 2. 5. 3. 3 which gives the radial power distributions for the two situations. The computed reactivity increment following loss of the central rod is 0. 16 percent (48 cents). This is a small reactivity increment for safety considerations, and it allows some scope for conceiving a control rod pattern which may give a somewhat larger maximum rod worth.

The power density was computed to increase by about 16 percent in the central region of the core upon removal of the central rod.

Stuck Rod Criteria - A highly-reactive fuel configuration would occur, if during refueling, the new batch of fuel for zone 1 were to be loaded in a cluster at the center of a freshly-fueled core. One batch consists of 19 fuel assemblies for zone 1. An estimate has been made of the reactivity worth of the central control rod for this situation assuming all fuel assemblies are at the specified composition for a freshly-fueled core. The rod worth was computed to be approximately -0. 14 percent (-42 cents) for the situation in which the reactor is at its minimum system temperature of 250 F. This means that if all rods are inserted except the central rod, which is assumed to be stuck in the fully withdrawn condition, the system, as described above, would be approximately 1.4 percent subcritical. It is noted that the central rod is worth slightly less in this situation than in the rod-ejection accident computed earlier. This is due to the fact that in the rod-ejection accident the central rod was surrounded by a volume of uncontrolled fuel, whereas in the present case the corresponding volume surrounding the central rod contains all rods in the full-inserted condition.

Fuel Assembly Worth - The reactivity addition resulting from placing a single fuel assembly in the center of the core, replacing an equivalent volume of sodium, was computed to be 0. 20 percent (60 cents) in the hot operating condition.

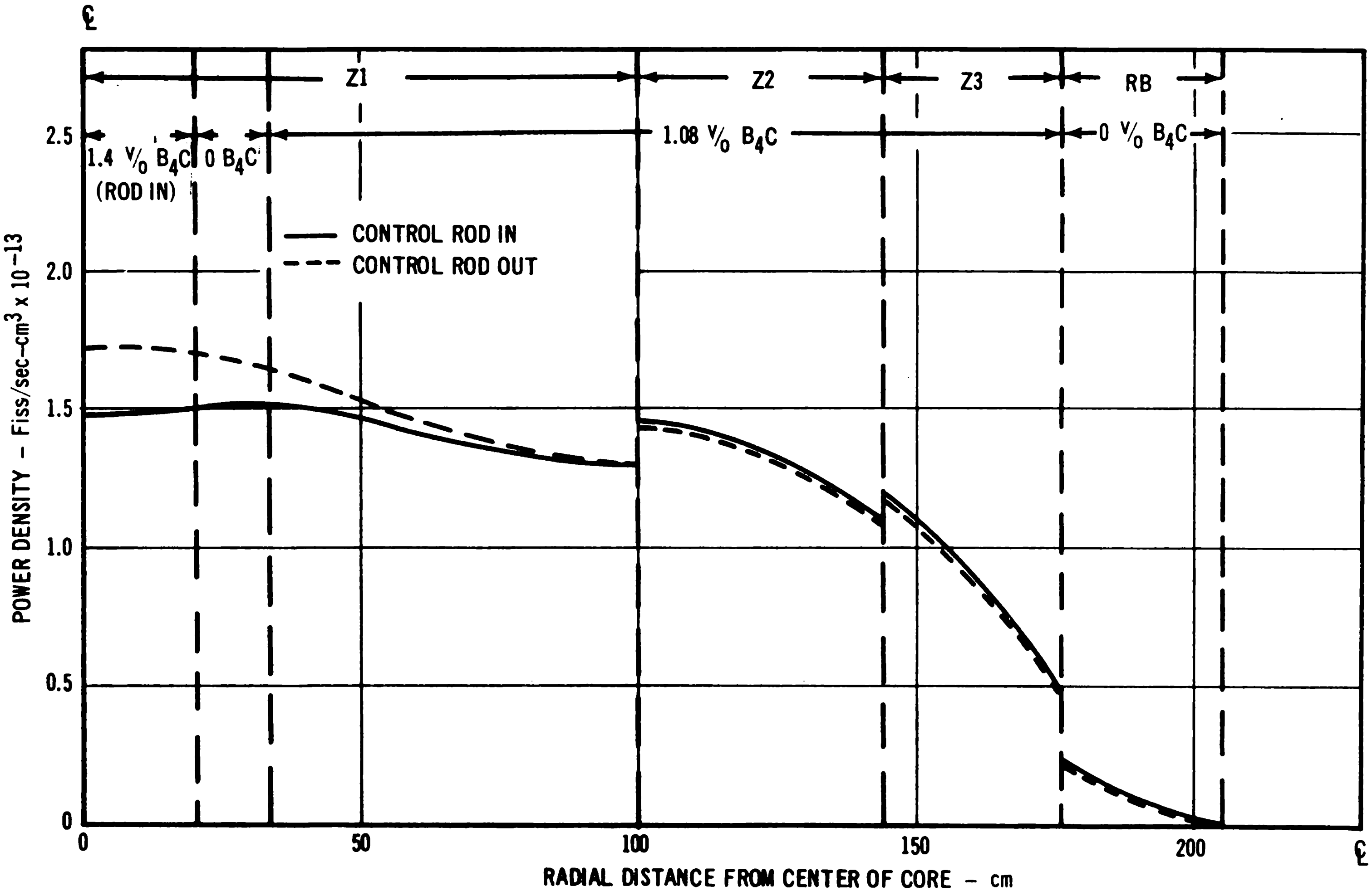

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Figure 2.5.3.3. Power Change for Control Expulsion Accident

2.5.4 REACTIVITY COEFFICIENTS AND KINETICS PARAMETERS

2.5.4.1 Doppler Effect

Two curves of reactivity due to the Doppler effect versus fuel temperature are shown in Figure 2.5.4.1. The dashed curve is for the isothermal Doppler effect which is calculated for a uniform fuel temperature and a uniform change of the fuel temperature. The solid curve is for the calculated Doppler reactivity versus the average fuel temperature (defined as the average fuel temperature in an oxide pellet located at a position of average core power density). The solid curve is corrected for the spatial distribution of the core fuel temperature at power and of the fuel temperature change which is proportional to the power distribution. At the high-temperature end, the solid curve has also been corrected for the fact that ~20 percent of the fuel in the core undergoes a change of phase from the solid to the liquid state and, hence, does not rise in temperature during a portion of the rise of the average fuel temperature. [The heat of fusion of the oxide fuel was assumed to be 108 Btu/lb.* (2)]

The portion of the solid curve above 2400 F, the average fuel temperature at full power, corresponds to a rapid fuel temperature transient (as discussed below) starting from the full power condition; where, the portion of the curve at lower temperatures is for slow transitions in steady-state power levels.

Doppler coefficients at 400, 800, and 2400 F are listed in Table 2.5.4.1. These coefficients are the slopes of the solid curve in Figure 2.5.4.1. Doppler reactivity changes corresponding to rapid fuel temperature transients are als listed in the table. In these transients, the fuel is assumed to rise from an initial to a final temperature distribution in such a short time that heat conduction from the fuel is negligibly small. The $(\Delta k)_{T_1 \rightarrow T_2}$ values for these transients, where T_1 and T_2 are the average fuel temperatures at the beginning and at the end of the transient, respectively, correspond to the solid curve of Figure 2.5.4.1 if T_1 = 2400 F and $T_2 > T_1$. The peak fuel temperatures (center of the hottest oxide pellet) reached at the end of the computed fuel temperature transients are also listed in Table 2.5.4.1.

Power flattening reduces the Doppler coefficient at full power for a reactor of given design (including fuel rod dimensions), assuming that the peak fuel temperature is held just below the melting point at full power. This is because the average fuel temperature is higher in the power flattened reactor than for the unflattened case with uniform fuel composition. Also the average composition is higher for the power-flattened core, yielding a slightly harder spectrum and, hence, a lower Doppler coefficient. The reduction in the Doppler coefficient due to these factors is about 15 percent.

*The heat of fusion is equivalent in energy density to a fuel temperature rise of 1200 F.

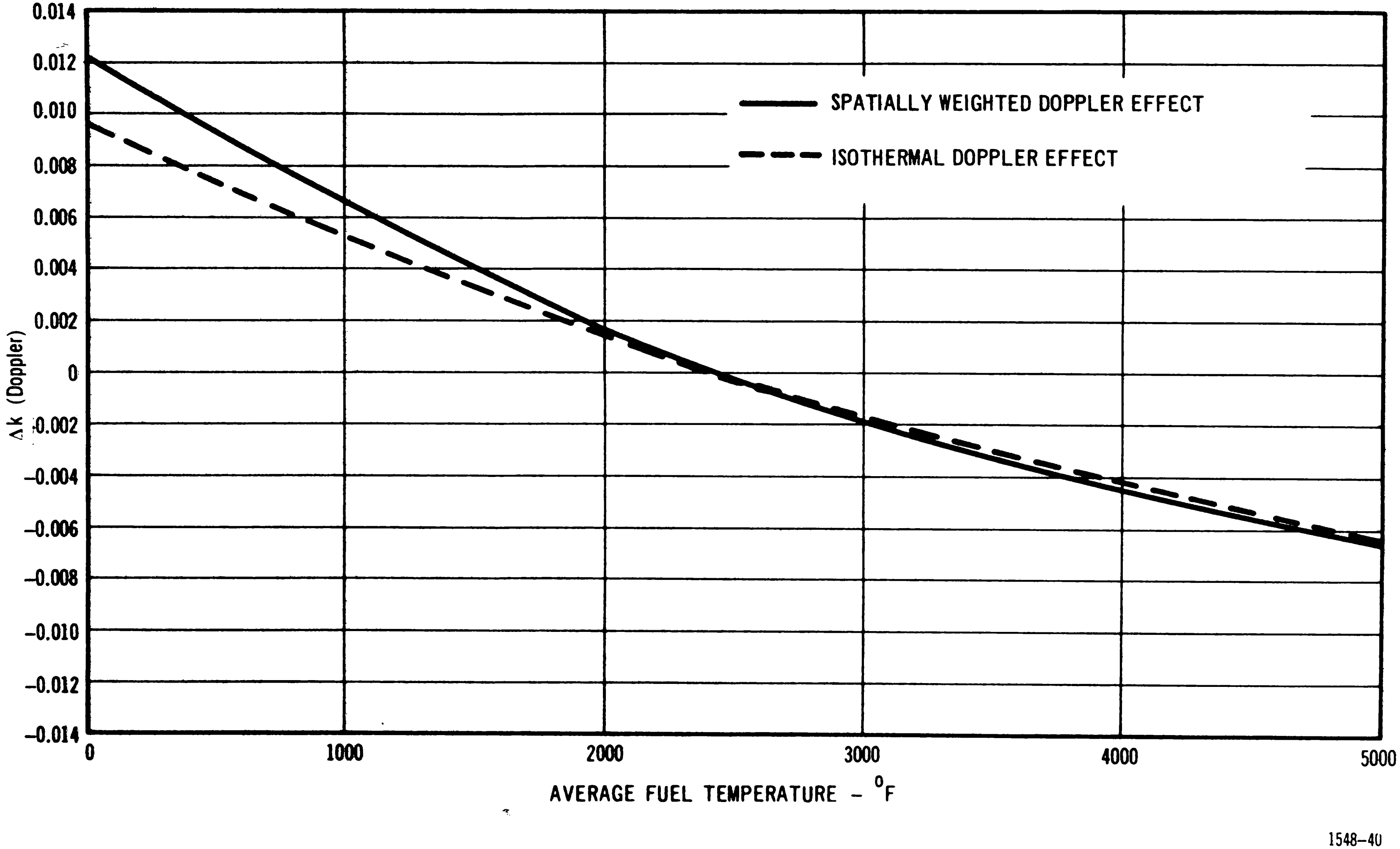

Figure 2.5.4.1. Doppler Reactivity Effect

TABLE 2.5.4.1

DOPPLER COEFFICIENTS AND REACTIVITY CHANGES FOR FUEL TEMPERATURE TRANSIENTS

Average Fuel Temperature (°F)	Doppler Coefficient $\Delta k/°F$
400	-1.1×10^{-5}
800	-7.5×10^{-6}
2400	-3.5×10^{-6}

Initial Average Fuel Temperature (°F)	Final Average Fuel Temperature (°F)	$(\Delta k)_{T_1 \rightarrow T_2}$ In Rapid Transient	Final Peak Fuel Temperature (°F)
400**	2400	-1.14 percent	3500
400**	4700*	-1.81 percent	6500
800**	4700*	-1.50 percent	6500
2400***	4000	-0.45 percent	6000
2400***	4330	-0.51 percent	6500
2400***	4640	-0.57 percent	7500

Although power flattening reduces the Doppler coefficient for the reasons stated above, the available negative Doppler reactivity in a fuel transient is not necessarily reduced. This is because the rise of the peak fuel temperature relative to that of the average fuel temperature is much less in the power flattened case, and, hence, the average fuel temperature may be raised to a higher level for the same limiting temperature condition on the peak fuel temperature as in the nonflattened case. If, for example, 6500 F is taken as a limiting condition‡ on the peak fuel temperature in a transient, it is shown in Table 2.5.4.1 that there is an available negative Doppler reactivity of -0.51 percent for a transient starting at full power (2400 F average fuel temperature) and ending at 6500 F peak temperature. For a core with a uniform composition having a 15 percent higher Doppler coefficient and a 2000 F average fuel temperature at full power, the corresponding available negative Doppler reactivity is actually lower (-0.44 percent).

*For these cases the average fuel temperature reached the melting point and would have risen to about 5000 F if it had not been restrained by the phase transition.

**Isothermal fuel temperature at low power.

***Full power conditions.

‡Limiting in the sense that a higher temperature could lead to serious fuel damage and possibly to a more serious meltdown accident.

2.5.4 REACTIVITY COEFFICIENTS AND KINETIC PARAMETERS (Continued)

2.5.4.2 Reactivity Effects Due to Sodium Loss

Figure 2.5.4.2 illustrates the increment in reactivity incurred when all the sodium is lost from various regions within the reactor as illustrated in Figure 2.5.1(a). The removal of sodium is for void shapes located symmetrically about the center of the core. Curve A gives the reactivity increment for voids that extend over two-thirds of the core height as a function of the radial area of the void. The peak reactivity increment of 1.36 percent is observed to occur for a central void volume of about four-ninths that of the core volume.

Curve B represents similar sodium voiding for various central radial areas, but for axial loss over the entire height of the core. Curve C shows the reactivity increment as a function of the radial void area for axial loss throughout the entire length of the reactor (both core and axial blankets). Total loss of sodium from the reactor is seen to give a small negative reactivity (~0.1 percent).

For partial removal of sodium in any one region, the reactivity increment may be estimated from the reactivity increment for total removal from that region (Figure 2.5.4.2) by assuming that the reactivity increment varies linearly with sodium loss. This was calculated to be very nearly the case for a reactor of uniform fuel content having a composition and dimensions very nearly equal to those of the present 1000 MWe design.(3) In particular, this assumption can be applied to obtain regional and the total sodium temperature coefficients of reactivity, making use of the fact that a 1 F rise of the sodium temperature produces a 0.016 percent reduction of the sodium density. The over-all isothermal sodium temperature coefficient of reactivity obtained in this manner is very slightly negative ($-2.4\times10^{-7}\ \Delta k/^\circ F$). A perturbation calculation of the over-all sodium reactivity temperature coefficient gave a very slightly positive value ($+1.5\times10^{-7}\ \Delta k/^\circ F$). Both of these results represent an essentially zero isothermal sodium coefficient.* (With orificing of channel coolant flow corresponding to the radial power distribution, the isothermal sodium temperature coefficient is directly applicable to the calculation of the power coefficient of reactivity due to sodium thermal expansion.) The sodium temperature coefficient of reactivity for the central four-ninths of the core volume, corresponding to the peak of curve A in Figure 2.5.4.2, is $+2.2\times10^{-6}\ \Delta k/^\circ F$.

The maximum reactivity increment due to partial sodium voiding in the power-flattened core, represented by the peak of curve A in Figure 2.5.4.2, is about 0.3 percent higher than the corresponding value computed for a nonpower-flattened core.(3) The core composition and geometry of the nonflattened core closely paralleled those of the present 1000-MWe design.

*A perturbation calculation of the reactivity change due to removal of 10 percent of the sodium from the reactor volume corresponding to the peak of curve C of Figure 2.5.4.2 gave slightly less than 10 percent of the reactivity value at the peak of curve C.

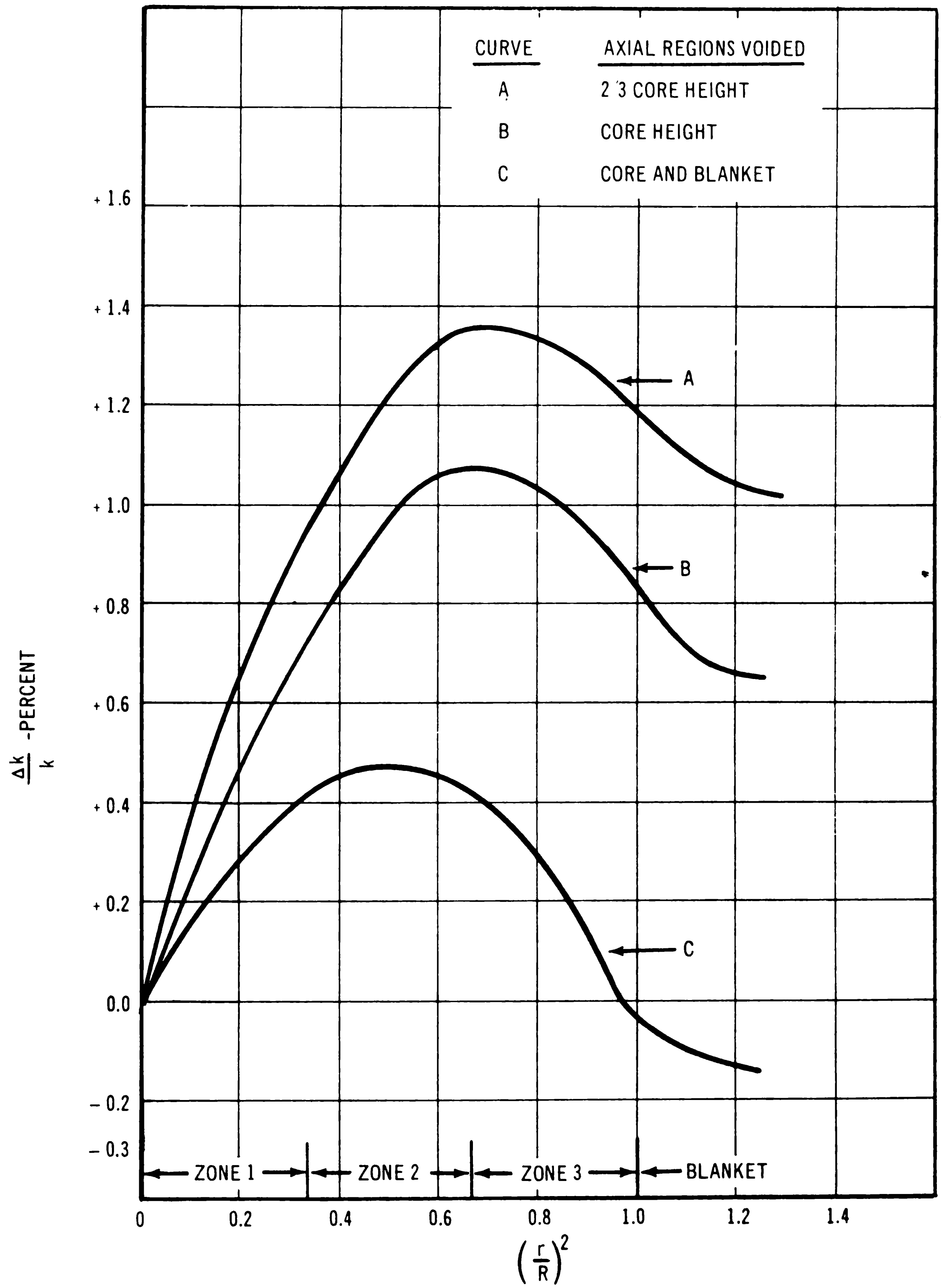

Figure 2.5.4.2. Reactivite Effects of Partial Sodium Voiding

2.5.4 REACTI :OEFFICIENTS AND KINETIC PARAMETERS (Continued)

2.5.4.2 Reactivity Effects Due to Sodium Loss (Continued)

The reactivity change due to losing sodium from the top axial blanket only may be estimated from the curves of Figure 2.5.4.2 by taking one-half of the difference between curves C and B at the radial position within which the top axial blanket is voided. This procedure[4] has been found to give a good estimate from other computations. For example, if sodium is lost from the entire radius of the top axial blanket, the reactivity change is -0.4 percent. This procedure may be extended, using all three curves, to evaluate the reactivity change as sodium voidage proceeds downward from the top blanket. For example, sodium loss down to the bottom one-third of the core height gives a reactivity change equal to one-half the sum of the values of curves A and C for the radius within which sodium is voided. For such sodium voidage over the entire radius, the reactivity change is +0.43 percent.

Loss of sodium throughout the core has been estimated to result in a 20 percent reduction of the Doppler coefficient. This is due to the shift of the neutron spectrum to higher energies (away from the region of strong U-238 resonances below 10 Kev) due to loss of the sodium slowing down cross section.

Three considerations are not taken into account in the curves in Figure 2.5.4.2 and the sodium coefficients obtained from these curves. One pertains to recent analysis of the cross section information which indicates that the sodium capture cross section in the 2.8 Kev resonance is highly overestimated. This is discussed in Section 3.6, and the indicated correction would reduce the computed reactivity change due to total loss of sodium from the entire reactor by 0.56 percent. The other considerations involve the beryllium (n, 2n) reaction and the overlapping of low energy Pu-239 resonances by U-238 resonances which would increase the computed reactivity change due to total sodium loss by 0.08 percent and 0.10 percent, respectively. The sum of these three effects would alter the reactivity change due to total sodium loss from the reactor from -0.15 percent to -0.53 percent. The peak reactivity change due to sodium voidage in the center four-ninths of the core volume would also be reduced from 1.36 percent to an estimated 1.16 percent (3.5 dollars).

2.5.4.3 Other Reactivity Effects

The principal delayed (long time constant) temperature coefficient of reactivity is that due to radial expansion of the core accompanying heating of the support structure. Radial core expansion decreases reactivity by reducing the concentrations of fuel and steel isotopes in the core. It produces a very small compensating positive reactivity by increasing the core radius and the volume fraction of sodium in the core, and decreasing the BeO volume fraction. Axial expansion of the steel clad upon raising the clad temperature reduces the steel content of the core and produces a small reactivity effect.

2.5.4 REACTIVITY COEFFICIENTS AND KINETIC PARAMETERS (Continued)

2.5.4.3 Other Reactivity Effects

In addition to theDoppler effect, a prompt negative reactivity coefficient results from thermal axial expansion of the fuel when the fuel temperature is raised. Radial thermal expansion of the fuel may stretch the clad and, hence, produce a reactivity effect by reducing the volume of the sodium channels. As seen in Tables 2.5.4.4(a) and (b), the radial fuel expansion is considerably smaller than the Doppler effect. Also the reliability of expansion coefficients with oxide fuel is highly uncertain.

The effect on reactivity of fuel rod bowing is expected to be small although no specific calculations have been made to verify this. Fuel rod bowing occurs when there is a substantial radial temperature gradient across a rod.

The flattened power distribution of the proposed 1000-MWe design will minimize the radial temperature gradients except near the radial periphery of the core where the power decreases rapidly with radius. A significant temperature gradient in this region is experienced only over the relatively-short (2-foot) length of active core fuel. The mechanical effect of bowing would be to spread the peripheral fuel rods slightly further apart mostly over the upper portion of the reactor.

Another source of reactivity perturbations due to geometric distortion is "dishing." This term is used to describe the vertical sagging of the inner portion of the core due to the weight of the fuel and lower support plate. The amount of dishing will depend on the coolant flow condition, being less at full flow than at no flow. The effect of changes in coolant flow will be small, however, due to the low pressure drop across the reactor (~25 psi across the entire axial length), hence the dishing effect is expected to be largely a static condition. The mechanical effect of dishing would be to spread the rods further apart over the lower portion of the reactor and squeeze them together over the upper part.

2.5.4.4 Summary of Temperature and Power Coefficients

Table 2.5.4.4(a) lists temperature coefficients of reactivity due to the Doppler effect and thermal expansion of various core components and structure. The corresponding power coefficients are listed in Table 2.5.4.4(b). It is seen that the prompt negative power coefficient due to the Doppler effect dominates completely the delayed power coefficients, in addition the isothermal (negative) temperature coefficient due to core radial expansion is comparable in magnitude to the Doppler temperature coefficient. The prompt negative effect of uniform fuel axial expansion does not have to be relied on as an essential core safety factor.

2.5.4.5 Neutron Lifetime and Effective Delayed Neutron Fraction

These kinetics parameters were computed to have the following values:

$$l = 5.7 \times 10^{-7} \text{ sec}$$
$$\beta_{eff} = 3.3 \times 10^{-3}$$

TABLE 2.5.4.4(a)

REACTIVITY TEMPERATURE COEFFICIENTS

	$\Delta k/°F \times 10^6$*
Prompt	
Doppler Effect	~~-5.6~~ -3.5
Fuel Axial Expansion	~~-1.3~~ -0.8
Fuel Radial Expansion	+0.01
Delayed	
Sodium Density Reduction	+0.15
Steel Clad Axial Expansion	-0.22
BeO Axial Expansion	+0.90
Core Radial Expansion	-3.7
Total	~~-7.17~~ -6.67
Reduction of Fuel and Steel Atom Density** (-6.4×10^{-6})	
Increase of Core Radius** ($+0.6 \times 10^{-6}$)	
Increase of Sodium Volume** (-0.4×10^{-7})	
Reduction of BeO Atom Density** ($+2.1 \times 10^{-6}$)	

TABLE 2.5.4.4(b)

REACTIVITY POWER COEFFICIENTS

	Δk per $(\Delta P/P) \times 10^3$***
Prompt	
Doppler Effect	-5.6
Fuel Axial Expansion	-1.3
Fuel Radial Expansion	+0.01
Delayed	
Sodium Density Reduction	+0.023
Steel Clad Axial Expansion	-0.033
BeO Axial Expansion	+0.14
Core Radial Expansion	-0.56
Over-All Power Coefficient	-7.3

---

*For a uniform 1 F temperature rise starting from full power, except Doppler effect which is for a 1 F rise of the average fuel temperature with the rise of the fuel temperature distribution following the power distribution.

**Components of the core radial expansion temperature coefficient.

***For a slow and small change of the power level near full power. The temperature change of the core support structure is assumed to be the same as that of the coolant in computing the core radial expansion power coefficient. A 1 percent increase of reactor power produces a 1.5 F rise of the average sodium temperature and 16 F rise of the average fuel temperature.

2.5 REACTOR PHYSICS DATA (Continued)

2.5.5 STARTUP AND TRANSITION TO EQUILIBRIUM CYCLE

The 1000-MWe study was based on an established equilibrium cycle in accordance with the assumptions listed in Section 2.5.1.1. The startup core and the period of transition to reach a near-equilibrium condition represent a significant portion of the reactor life and hence will affect the economics and the aspects of safety associated with the reactivity coefficients and control requirements for burnup.

It is considered that the startup core will be fueled with available plutonium that has been discharge from operating thermal reactors. Plutonium discharged from a BWR at a burnup around 15,000 MWD/T has a considerably lower Pu-240 content and a higher Pu-241 content than that of the equilibrium cycle. These changes of isotopic plutonium composition in the startup core would tend to lower the required core fissile inventory slightly, and would make a significant safety improvement in the sodium loss reactivity effects. (Decreasing Pu-240 content reduces the positive component of the sodium loss reactivity change, and replacing Pu-239 with Pu-241 produces an effect in the same direction.) The Doppler effect would not be expected to be very sensitive to the change of the plutonium isotopic composition.

Other composition changes likely in the startup core would be clean blankets (depleted uranium) and absence of fission products. These changes produce reactivity effects of opposite sign, the difference being a 0.4 percent reactivity decrease. The net effect on fissile contents and the Doppler coefficient would be small, with the sodium coefficient again being improved (rendered more negative) by both absence of fission products in the core and a clean blanket.

Control requirements for the startup core will depend upon the selected method of transition to equilibrium. One approach which eliminates unusual control requirements and achieves the near equilibrium cycle in the second core is to start with the same refueling schedule as projected for the equilibrium cycle. This requires that, on the average, the first core is subjected to only 60 percent of the burnup of the equilibrium cycle. The resultant penalty in the fabrication and reprocessing cost for the first core (see Section 2.9) would be compensated partially by a lower inventory (due mostly to the clean blanket condition at startup) and a slightly-higher breeding ratio (~0.04) due to a lower level of fission products during the first core lifetime.

The fissile material contents for the three zones of the first core would be selected to give an initial power distribution characteristic of the equilibrium core. The power peaking factor due to having fuel elements of different burnup in the same zone will be less during the early portion of the first core lifetime than for the equilibrium cycle. On the other hand, the power level in the core will have to be higher for the startup core to compensate the lower power generation in the blankets.

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1. Smith, D.C.G., "Rapid Startup and Shutdown of Fast Power Reactors." Paper presented at the Conference on Breeding, Economics and Safety in Large Fast Power Reactors, Argonne National Laboratory, September 10, 1963.

2. Mills, R. G., and Hausner, H. H., "Uranium Dioxide for Fuel Elements," Nucleonics, July 1957.

3. Greebler, P., "Recent Improvements in Calculations of Doppler and Sodium Loss Reactivity Effects for Large Fast Reactors," GEAP-4410 (to be issued). Paper presented at the Conference on Breeding, Economics and Safety in Large Fast Power Reactors, Argonne National Laboratory, October 7-10, 1963.

4. Cohen, K. P., et al., "Reactor Safety and Fuel Cycle Economics Considerations for Fast Reactors," GEAP-4414 (to be issued). Paper presented at the Conference on Breeding, Economics and Safety in Large Fast Power Reactor, Argonne National Laboratory, October 7-10, 1963.

2.6 SYSTEM DYNAMICS

2.6.1 NORMAL OPERATION

2.6.1.1 Power Level Control

2.6.1.2 Stability

2.6.1.3 Load Changes

2.6.2 ACCIDENTS

2.6.3 CONTROLLED ACCIDENTS

2.6.3.1 Reactivity Insertions at Full Power

2.6.3.2 Refueling Accidents

2.6.3.3 Loss of Cooling

2.6.4 UNCONTROLLED ACCIDENTS

2.6.4.1 Model for Maximum Hypothetical Accident

2.6.4.2 Calculation of Energy Release

2.6.4.3 Post-Accident Condition

2.6 SYSTEM DYNAMICS

Preliminary analyses of over-all dynamic characteristics for fast oxide breeder reactor cores were conducted prior to the design of the reference plant to generate design specifications for that plant. These over-all considerations are discussed in Section 3.2 under Safety. Subsequent specific analyses of performance characteristics for the reference plant are covered in this section under the headings Normal Operations, Controlled Accidents, and Uncontrolled Accidents.

2.6 SYSTEM DYNAMICS (Continued)

2.6.1 NORMAL OPERATION

The reference plant does not have inherent load-following (or opposing) characteristics, because the reactivity is negligibly affected by coolant temperature. Further, the primary system is isolated from the steam raising plant by a secondary coolant system, and there is a very large thermal capacitance within the primary cooled system. The control system has, therefore, been designed to provide automatic load-following capabilities for the plant. Preliminary evaluations indicate no inherent limitations of the plant design which would prevent a load-following ability typical of modern power plant practice, i. e., ability to accommodate step stages in power in the range 5 to 10 percent (this step change actually occuring in ~2 seconds, namely, time taken for valve operation) and ramp changes of the order of 0. 5 percent/second between hot standby conditions and full power.

2.6.1.1 Power Level Control

The secondary sodium and steam circuits and the control block diagram are shown in Figures 2.4.1.1 and 2.6.1.1, respectively. The philosophy of operation is to isolate the core as far as possible from any feedback effects due to the power transfer system while maintaining constant steam conditions at all power levels. This is an appropriate and feasible scheme of operation for this unit for the following reasons:

1. The core is automatically isolated to a large degree due to the secondary sodium loop and the large primary sodium reservoir. Inlet temperature to the core will change very slowly, even if no attempt is made to control it.

2. Coolant feedback effects are not large enough to be useful in giving any automatic load-following ability. They are minimized by maintainence of a constant inlet temperature.

3. The large primary tank is an important part of the structure of the plant and contains the whole primary sodium system. Changes in the temperature of these components are minimized (a desirable result) by maintaining the tank and the sodium in it at a fairly-constant temperature.

4. The secondary sodium system can provide the means of simultaneously maintaining steam conditions and reactor inlet temperature constant for all power levels except very low ones.

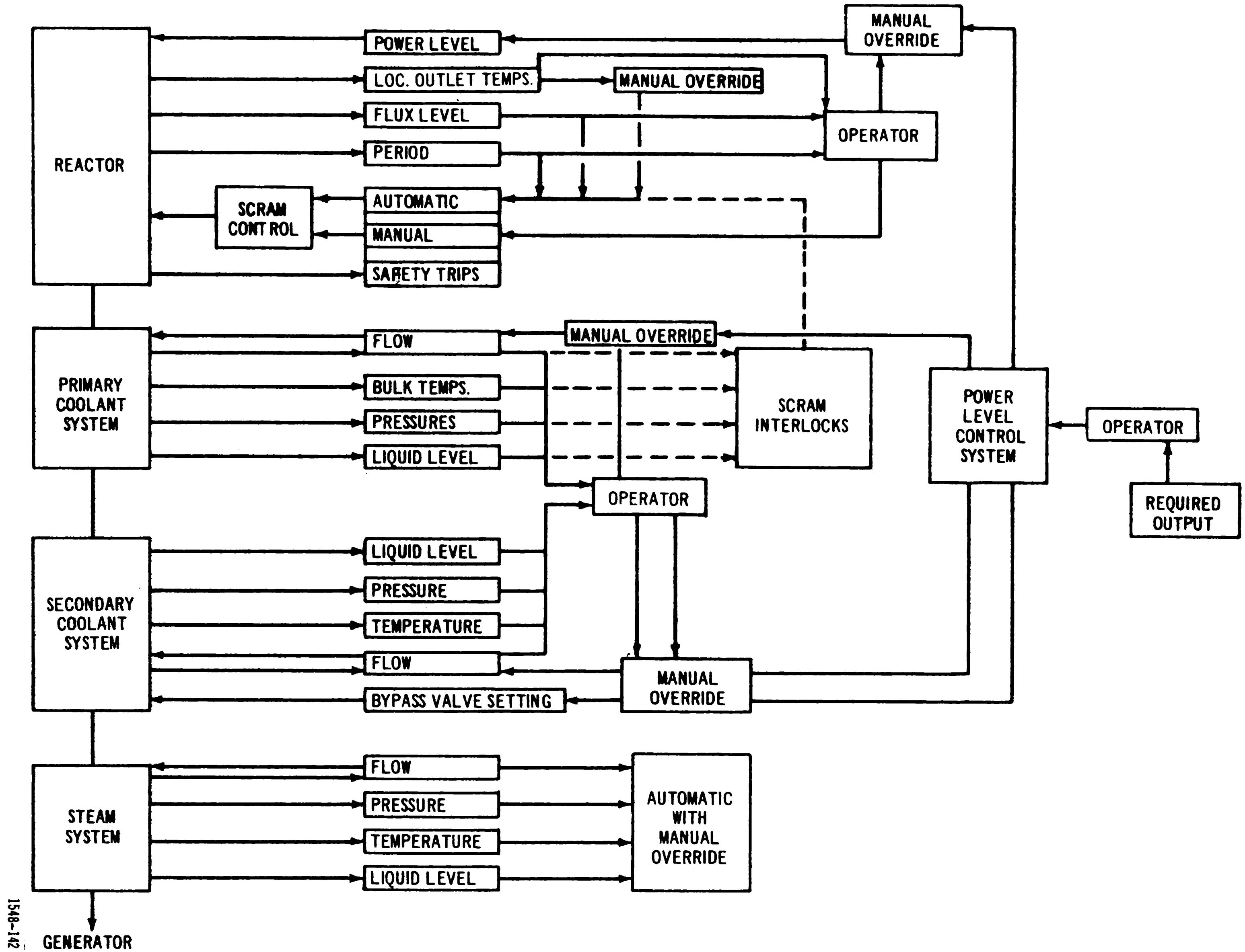

Figure 2.6.1.1. Control Block Diagram

2.6.1 NORMAL OPERATION

2.6.1.1 Power Level Control (continued)

The method used to maintain steam conditions constant is to vary the bypass flow in the secondary sodium circuit together with the total secondary flow.[1] The mean temperature of the secondary sodium in the intermediate heat exchanger can then be held constant for different power levels. Since the inlet core temperature is to be constant, the outlet temperature on the primary side of the intermediate heat exchanger must be constant and log mean ΔT across the intermediate heat exchanger is a function of core outlet temperature only. Thus, core outlet temperature must increase or decrease with the power level of the plant. A comparison of full-power and half-power conditions in the primary system is given in the following table:

	Core Inlet	Core Outlet	Percent Full Flow	Average Secondary Sodium Temperature
Full Power	800	1100	100	885
Half Power	800	1035	64	885

Observe that primary flow must also be varied, and reductions in power do not reduce the required primary flow proportionately.

To achieve smooth efficient changes of power level in this system where primary flow, control rod position, secondary sodium flow, secondary sodium bypass flow, and steam flow all must be adjusted, a preprogrammed power level control system will be used to co-ordinate the various adjustments required after the desired power level has been determined by the operator.

2.6.1.2 Stability

The stability of the reactor was analyzed to determine its ability to maintain the desired power level, once established, and its sensitivity to possible disturbances. The effects of coolant flow rate and inlet temperature variation were not considered.

A mathematical model of the dynamic reactivity effects of the reactor core has been set up and is described below. The reactor was considered independent of the coolant systems and power conversion equipment. This model was found to be extremely stable for the expected values of dynamic reactivity effects. No attempt was made to determine the "stability limit" for any given parameter. With the knowledge gained from Experimental Breeder Reactor-, Sodium Reactor Equipment-, and Fermi-plant experience it is felt that a stable system of the 1000-MWe type can be designed.

The mathematical model that was used for the analysis is shown in Figure 2.6.1.2(a). The symbol "s" is the Laplace transform variable. The numerical expressions given below were used in the block diagram of Figure 2.6.1.2.(a). The sodium reactivity effect was taken for the core only, and neglected the negative contribution of coolant temperature rises in the blanket. Thus, a positive coefficient was used, rather than a zero one.

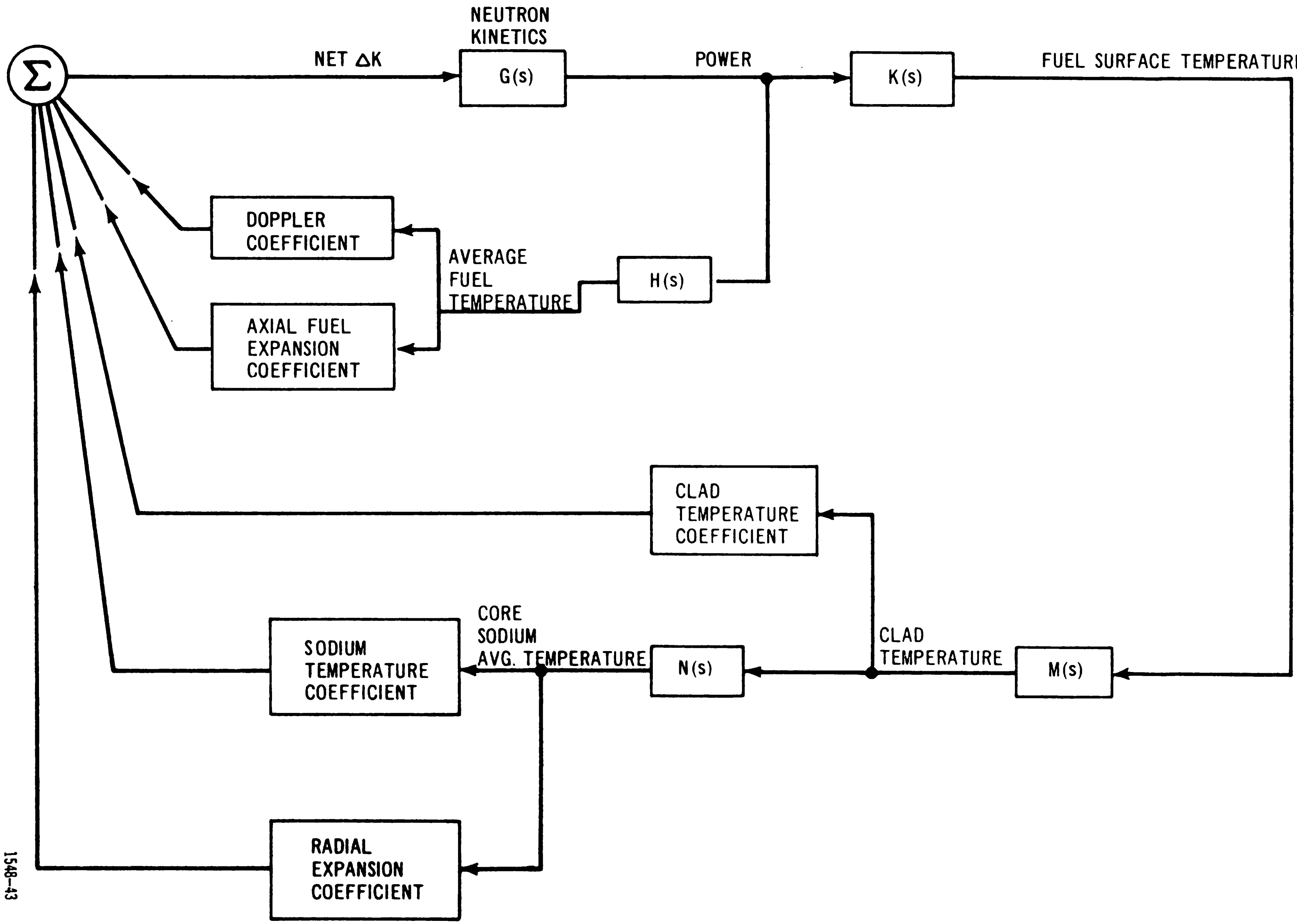

Figure 2.6.1.2(a). **Block Diagram for Stability Model**

2.6.1 NORMAL OPERATION

2.6.1.2 Stability (continued)

Neutron Kinetics:

$$G(s) = \frac{0.00063 + 0.2840\ s + 1.495\ s^2 + s^3}{0.0822\ s + 1.174\ s^2 + 1.0\ s^3 + 0.00017\ s^4}$$

Power-to-Fuel Surface Temperature:

$$K(s) = \frac{1 + 0.4805\ s}{1 + 2.439\ s + 0.7807\ s^2} = \frac{1 + 0.4805\ s}{(1 + 2.060\ s)(1 + 0.379\ s)}$$

Power-to-Fuel Average Temperature:

$$H(s) = \frac{1 + 0.7874\ s + 0.1124\ s^2 + 0.0019\ s^3}{1 + 2.569\ s + 1.099\ s^2 + 0.1059\ s^3 + 0.0014\ s^4}$$

$$= \frac{1 + 0.7874\ s + 0.1124\ s^2 + 0.0019\ s^3}{(1 + 2.06\ s)(1 + 0.379\ s)(1 + 0.114\ s)(1 + 0.016\ s)}$$

Fuel Surface-To-Clad Temperature:

$$M(s) = \frac{1}{1 + 0.02\ s}$$

Clad-to-Sodium Temperature:

$$N(s) = \frac{1}{1 + 0.2\ s}$$

Doppler Coefficient	= -1.39 cents/percent Power
Fuel Axial Expansion Coefficient	= -0.32 cent/percent Power
Clad Temperature Coefficient	= +0.0138 cent/percent Power
Sodium Temperature Coefficient	= +0.0057 cent/percent Power
Radial Expansion Coefficient	= -0.0043 cent/percent Power

Note that the dominant fuel time constant of 2 seconds will tend to suppress any resonance effects above approximately 0.1 cycle per second. It does not appear that there will be any reactivity effects of appreciable magnitude above this frequency. In the low frequency region where the fuel does not obliterate fission power oscillations, careful attention to design details will prevent appreciable positive reactivity effects, or reactivity effects with undesirable phase shifts.

2.6.1 NORMAL OPERATION

2.6.1.2 Stability (continued)

The model represents the core as a "point" system. It is felt that this representation is justified since the core transit time (0.2 second) is short compared to the dominant fuel element time constant (2.0 seconds). Thus, the sodium coolant temperature essentially remains in thermal equilibrium with fuel heat flux changes. The heat capacitance effect in the coolant can then be approximated as shown in Figure 2.6.1.2(a). If there were large reactivity effects associated with the sodium temperature (such that the high frequency attenuation of power oscillation by the fuel element would be cancelled), then this model might not be applicable and a "distributed" model might be required.

The parameter values from which the constants used in the model were determined are listed in Table 2.6.1.2. A comparison of the curves used in these calculations with the values finally calculated for the reference design [see Section 2.5, Tables 2.5.4.4(a) and (b)] shows that they are conservative. The reactor is actually more stable than computations shown in this section indicate.

The results of the frequency response calculations are shown in Figures 2.6.1.2(b) through (d). The response of fission power to a sinusoidal oscillation in Δk as a function of frequency for the following two conditions can be seen in Figure 2.6.1.2(b).

1. Neutron kinetics with Doppler, fuel expansion, sodium, clad, and structure feedback effects
2. Neutron kinetics with Doppler and fuel expansion feedback only.

The addition of the positive feedback effects in item 1 is seen to produce an insignificant effect. The effect of reducing the negative feedback by omitting the fuel axial expansion reactivity effect is illustrated in Figure 2.6.1.2(c). Again, it is seen to be quite stable, and only slightly different from Figure 2.6.1.2(b). The phase angle vs frequency corresponding to the curves of Figures 2.6.1.2(b) and 2.7.1.2(c) can be seen in Figure 2.6.1.2(d). Since the effects of adding the small positive reactivity feedbacks are insignificant, just the effect of adding the axial fuel expansion reactivity coefficient is shown in Figure 2.6.1.2(d). At high frequency, the frequency response approaches that of the zero power neutron kinetics. The low-frequency response approaches that which would be obtained for the steady-state change in power for a given external reactivity change.

2.6.1.3 Load Changes

The reactor will have virtually no inherent load-following ability, as discussed in Section 2.6.1.1. Desired changes in load must be communicated to the power level control system, either by the operator or possibly automatically. No analysis of the permissible rates of change of power level has been made, but it would appear that the following factors must be considered:

1. Maximum allowable rate of change of coolant outlet temperature.
2. Maximum allowable rate of change of reactivity (control rod movement).

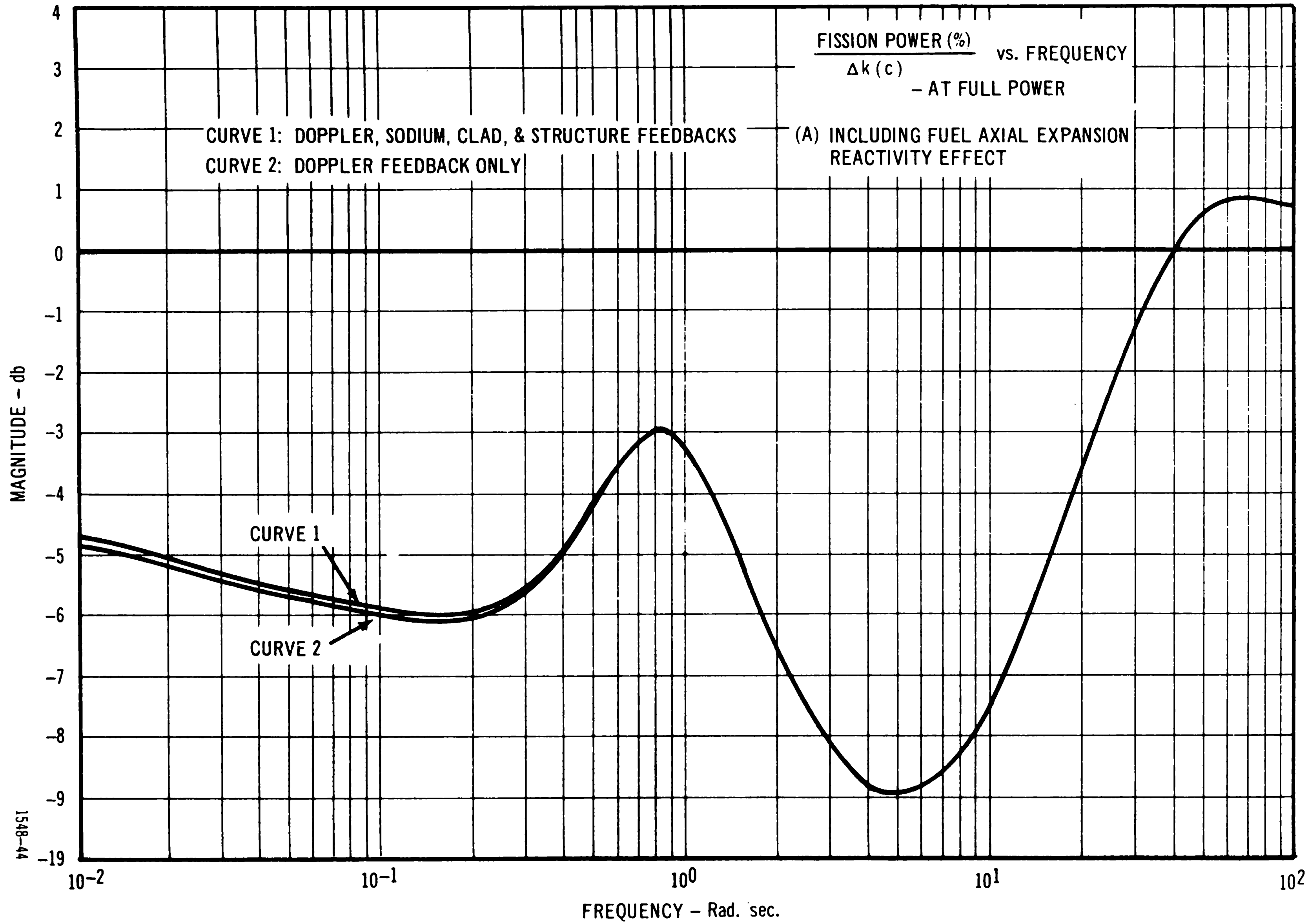

Figure 2.6.1.2(b). Closed Loop Frequency Response (A)

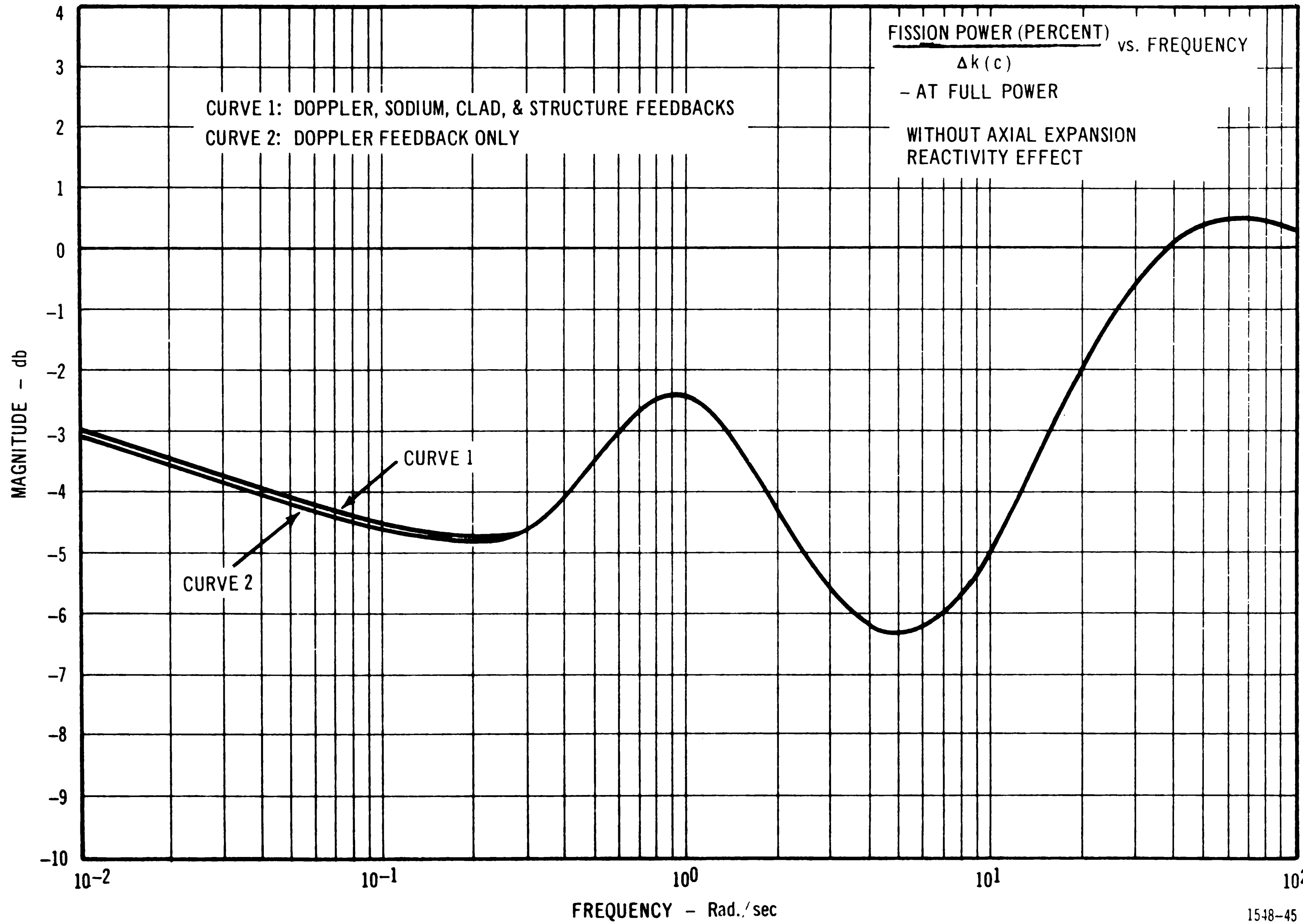

Figure 2.6.1.2(c). Closed Loop Frequency Response (B)

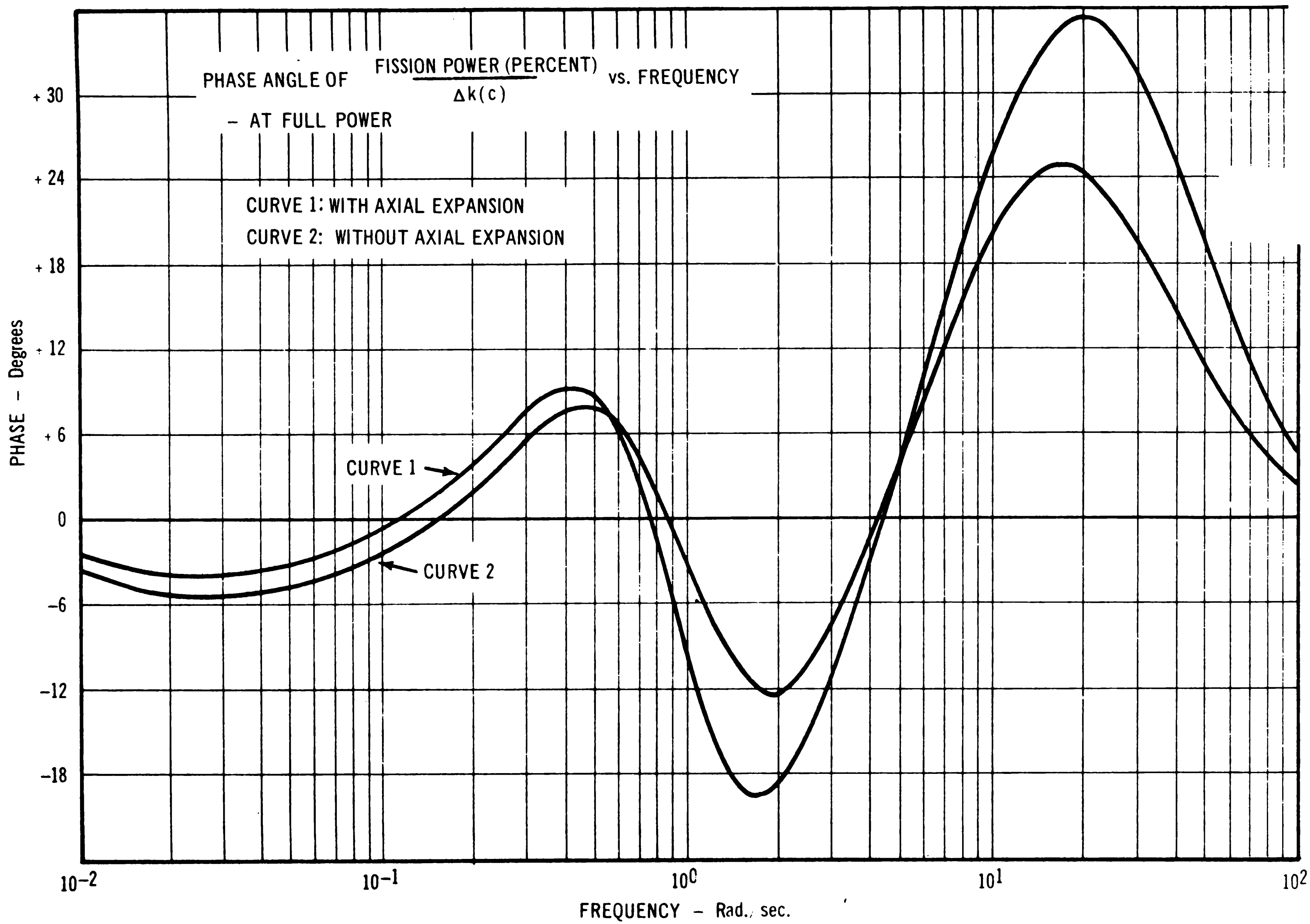

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Figure 2.6.1.2(d). Closed Loop Frequency Response (Phase Angle)

TABLE 2.6.1.2

REACTOR PARAMETER VALUES

NEUTRON KINETICS

Three delay groups:

β_i	λ_L
0.252	0.0257
0.591	0.193
0.157	0.1276

Prompt Lifetime $= \frac{l}{\beta} = 0.000171$ second

Delayed Neutron Fraction $= \beta = 0.0035$

FUEL ELEMENT

Oxide Heat Capacity	=	0.08 Btu/lb-°F
Oxide Density	=	575 lb/ft^3
Oxide Conductivity		1.55 Btu/hr-ft-°F
Clad Conductivity	=	12.0 Btu/hr-ft-°F
Clad Heat Capacity	=	0.12 Btu/lb-°F
Clad Density		500 lb/ft^3
Clad Thickness	=	0.015 inch
Oxide Diameter	=	0.220 inch
Oxide-Clad Gap Coefficient	=	1500 Btu/hr-ft^2-°F
Surface Coefficient	=	10,000 Btu/hr-ft^2-°F
Average Oxide Temperature	=	2315 F
Average Clad Temperature	=	1030 F

$\left(T \frac{dk}{dT}\right)_{\text{Doppler}} = -1 \times 10^{-2}$ (T in °K or °R) (-1×10^{2})

$\left(\frac{dk}{dT}\right)_{\text{Axial Expansion}} = -0.82 \times 10^{-6}/°F$ $(-0.8 \times 10^{-6}/°F)$

$\left(\frac{dk}{dT}\right)_{\text{Clad}} = +0.21 \times 10^{-6}/°F$ $(-0.22 \times 10^{-6}/°F)$

TABLE 2.6.1.2 (continued)

PARAMETER VALUES USED FOR STABILITY STUDY

SODIUM COOLANT

Average Sodium Temperature = 950 F
Coolant Transit Time Through Core = 0.18 second

$\left(\frac{dk}{dT}\right)_{Sodium} = +1.33 \times 10^{-6}/°F \qquad (+0.15 \times 10^{-6}/°F)$

STRUCTURE

Average Structure Temperature = 950 F

$\left(\frac{dk}{dT}\right)_{Radial} = -1 \times 10^{-6}/°F \qquad (-3.7 \times 10^{-6}/°F)$

Values in parenthesis indicate final data as presented in Table 2.5.4.4(a) which were not available until after completion of dynamic studies.

2.6.1 NORMAL OPERATION

2.6.1.3 Load Changes (continued)

The rate of change of coolant temperature will be limited automatically by the design itself because the heat fluxes from the fuel will not change as rapidly as the power level in the core. Since the time constant of the fuel is about 2 seconds, it probably would be impossible to change the coolant outlet temperature more than 100 F/sec during normal operation. This may be a greater change than the heat exchanges can tolerate. The rate at which reactivity may be changed is dependent on the ability of the reactor to withstand accidental reactivity insertions. It will be shown in Section 2.6.2 that the ejection of two control rods simultaneously is a minor accident for this core and, therefore, it is probable that the thermal stresses in the heat exchanger, rather than the control rod withdrawal rate, will limit the rate of power change.

Maximum rates of load increase may be greater than the rate of power change, due to the stored energy in the system, if some temporary reduction of reactor inlet temperature and degradation of steam conditions are allowed. It is estimated that the primary and secondary sodium systems contain 4×10^6 pounds of sodium, at about 800 F, from which energy could be withdrawn at the rate of 1200 MWt at the expense of only a 1 F/sec drop in average temperature.

With a suitable system for accomplishing and co-ordinating the adjustments required to change power level, the fast breeder reactor ~~breeding ratio~~ should be capable of rapid changes in power level, limited only by the thermal stresses in the coolant circuit components, and should also be capable of even more rapid changes in load with subsequent adjustment of the power level to match. Excellent load-following ability will be available if an automatic power level control is provided, with controlling inputs from the steam system and generator.

2.6 SYSTEM DYNAMICS (continued)

2.6.2 ACCIDENTS

The dynamic characteristics of the core and the effectiveness of the design of the mechanical control system in working in conjunction with these characteristics provides the assurance that credible accidents can be controlled and do not escalate into uncontrolled ones. We shall assess here the dynamic characteristics of the core under accident conditions and evaluate the effectiveness of the control system and over-all reactor design in surviving such accidents. The discussion excludes from consideration the far more frequent operating incidents, which are minor perturbations which can easily be accommodated either through the inherent characteristics of the reactor or by normal responses of the control system. Such a discussion would be appropriate to a later detailed design. Attention is focused at this point on the response of the reference design to serious accidents.

2.6 SYSTEM DYNAMICS (continued)

2.6.3 CONTROLLED ACCIDENTS

A controlled accident is defined as one in which the reactor is finally shut down by design procedures; that is, by the control system with the aid of the excursion-limiting Doppler coefficient. It may involve some degree of core damage, particularly if some loss of cooling occurs, but it does not produce core disassembly, and is not a hazard to personnel or to any part of the plant outside the primary coolant system.

To lead to an accident condition, large reactivity changes need to be introduced rapidly (see Section 3.2 for discussion). Possible sources of major reactivity changes (without concern at this point for this probability) are control rod expulsion, fuel assembly drop-in during refueling, fuel slumping within the clad, and loss of coolant. Distortion of the core due to thermal and hydraulic effects should be prevented largely by the mechanical design, and its effects on reactivity should be small owing to the pancake shape of the core. The introduction of a spectrum softening material into the core, except possibly in very large amounts, would reduce reactivity. Loss of BeO in large volume would increase reactivity slightly, but this is not a credible event unless the reactor as a whole has first been badly disrupted. Such an occurrence would be a consequence of an uncontrolled accident, which is discussed in the next section.

In general, if the reactor is shut down without gross melting or collapse of the core, the accident will have been controlled preventing the possibility of a core compaction and explosive energy release. In order to prevent melting or collapse of the core, coolant loss, if any, must be limited to a fraction of the core. Fuel element clad rupture, which will tend to increase sodium expulsion due to the more rapid heating of the coolant in contact with the fuel, must be prevented during an excursion, except as an isolated occurrence in one or two subassemblies (fuel element rupture following loss of coolant in a limited region of the core is not in itself considered an uncontrolled accident in this context). The criterion for prevention of fuel element rupture, if cooling is maintained, is postulated to be a maximum fuel center temperature of 6500 F, to keep internal pressure (particularly fuel vapor pressure) within limits that can be contained by the cladding (see Section 3.2).

The analysis of controlled accidents presumes that the control system operates properly and scrams the reactor. Since the Doppler effect does not shut down the reactor, failure of the control system (or systems) to scram will result in destruction of the reactor for large, rapid reactivity insertions. However, there is a magnitude of reactivity insertion for which the reactor will assume a new steady-state power level which will cause no immediate damage, although it may be much too high to tolerate over an extended period of time. For example, if the reactor could operate for a significant time with a peak fuel temperature in the vicinity of 6000 F (including a hot spot factor of 50 percent) which may in fact be possible, and if the equivalent thermal conductivity of the fuel were about the same as in the solid state, the power level would be about 20 percent higher than normal full power, and the sodium outlet temperature would be 1160 F instead of 1100 F. The limiting factor would probably be the ability to dispose of the excess heat, although the heat capacity

2.6 SYSTEM DYNAMICS

2.6.3 CONTROLLED ACCIDENTS (Continued)

of the large volume of primary sodium will alleviate the problem. The Doppler feedback due to the higher temperatures would be about 50 cents. Thus, a reactivity insertion of 50 cents could be tolerated at rated power even with a failure to scram, if some other means of correcting the situation could be put into effect within a reasonable length of time. The time available would depend on the size of the initial reactivity insertion, but it easily could be that operation at 6000 F fuel temperature could be tolerated for some tens of minutes.

In the controlled accidents which will be considered here, the reactivity insertions considered are all over 1 dollar, since this places the most stringent requirements on the control system and reactor design.

2.6.3.1 Reactivity Insertions at Full Power

For short times before the control rods can operate, the reactor depends chiefly on the Doppler coefficient to limit the severity of accidents. Since the Doppler coefficient is approximately inversely proportional to the absolute fuel temperature, the full power operating condition will correspond to the lowest Doppler coefficient in normal operation; furthermore, there is a smaller capability of energy absorption before core damage will occur. Consequently at full power the reactor is most vulnerable to an accidental reactivity insertion.

An important variable in accident calculations is the speed with which the control system can respond and shut down the reactor. In these calculations it was assumed that the insertion of negative reactivity due to scram would commence 200 milliseconds after the start of the accident which normally would be detected by the flux monitors during the prompt power burst, and that scram would be complete 600 milliseconds after the start of the accident. *

Among the possible sources of reactivity mentioned in Section 2.6.2, control rod expulsion is a possible source of a large reactivity increment that might initiate an accident. Fuel slumping and local coolant loss are more likely to be accompaniments of an accident rather than the cause. An analysis was performed, using the FORE[2] code, of an accident where two control rods were ejected within about 10 milliseconds. This is felt to be an extremely-conservative assumption for the following reasons:

1. The control rods will be driven individually.
2. The pressure drop through the core, about 25 psi, is insufficient to lift the control rod assembly weight.
3. The control rod drive system should have no mechanism for energy storage with the possible exception of a spring that can act in the downward direction only. There are stops to prevent control rods from falling through the core.

*In actual operation, scram may be programmed in some way to minimize unnecessary shutdowns, and reduce thermal shock. For example, an initial reduction of power followed after a short delay by a complete scram might be used in some circumstances. These refinements were not considered in the analysis.

2.6.3 CONTROLLED ACCIDENTS

2.6.3.1 Reactivity Insertions at Full Power (Continued)

The calculated maximum worth of a control rod is about 50 cents. An additional 10 percent was added to this, giving a total of 1.1 dollar reactivity insertion for the two rods. The reactivity feedbacks included in the calculation Doppler, coolant density changes, steel density changes and core radial expansion. No credit was taken for longitudinal expansion of the fuel, which is a prompt negative reactivity, due to the uncertainty of its magnitude and dependability.

For the purpose of defining the reactivity insertion which would produce fuel damage, a 1.2 local peaking factor was used, in addition to the gross radial and axial flux peaking factors. This corresponds to an allowance of 1.09 for the variation in fissile plutonium content between a new fuel assembly and one which has accumulated approximately 50,000 MWD T irradiation, plus an additional 1.1 for other nonrandom factors. The composite ~~total~~ local peaking factor used for core design was about 1.5. This was not used here because it includes many random factors, and it was deemed a negligible probability that these would be combined at the worst possible location at precisely the time when an accident occurred.

Figure 2.6.3.1(a) shows the power and reactivity vs. time for the two-rod expulsion accident. It can be seen that the Doppler coefficient brings the net reactivity below 1 dollar, in 3 milliseconds, and the power level falls quickly to about seven times full power, falling slowly thereafter until scram begins at 0.2 second. Figure 2.6.3.1(b) shows the average fuel temperature, the nominal peak fuel temperature (gross radial and axial power peaking included), and the "local" peak fuel temperature which includes the 1.2 local power peaking factor. The rise in fuel temperatures can be observed to be relatively mild, and only the local peaking causes temperatures to reach the melting point. No fuel is near the failure point of 6500 F, and also, the clad and coolant temperatures are not too high [see Figure 2.6.3.1(c)]. The coolant temperature to produce bulk boiling would be about 1850 F, since the pressure at the core outlet is 35-40 psia. This can be concluded to be a minor accident.

However, there are additional sources of reactivity to be considered, such as fuel slumping or localized coolant losses (due to local blockage or distortion) which might compound the accident. The total reactivity available from fuel slumping within the clad is approximately ~~4.5~~ 2.9 dollars. Only fuel which is molten has a significant probability of slumping. Although some fuel in the center of the pellets reached the melting point, in no location was there enough energy to completely supply the heat of fusion, and where the center of the pellet did reach the melting temperature, most of the pellet is still below melting. Therefore, fuel slumping could not be a significant factor in this accident, unless other events first increased the transient energy release.

If a local flow blockage should occur in the core because of fuel distortion during the course of the above accident the resultant overheating of the coolant would cause boiling accompanied by sodium expulsion. If this were to happen in five subassemblies at once in a location where sodium loss gave the most positive effect, 15-20 cents might be added. *

*The analyses discussed in Section 3.2.4.4 indicate that there will be a delay of several tenths of a second before this occurs, and the reactor will have been scrammed. The result therefore would be limited to damage to the subassemblies from which sodium was lost.

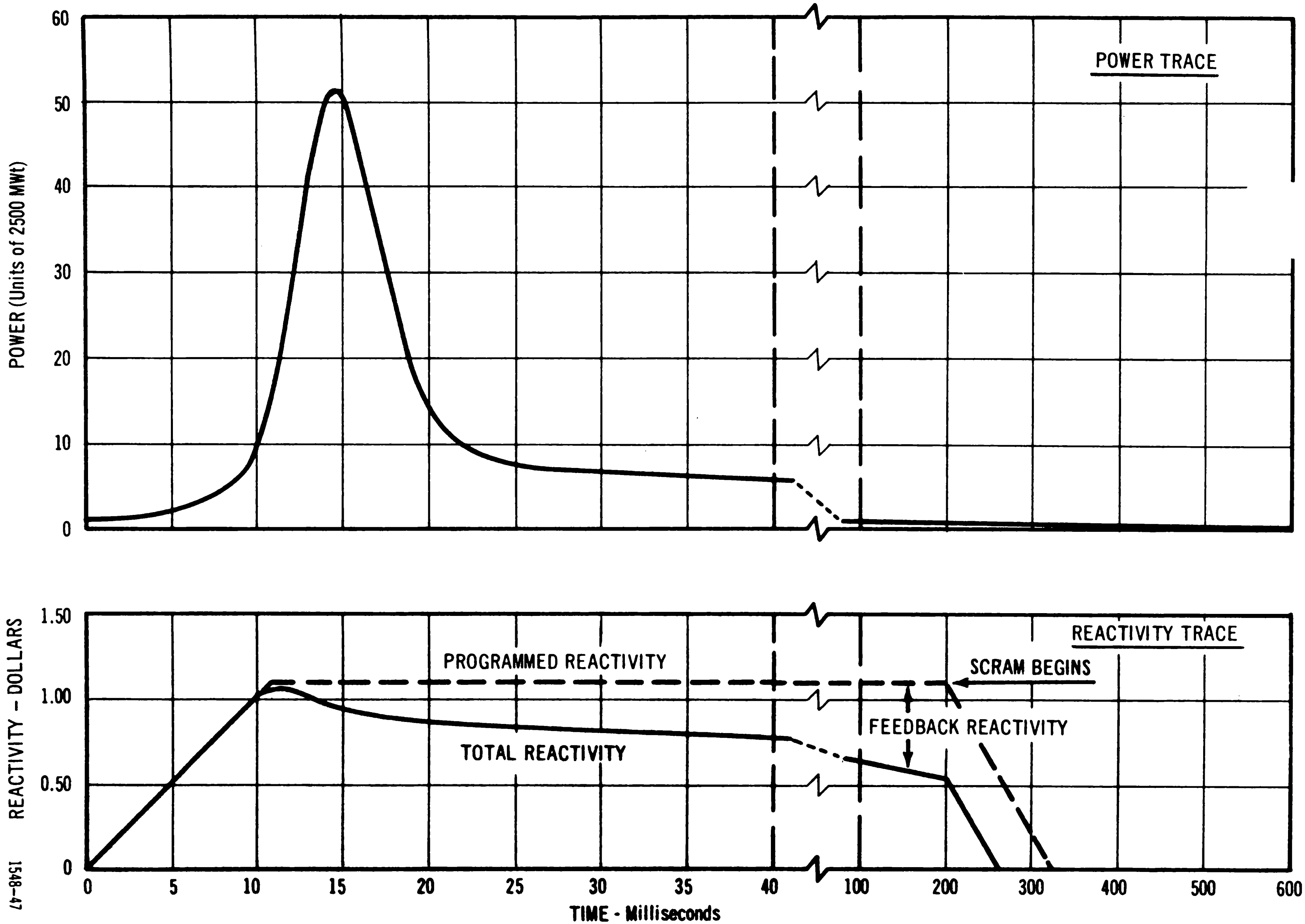

Figure 2.6.3.1(a). Control Rod Expulsion Accident - Power Transient

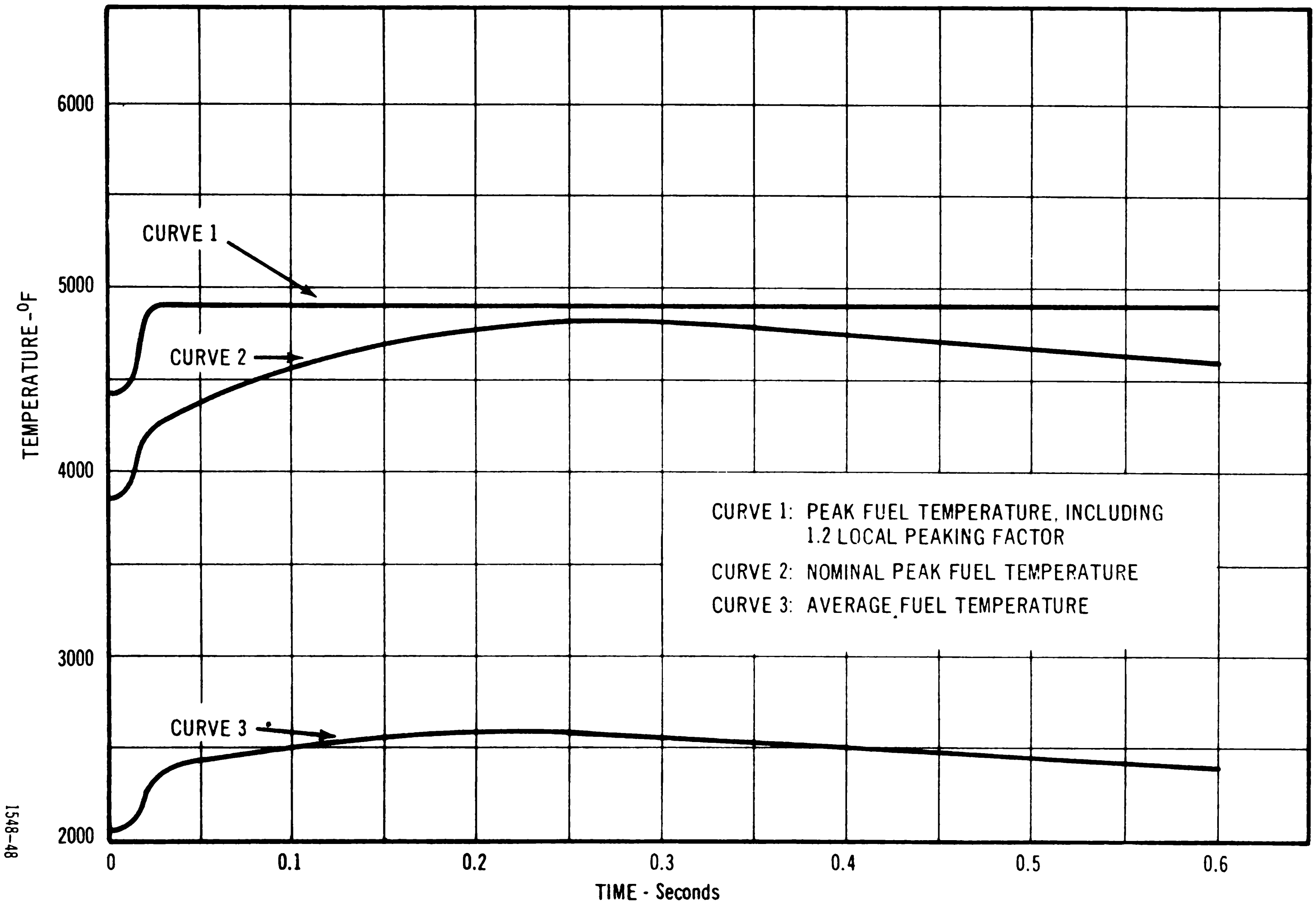

Figure 2.6.3.1(b). Control Rod Expulsion Accident - Fuel Temperature

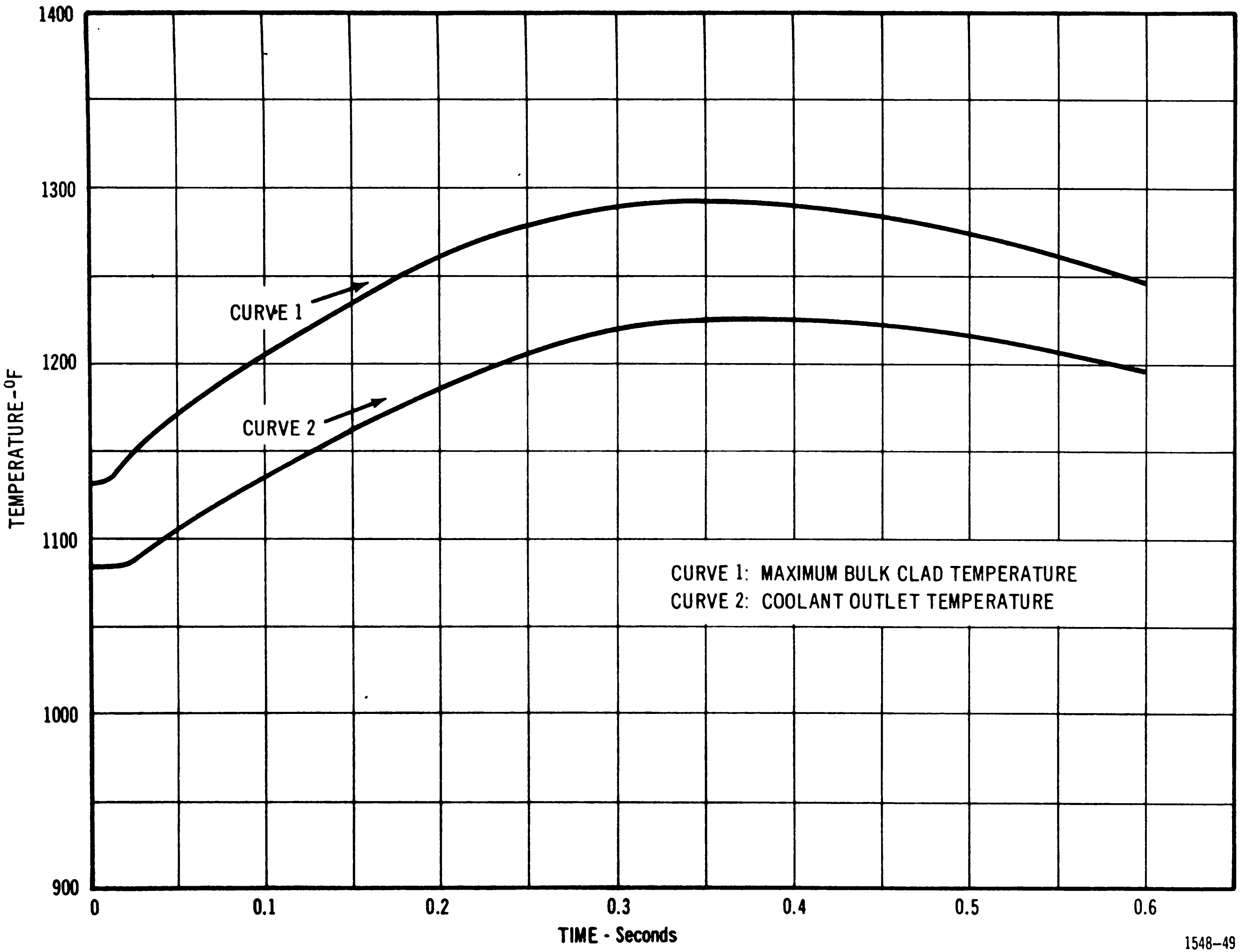

Figure 2.6.3.1(c). Control Rod Expulsion Accident - Coolant Temperatures

2.6.3 CONTROLLED ACCIDENTS

2.6.3.1 Reactivity Insertions at Full Power (Continued)

A calculation was performed to determine the maximum amount of reactivity that could be inserted without core damage. It was assumed that all of the reactivity was inserted at the beginning of the accident at 100 dollars/sec. The result showed that 1.75 dollars could be tolerated. The power, reactivity, and temperatures are shown in Figures 2.6.3.1(d) and (e). The maximum local fuel temperature is close to the estimated failure point of 6500 F. The maximum clad surface temperature is 1520 F, occurring in the core, which is close to the maximum bulk coolant outlet temperature. This then shows that over and above the two-rod expulsion at least an additional 65 cents could be tolerated from other sources. In the case where effects such as fuel slumping contribute reactivity during the accident, it will be at a slower rate and a later time than were assumed for this problem. In addition, some fuel failure might occur without causing the accident to become uncontrolled. If these factors are taken into consideration, the total reactivity insertion that might occur at full power and still allow a more or less normal shutdown will be significantly more than 1.75 dollars.

2.6.3.2 Refueling Accidents

During refueling the potential for an accident is in some ways greater, due to the complicated and difficult nature of the refueling operation. For example, one can imagine fuel assemblies falling into the core under conditions where the reactor is just subcritical due to a series of previous errors. Calculations were performed to determine the magnitude of reactivity insertion that could be withstood in the cold condition, if the reactor were just critical.

Since the initial temperatures are 400 F, which is the temperature at which the sodium is held during refueling, the Doppler coefficient is large compared to that at the full-power condition. The power was assumed to be at 10^{-9} times full power, which results in a greater time delay before the transient temperature rise becomes large enough to supply significant Doppler feedback.

Since the flow is only a fraction of normal, and the fuel temperature is very low initially, the limitation in this type of accident is coolant boiling, rather than fuel element rupture due to internal pressure. The flow rate becomes a key variable. In this type of operation, 10 percent of full flow is specified, which would give a 30 F ΔT across the core if the shutdown heat were 1 percent of full power. The results showed that a minimum of 3.5 dollars could be inserted with no resultant damage. Power, reactivity, and temperatures are shown in Figures 2.6.3.2(a) through 2.6.3.2(d). Maximum fuel temperatures do not reach melting. The problem was not run to the point where sodium temperatures began to decline, but an estimate of 1560 F for the maximum sodium temperature was made by assuming that at 0.7 second flow stopped, and the stagnant sodium in the core came to equilibrium temperature with the fuel and clad. This estimate plainly gives a conservatively high answer.

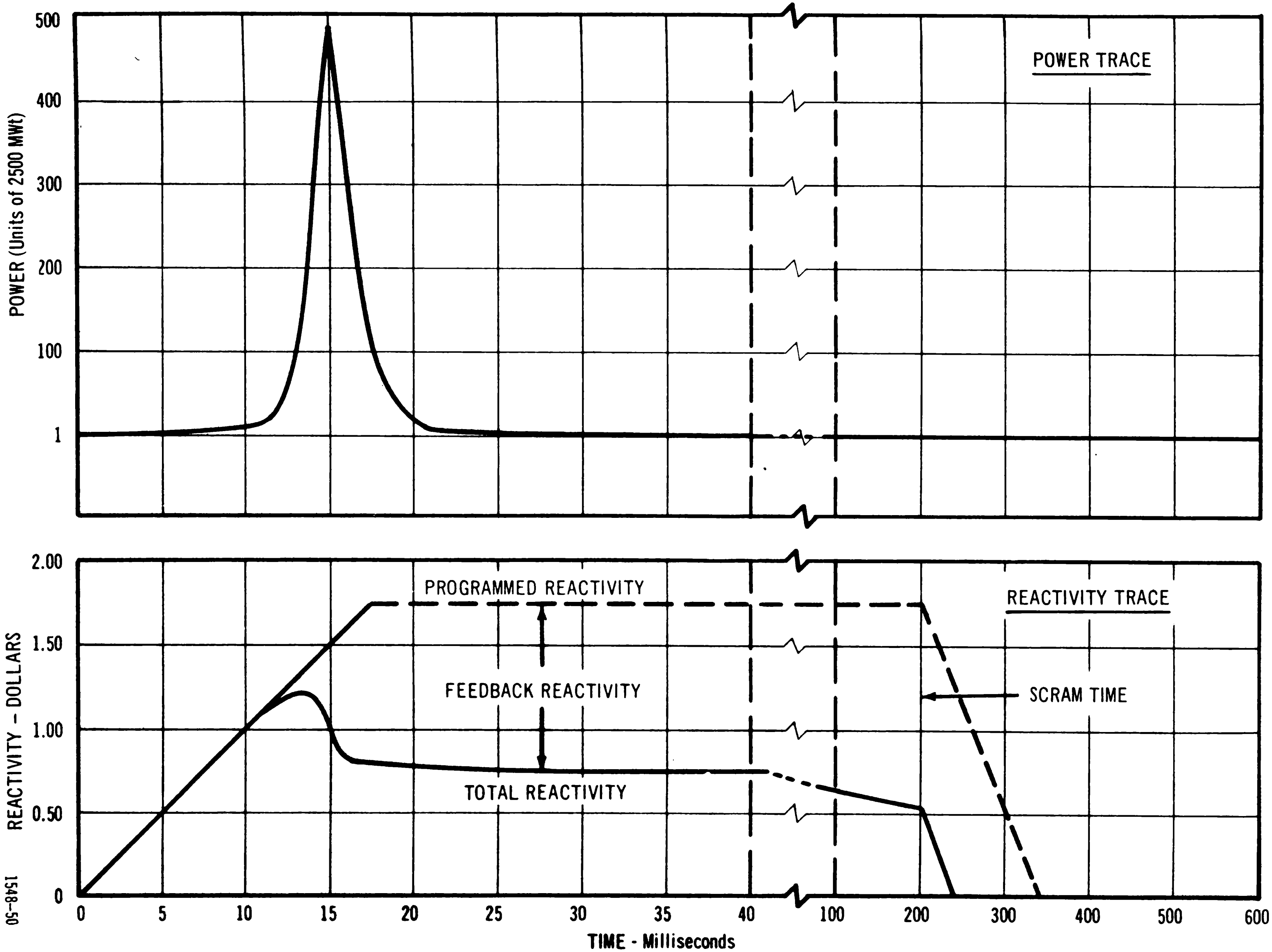

Figure 2.6.3.1(d). Major Reactivity Insertion - Power Transient

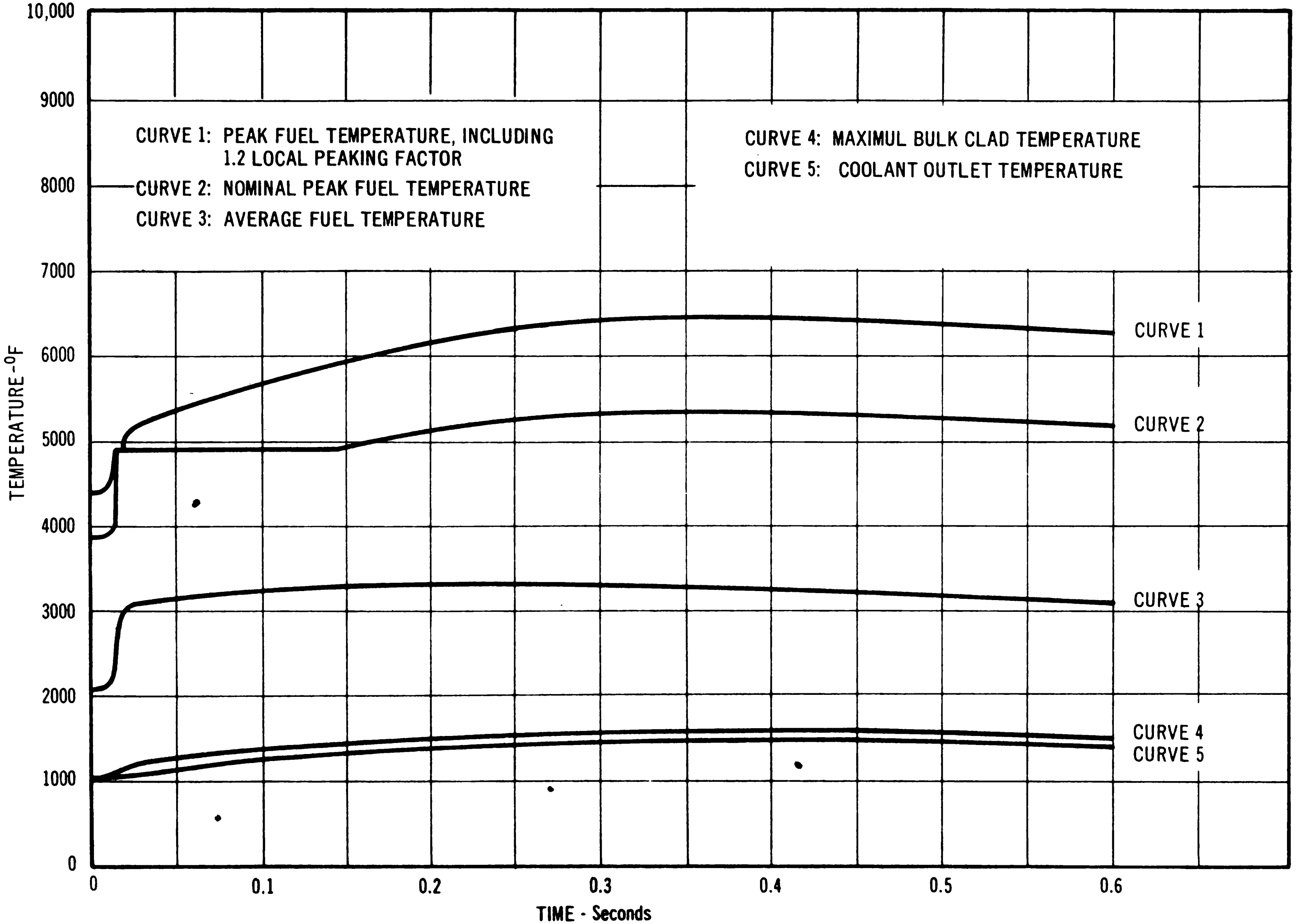

Figure 2.6.3.1(e). Major Reactivity Insertion - Fuel Temperatures

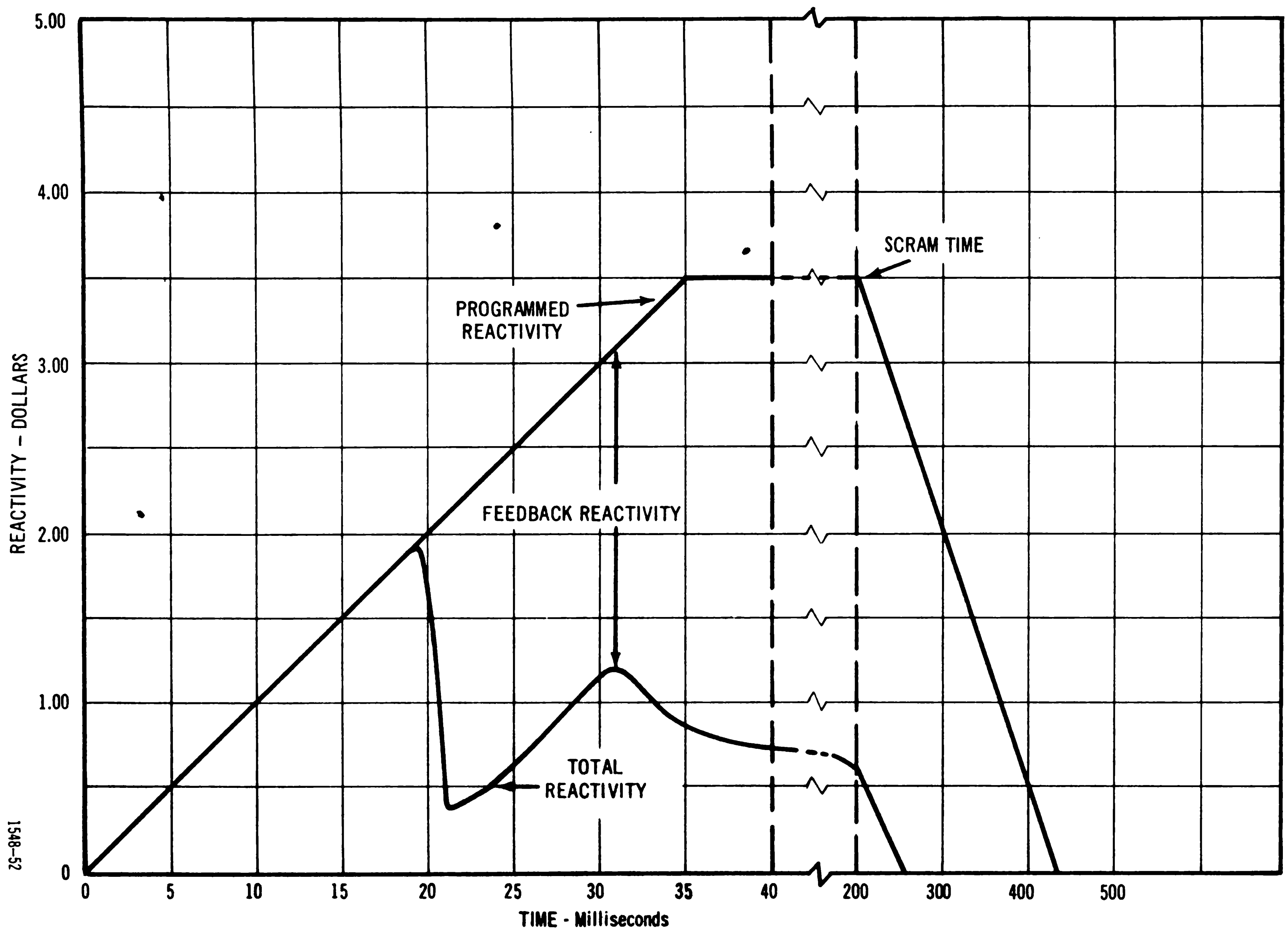

Figure 2.6.3.2(a). Refueling Accident - Reactivity Ramp

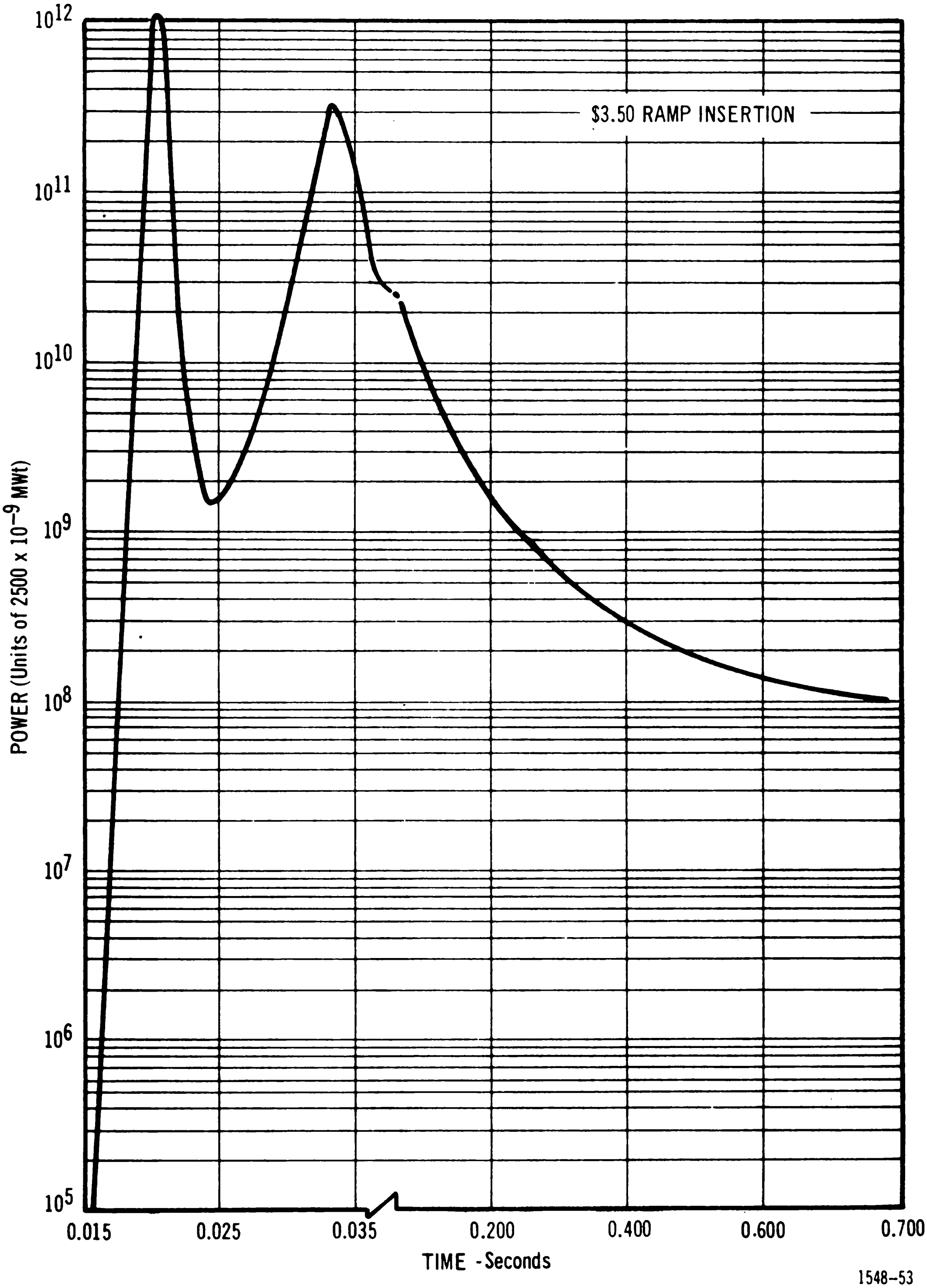

Figure 2.6.3.2(b). Refueling Accident - Power Transient

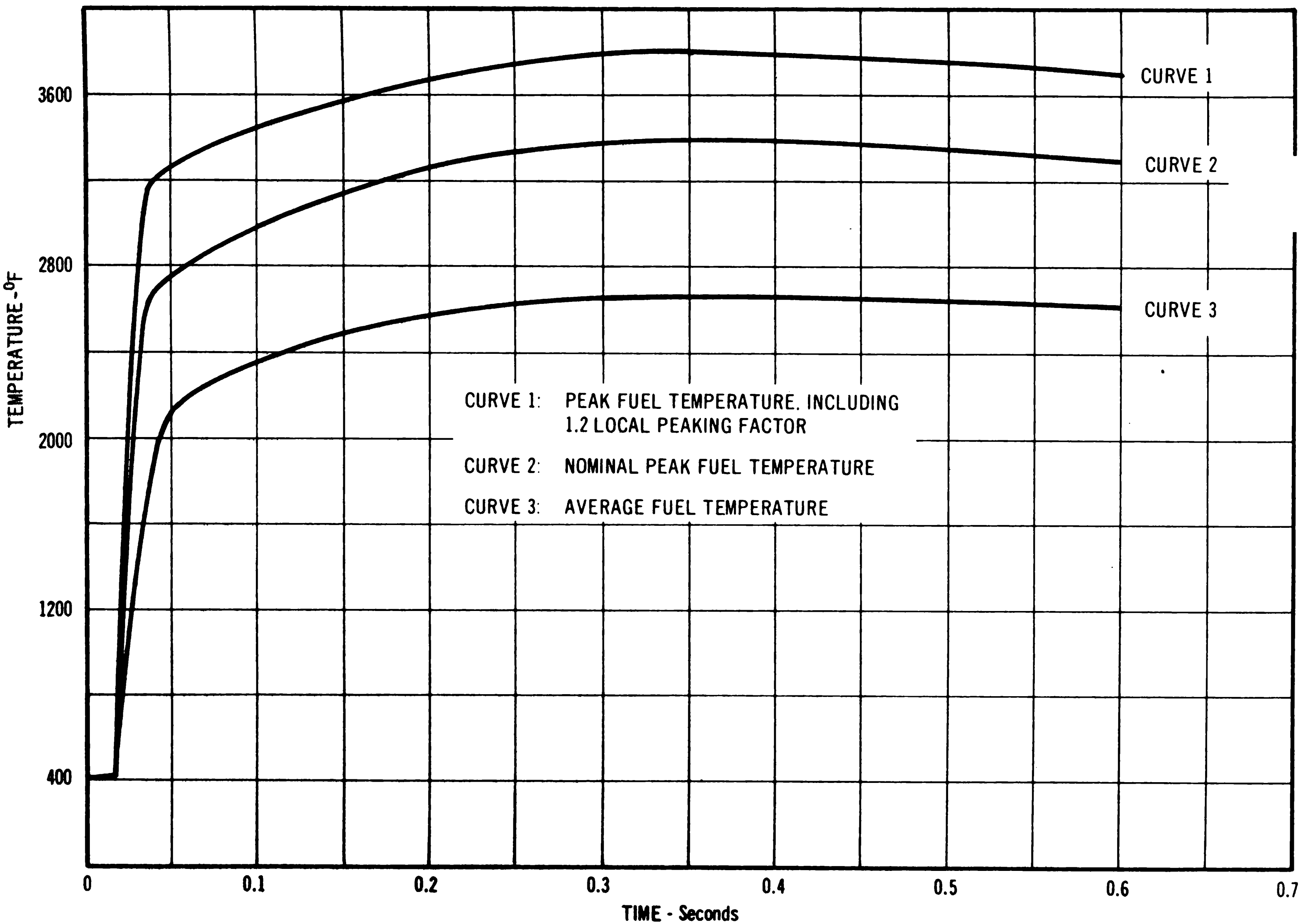

Figure 2.6.3.2(c). Refueling Accident - Fuel Temperatures

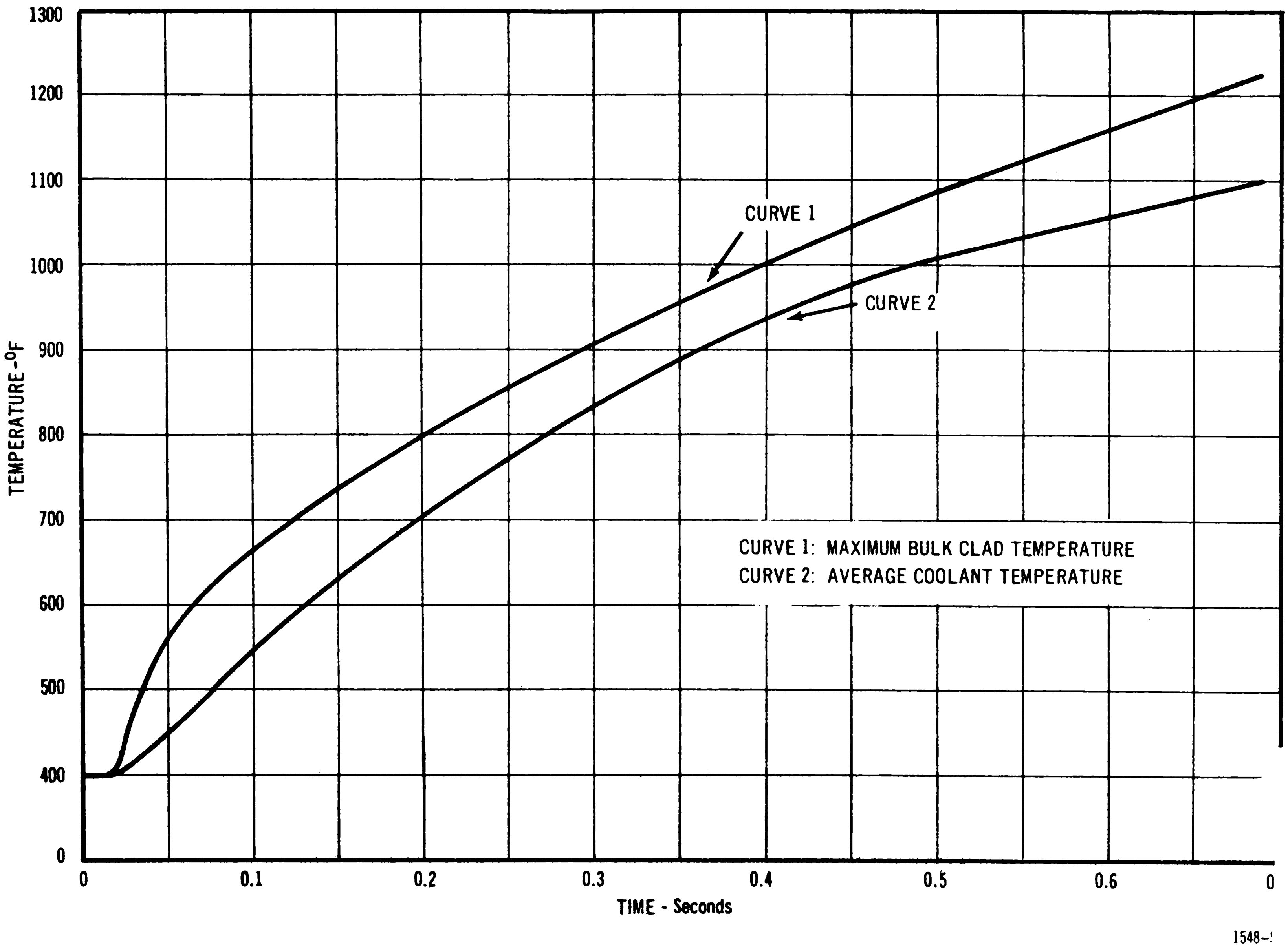

Figure 2.6.3.2(d). Refueling Accident - Coolant Temperatures

2.6.3 CONTROLLED ACCIDENTS

2.6.3.2 Refueling Accidents (continued)

Since the maximum reactivity worth of a fuel bundle is only about 70 cents, the 3.5 dollars allow margin for a very large error in the fabricated enrichment, and it can be considered incredible that a reactivity insertion of this magnitude would occur under these conditions.

2.6.3.3 Loss of Cooling

Although the reactivity change due to total coolant loss is close to zero, partial sodium loss in the center of the core can give significant positive reactivity additions, as has been shown in Section 2.5. In addition, loss of cooling in any core location could lead to overheating and failure of the clad. A large-scale occurrence of this sort would have at least the potential for developing into a meltdown and compaction accident if scram did not occur; therefore, one must consider coolant loss in some detail.

Possible coolant loss events may be categorized as follows:

1. Large-scale loss of flow due to pump failure or loss of electrical power.
2. Generalized overheating and boiling of the coolant due to a power excursion.
3. Local voiding due to flow blockage, local hot spots, gas entrainment, or other causes.

Since there are six coolant loops in the design, it may be assumed that at least two-thirds of normal flow will always be available. This would result in a steady-state coolant temperature rise of 450 F at full power, rather than 300 F.

Since the normal outlet temperature is more than 700 F below the point at which boiling would occur, and film temperature drops are about 100 F maximum, no changes to the core would exist, even if scram were delayed several minutes.

If power to the pumps is lost and the backup system provided fails to operate, there will still be enough inertia in the pumps and coolant circuit so that flow will not immediately stop and normal scram, which would be triggered by the loss of power, will shut the reactor down. It may be concluded that gross loss of flow without scram can be eliminated from consideration.

The second cause of coolant loss requires first that there be an excursion large enough to cause fuel failures and/or coolant boiling. As was shown in Section 2.6.3.2, the reactivity insertion required would be at least 1.75 dollars, which may preclude this excursion as a credible event. Calculations of the pattern of sodium voiding (see Section 3.2.4.4) indicate that if an overpower condition does cause boiling, the voids will form first at the core outlet, which will cause a negative feedback due to the position-dependent characteristics of the reactivity worth of sodium. This would tend to terminate the accident. The negative reactivity effect could be a dollar or more, although as the entire core becomes voided of sodium, the reactivity effect will apprach zero.

2.6.3 CONTROLLED ACCIDENTS

2.6.3.3 Refueling Accidents (continued)

The third category of coolant loss, a more probable type of occurrence than the first two, is much less serious. One may consider a complete loss of coolant from as many as ten subassemblies. . The reactivity addition would be no more than 40 cents. In itself, this is not sufficient to cause a significant excursion.

2.6 SYSTEMS DYNAMICS (continued)

2.6.4 UNCONTROLLED ACCIDENTS

As will be discussed in Section 3.2, the class of accidents wherein the combination of inherent Doppler feedback and control system action is insufficient to shut down the reactor may be viewed as "uncontrolled." In these extreme accidents the mechanism for shut down becomes core disassembly, with large accompanying energy releases.

These accidents would not generally be considered credible. The safety criteria of the fast breeder reactor as outlined in Section 3.2 require that all credible accidental reactivity insertions can be handled by the Doppler effect and the reactor control system, and as indicated in the foregoing analysis of controlled accidents, this criterion can be satisfied. Nevertheless, it is informative to hypothesize an uncontrolled accident, and to analyze its consequences.

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ion can be satisfied. It is informative
consequences.

Since the core will be largely destroyed in this type of event, the question of primary interest is the magnitude of accompanying energy release. This will determine whether the accident could be contained within the reactor building, or whether it would pose a hazard to areas surrounding the reactor site.

2.6.4.1 Model for Maximum Hypothetical Accident

Although one can postulate reactivity insertions of any magnitude and rate, resulting in accidents of unlimited severity, these have no meaning unless related to some actual chain of physical events. Consequently, it is necessary to have a model for the maximum hypothetical accident to define the initial conditions and the rate and magnitude of reactivity insertion.

To gain maximum information on the dynamic performance of the reference fast oxide breeder design, the model should be the worst that can be conceived.

Hypothetical uncontrolled accidents may arise from coincident unrelated failures of many components of the reactor system and its controls. The following are illustrations of such conditions, and in most cases at least partial core meltdown and consequent further reactivity addition may occur at some stage in the accident:

1. Sudden complete flow stoppage during full power operation, coupled with failure to initiate control rod scram.

2. Sudden coincident ejection of a large quantity of a reactive component of the core during full power operation plus failure to initiate scram. The core component might be:

   a. All inserted control rods

   b. A substantial portion of the BeO rods

   c. The majority of the coolant from the core center region

2.6.4 UNCONTROLLED ACCIDENTS

2.6.4.1 Model for Maximum Hypothetical Accident (continued)

3. A mild excursion while at full power, failure to initiate scram, and consequent local excessive pressure generation resulting in rapid radial compaction of adjacent sections of the core.

The hypothetical accident condition resulting from sudden complete flow stoppage to all portions of the core while it is operating at full power, coupled with failure to scram, is probably representative of the worst conceivable event in that following the boil-out of all sodium, core meltdown is almost inevitable. This is the accident model which will be examined.

In view of the limited reactivity worth of core components such as control, BeO, or sodium, their removal from the core could not result in a major accident unless preceded by some violent event resulting in their explosive ejection. It is difficult to conceive of such an event leaving the core itself as an integral reactive assembly.

In view of the large structural material content of the reference fast oxide breeder core and the fuel support method chosen, and the limited pressure possible in a core configuration, it does not appear that major reactivity insertions could occur from radial compaction of the core.

In the model used, the loss of all of the sodium increases the potential severity of the accident by:

1. Leaving a large amount of void and thus allowing rapid core compaction following meltdown. Core compaction is by far the largest source of potential reactivity addition.

2. Removing a high vapor pressure material from the core, which would tend to cause earlier disassembly if it remained in.

3. Requiring that the fuel expand to fill a large space before very great pressures can be generated, thus allowing more time for reactivity insertion.

4. Reducing the Doppler coefficient to 80 percent of its normal value, due to the loss of spectrum softening effect from the sodium.

The net reactivity effect of total sodium removal from the core is zero, as shown in Section 2.5. This will occur shortly after the assumed flow stoppage.

After sodium has been lost from the core, the clad and fuel will melt and the core will collapse, giving rise to a high rate of reactivity addition. In view of the shape of the core (11.65 feet in diameter and 2 feet high), it is unreasonable to consider compaction into a spherical shape. Since the core is about 50 percent sodium, it would be possible for the upper half of the core to collapse into the lower half, resulting in a core of half the height of the original and twice the density. This would assume that the lower half of the core did not fall but remained in its original location, which

2.6.4 UNCONTROLLED ACCIDENTS

2.6.4.1 Model for Maximum Hypothetical Accident (Continued)

could occur, since the relatively cool lower blanket would tend to support it. The addition of reactivity due to such a compaction would be of the order of 50 dollars. The compaction is assumed to occur under the acceleration of gravity. The reactivity addition would be as shown in Figure 2.6.4.1, assuming the relationship between position and reactivity is linear.

It will be shown that only a few dollars of reactivity can be inserted (· 100 milliseconds elapsed time) before disassembly, and thus the initial slope of this curve is significant. * At 0.05 second, the rate is about 80 dollars/second. However, in this analysis, to ensure a conservative result, the assumption will be made that the rate is 200 dollars/second, and the energy release will be calculated on that basis.

2.6.4.2 Calculation of Energy Release

A FORE computer calculation was made of the results of putting 200 dollars/second ramp into a core initially at full power, from which the sodium had been removed. Shown in Figure 2.6.4.2(a) are the resultant reactivities as a function of time. Shown in Figures 2.6.4.2(b) and 2.6.4.2(c) are the power and the average fuel temperature, respectively. The calculation shows that even for a continuing ramp of 200 dollars/second, the Doppler feedback rate exceeds temporarily, the ramp rate, bringing total effective excess reactivity below a dollar. This causes a rapid fall in power, a corresponding reduction in the rate of fuel temperature rise, and a reduction in the rate of increase of the Doppler effect. Thus, the ramp insertion again causes the net reactivity to exceed a dollar, and another power burst results, again bringing the net reactivity below 1 dollar. This phenomenon of repeated power peaks, with the net effective reactivity oscillating about a value of 1 dollar, is typical for a fast reactor with a strong negative Doppler coefficient, when subjected to a rapid rate of reactivity insertion. The phenomena thus causes a progressive rise in maximum fuel temperature and ultimately leads to vaporization of the fuel and core disassembly.

The method used to calculate the energy release from the excursion after the core begins to disassemble due to the pressures generated is essentially a Bethe-Tait analysis[3, 4, 5] with the Doppler effect incorporated.

This type of calculation has been performed previously[6, 7] showing the strong influence of the Doppler coefficient in reducing energy release on meltdown and compaction accidents. The equations used in this instance are exactly those described by Wolfe, Friedman and Riley,[7] and were solved using a digital computer program. Important inputs to the calculation are the threshold energy (energy at which significant pressure contributing to disassembly is generated), the relationship between pressure and energy density (equation of state), and the reactivity at the time the threshold energy is reached, which is the zero time for the calculation.

*This factor removes any need to consider the possible separation of BeO from the core or other like occurrences which might be postulated if the core meltdown were to progress for a significant (several seconds) period of time.

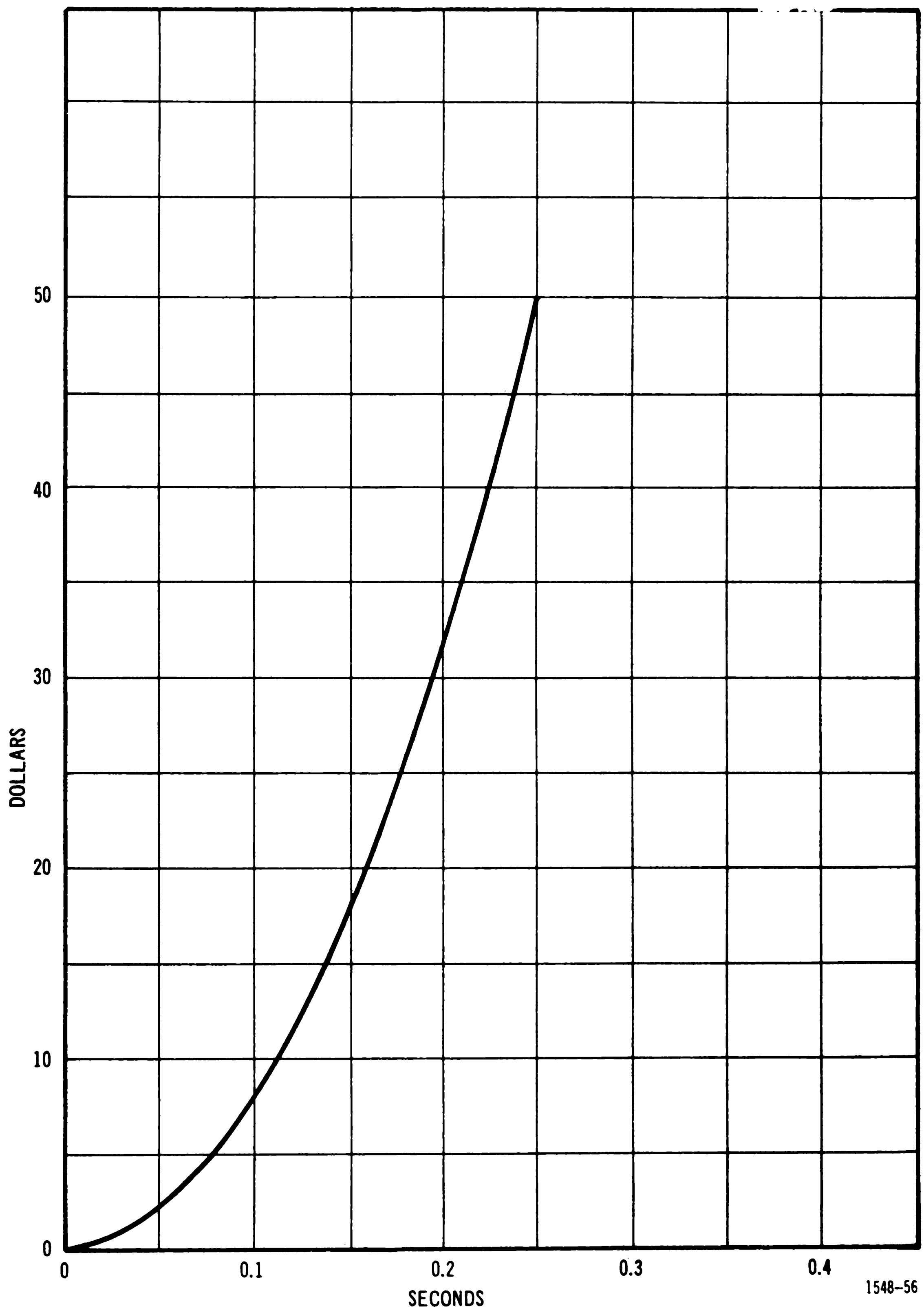

Figure 2.6.4.1. Maximum Hypothetical Accident - Reactivity Insertion

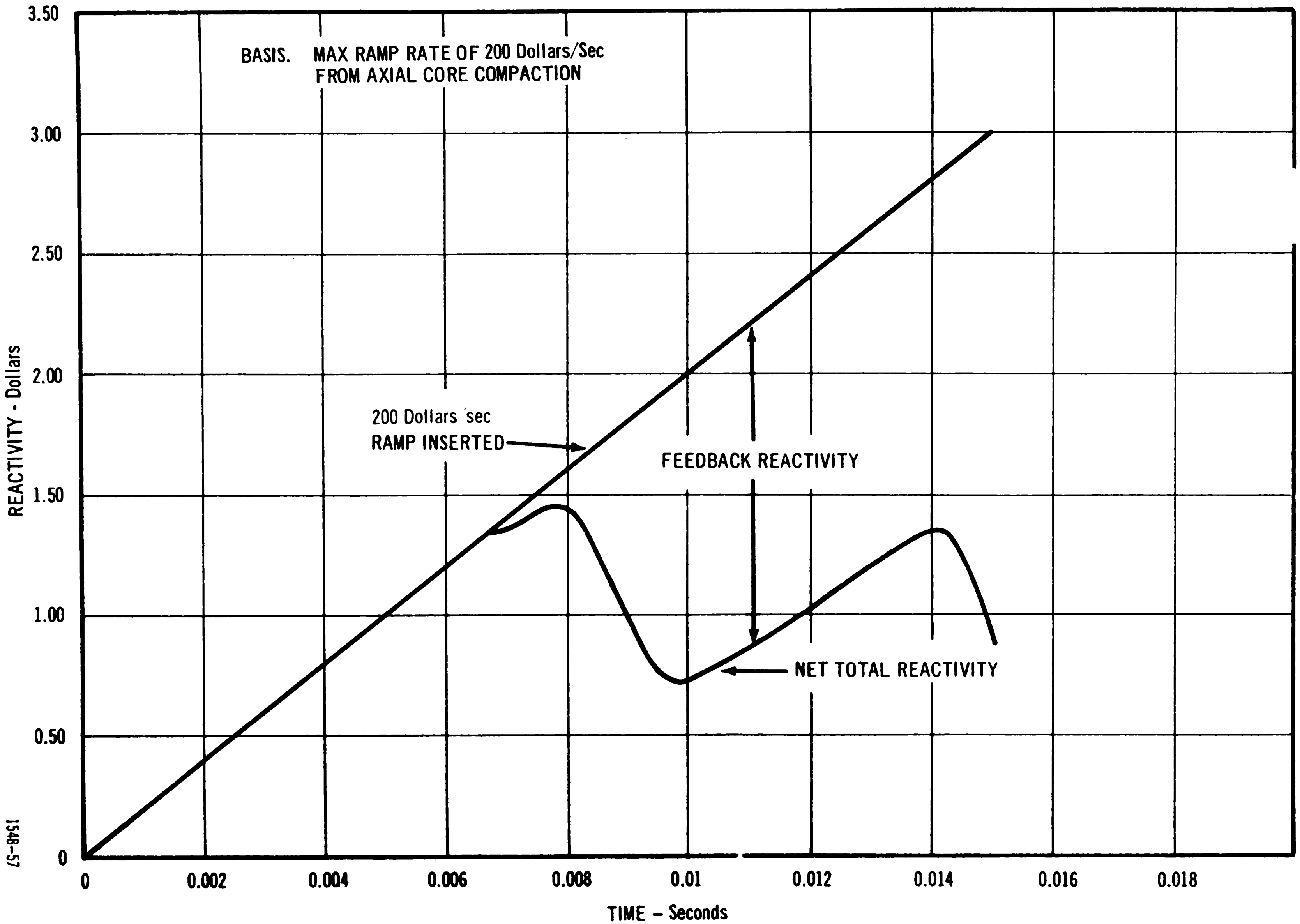

Figure 2.6.4.2(a). Maximum Hypothetical Accident - Reactivity Insertion

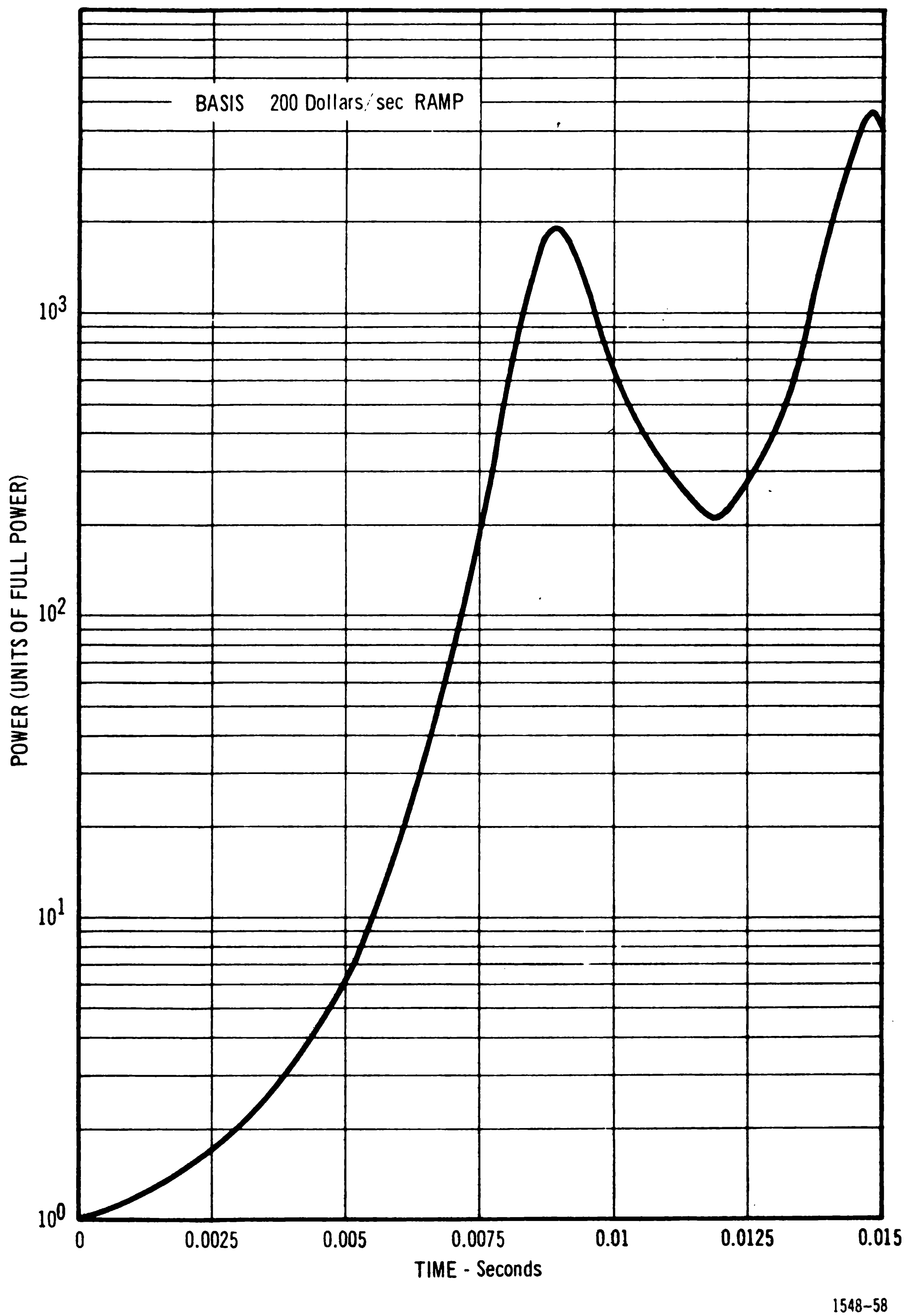

Figure 2.6.4.2(b). Maximum Hypothetical Accident - Power Transient

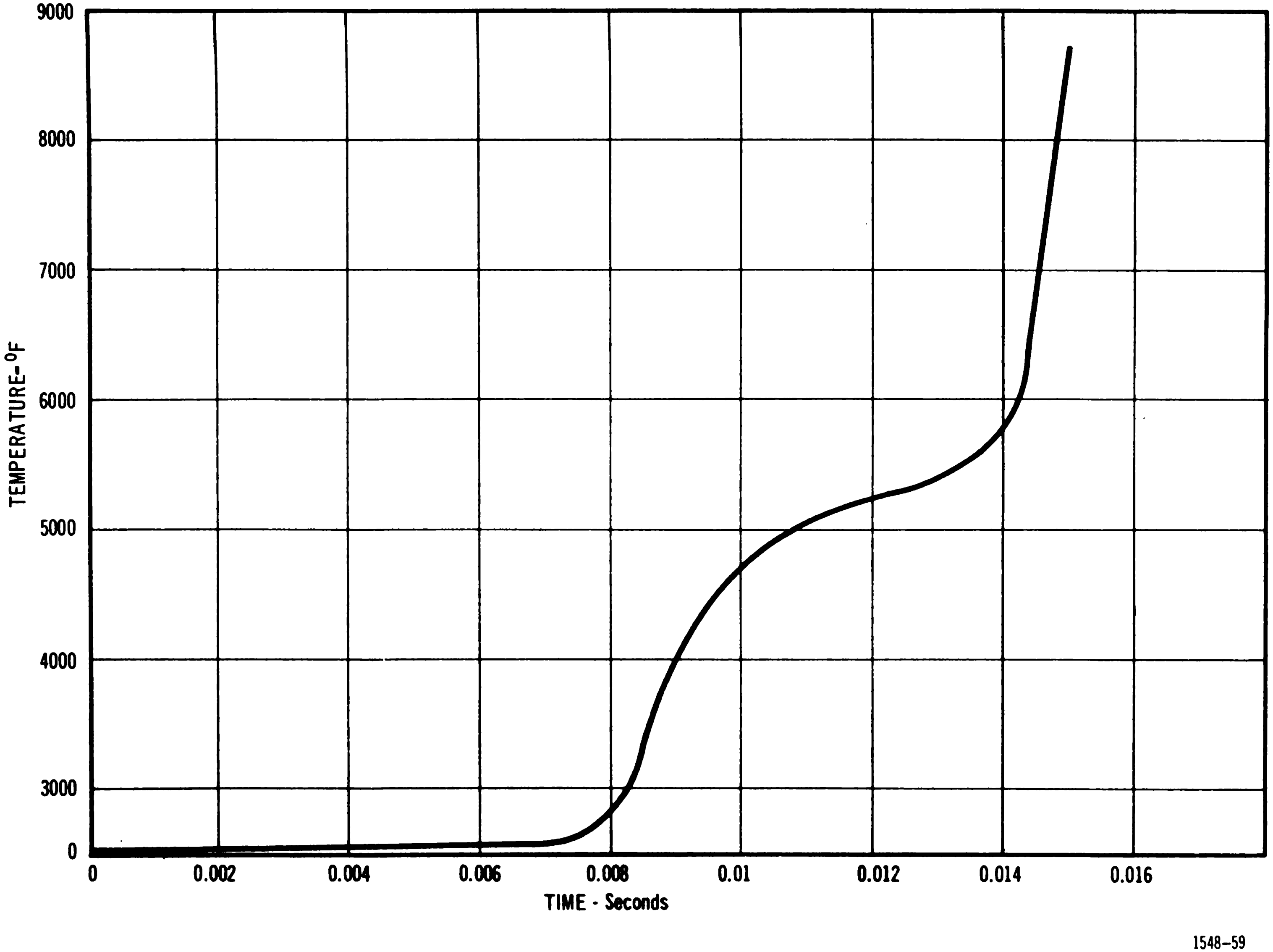

Figure 2.6.4.2(c). Maximum Hypothetical Accident - Fuel Temperature

2.6.4 UNCONTROLLED ACCIDENTS

2.6.4.2 Calculation of Energy Release (continued)

Because of the many assumptions necessary to complete the calculation, the results can be taken only as illustrative of an energy release which might be typical for a reactor similar to the 1000-MWe fast breeding reactor.

The pressure-energy relationship assumed for this calculation is shown in Figure 2.6.4.2(d). The relationship has been derived for UO_2 following the approach used by Brout [8] for metallic uranium. The assumption was made that no pressure would be generated until the fuel expanded to fill the space left by the sodium; therefore, the effect of vapor pressures below this point is not taken into account. In order to fill the void, the fuel specific volume must be about 80 cc/mole, which is close to the derived value for the critical volume of 77 cc/mole. The critical temperature and pressure used were 5150 K and 1650 atmospheres, respectively, which were derived using a method similar to that in reference. [9]

The lowest threshold temperature that could be assumed under any circumstances would be the estimated boiling point at atmospheric pressure (between 6000 and 7000 F), as can be seen in Figure 3.2.3.1(b). The assumed critical temperature is about 8800 F, which occurs only 0.5 millisecond later than the boiling point, as shown in Figure 2.6.4.2(c). Therefore, the threshold conditions were assumed to be at the critical temperature and pressure.

The specific heat was taken equal to 13.2 R, which is the maximum theoretical value for a gaseous UO_2 molecule.

The reactivity at the threshold conditions was assumed to be 35 cents above prompt critical. This is the maximum value shown on Figure 2.6.4.2(a) after the fuel had entered the critical temperature range.

The resultant energy release above the threshold energy (the latter is absorbed by the temperature rise of the core) is 850 MW-seconds, released over a period of less than half a millisecond. The total reactivity inserted after the threshold energy is reached is less than 10 cents, if it is assumed that the collapse of the core continued. In fact, the ramp insertion is reversed and the core disassembly contributes to the shutdown of the accident, although the Doppler effect is more important immediately after the threshold energy is reached.

2.6.4.3 Post-Accident Condition

The excess energy release of 850 MW-seconds must be related to some measure of damage in order to be meaningful. The amount of energy available for mechanical work will be some fraction of the 850 MW-seconds since some of the energy will go into heating of the core material. Assuming an isotropic expansion gives a conservative value of about one-half of the total energy. [10]

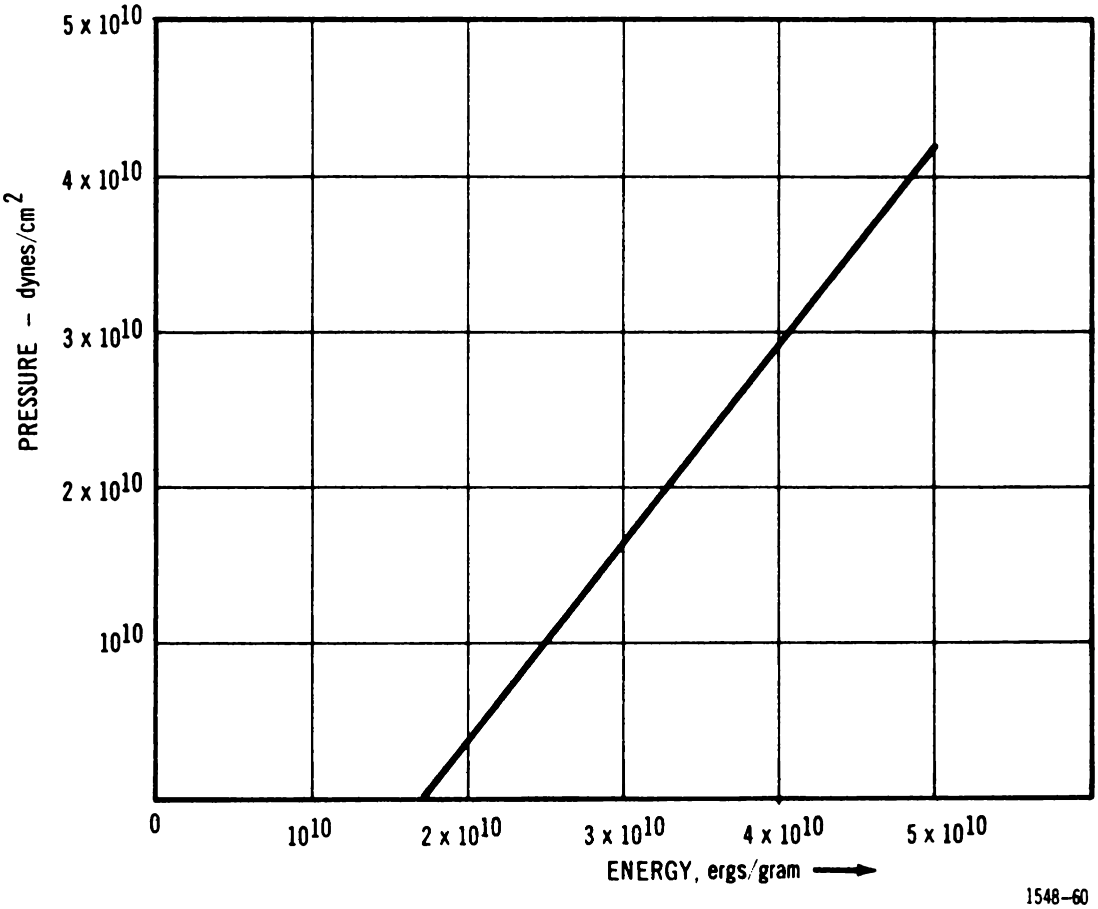

Figure 2.6.4.2(d). Pressure-Energy Relationship Assumed for Fuel

2.7 PLANT STAFF AND REFUELING OPERATIONS

2.7.1 OPERATING AND MAINTENANCE STAFF

2.7.2 REFUELING OPERATION

2.7.2.1 Time Requirements

2.7.3 REACTOR MAINTENANCE

2.7.3.1 General Philosophy

2.7.3.2 Remote Component Replacement

2.7.3.3 Component Removal and Repair

2.7.3.4 Small Component Maintenance

2.6.4 UNCONTROLLED ACCIDENTS

2.6.4.3 Post-Accident Condition (continued)

for the explosive energy; thus, about 425 MW/seconds of explosive energy might be released. Using a conversion of 2 MW/seconds of explosive energy for a pound of TNT, this hypothetical accident would be equivalent to the explosion of about 200 pounds of TNT.

The design of the containment building and of the primary loops were outside the scope of this design study. However, it should be feasible to contain this accident within the reactor containment building, with no danger to the reactor environs. The 200 pounds of TNT compares with a value of 1000 pounds in the maximum hypothetical accident for the Enrico Fermi plant and the containment for that plant is calculated to be capable withstanding the explosion corresponding to 1000 pounds of TNT. [10]

The primary sodium tank and associated shielding structure could probably be designed to absorb the 200 pounds of TNT without disruption of the main operating floor of the plant. The additional containment provided by the above grade zero leakage reactor building structure thus would be unaffected and able to prevent subsequent uncontrolled leakage of products released from the disrupted core or primary coolant system. Providing that the primary system design were adequate to prevent major exposure of sodium to the oxygen in the above grade portions of the containment building, the temperature rise in the primary tank necessary to absorb all of the released energy would be only 1 F.

There are no other sources of energy available unless the steam generator fails and allows the steam and sodium to mix. Since the steam system is isolated from the primary sodium by the secondary loop, it should be possible to ensure that the steam generator is not subjected to any conditions likely to cause failure. The failure of the intermediate heat exchangers is likely, with resultant mixing of primary and secondary sodium, but even if the valves have not closed in the secondary sodium piping, the mixture of primary and secondary sodium will not be at a high enough temperature to damage the steam generator.

Due to the many assumptions involved in the analysis, it is felt that the energy release and its effects are subject to considerable variation from the values calculated, which can only be considered a crude but probably conservative estimate of the magnitude of the accident. However, it may be concluded that even the maximum hypothetical accident described here would be contained by the reactor building, and that essentially no hazard would exist for the reactor environs.

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2.7 PLANT STAFF AND REFUELING OPERATIONS

2.7.1 OPERATING AND MAINTENANCE STAFF

The required operating staff for the 1000-MWe plant is estimated to consist of a trained crew of approximately 112 men. The estimate is based on water reactor experience.

Supervision is provided by five senior people under the general titles of Superintendent, Operations Supervisor, Maintenance Supervisor, Engineering Supervisor, and Services Supervisor. The Operations Unit is comprised of about 50 people including the Supervisor, five Shift Supervisors, five Chief Operators, 20 (four per shift) Operators, 20 Helpers, and a Planner and Scheduler. The Maintenance Unit is comprised of about 30 people including the Maintenance Supervisor, nine Instrument Mechanics, five Electricians, 12 Mechanics, and three Helpers. This staff will comprise the on-site staff and may be supplemented by a roving plant overhaul crew as required for major planned maintenance or emergencies. This roving crew will service a large number of reactor and conventional plants.

The Engineering Unit includes 13 people: the Supervisor, Nuclear Engineer, two Electrical Engineers, two Chemical Engineers, two Mechanical Engineers, three Junior Engineers, and two Technicians.

The Services Unit includes the Supervisor, four Clerks, four Janitors, two Storekeepers, and five Guards.

2.7 PLANT STAFF AND REFUELING OPERATIONS (Continued)

2.7.2 REFUELING OPERATION

Reactor refueling will be performed twice a year utilizing the semi-remote system described in Section 2.3.3. The detailed refueling procedure is described below.

The refueling operation will consist normally of the following major tasks in the sequence given:

1. Reactor shutdown
2. Physics testing prior to refueling
3. Control rod insertion and delatching of drives
4. Sodium tank cooling and refueling cell heating
5. Reactor access plug removal
6. Sodium level adjustment, lower (see Figure 2.3.3)
7. Fuel selection and extension rod attachment
8. Sodium level adjustment, raise (see Figure 2.3.3)
9. Refueling
10. Leaker detection and removal
11. Extension rod removal
12. Shield plug replacement
13. Refueling room cooling
14. Access plug securing
15. Control rod drive latching
16. Check out for startup

The equipment used for these operations has been described in Section 2.3.3. The operation of equipment will involve three operators plus maintenance support.

Upon completion of the physics tests, the refueling sequence will begin by driving the control rods to the refueling position, unlatching, and retracting the rod drives while purging the refueling cell with inert gas to remove any traces of contaminants. At this point a crew in fresh air masks will enter the refueling cell and perform the necessary operations to unseal the shield plug and unbolt the vessel head. Upon completion of the in-cell work, the cell atmosphere is again checked and the pressures equalized between the vessel and the cell. As soon as the personnel exit, the refueling cell is heated to about 400 F to control sodium vapors. The access shield plug with reactor vessel head attached (see Section 2.3.1) is next raised to the refueling position near the ceiling of the cell.

The heads of the fuel assemblies are exposed by adjusting cover gas pressures (see Section 2.3.3), and extension rods are placed on the fuel selected for removal using the manipulators. The fuel selected is checked for accuracy and photographed for accountability at this point. Then the sodium level is raised so that only the top of the extension and the control rod guide tubes are exposed.

2.7 PLANT STAFF AND REFUELING OPERATIONS (Continued)

2.7.2 REFUELING OPERATION (Continued

Each extension rod and attached fuel subassembly is now grappled, using the manipulators, and placed in the fuel transfer drum. A new subassembly is removed from the drum and placed in the hole left by the subassembly which was removed from the core.

Upon completion of refueling and removal of fuel extension rods, the sodium level is adjusted to check the level (axial position) of each core subassembly. After the check is complete, the access plug and reactor vessel head are lowered into place and the refueling cell is cooled.

When the refueling cell has cooled and activity levels permit, a crew again enters the cell in fresh air masks to seal the access plug and bolt down the reactor vessel head. The rod drives are then coupled to the rods and checked. Before the crew exits the cell, the seals are checked for cover gas leakage. The reactor is now ready to begin reactor physics tests prior to startup.

2.7.2.1 Time Requirements

An estimate has been made of the time required to perform the refueling operations described above. Assuming no equipment malfunctions, it is estimated that the various procedures necessary to the refueling task can be performed in seven 24-hour days. The basis for this estimate is shown in Table 2.7.2.1, which lists the required operations and the estimated time required to perform them.

Failed-fuel detection and handling would increase the refueling time. A system for detection will be necessary which can select failed subassemblies. Once detected, failed fuel will be removed and then placed in cannisters for shipment.

Although not a subject in this discussion, reactor physics testing associated with refueling and licensing requirements are expected to add to the refueling shutdown. Seven to 14 days have been assumed arbitrarily.

A consolidated estimate of the time required to refuel 50 subassemblies is as follows:

	24-hour days	
	Lo	Hi
Physics Testing	7	14
Refueling	7	14
System Maintenance	-	14
	14	42

Thus, a 1-month shutdown appears to be a fair approximation of an expected norm. Peak possible plant availability factor under these assumptions is 92 percent, and the expected norm is 83 percent

TABLE 2.7.2.1

FCR REFUELING OPERATION

	Unit Time (min)	Total Time (hr)	Accomplished During Shift Number	Possible Delays
1. Shutdown of physics testing		negligible	-	3-5 days
2. Drive Control rods to down position		negligible	1	
3. Uncouple rods and check		1	2	
4. Retract rod drives		1	2	
5. Position transfer tank		negligible		
6. Vent cover gas		2	2	
7. Reduce pump speed		negligible	3	
8. Cool sodium to 400 F		3	3	Stress in Intermediate Heat Exchanges Controlling
9. Set argon purge and purge		8	3	
10. Prepare for cell centry		2	4	
11. Purge and test for in-leakage		4	4	Fix leaks, 2-3 shifts
12. Equalize cover gas and refueling room		1	4	
13. Enter cell with fresh air masks (2 men)		1	5	
14. Test for in-leakage		negligible		
15. Purge as required		negligible		
16. Uncouple leads temperature* (15 junctions)	10	4	5	1 shift
17. Exit cell			5	
18. Enter cell with new shift		1	6	
19. Unbolt reactor vessel head		3	6	
20. Unseal shield plug		3	6	
21. Exit cell		negligible	6	
22. Heat refueling room		8	7	
23. Raise plug		1	7	

*Possible to eliminate

TABLE 2.7.2.1 (Continued)

	Unit Time (min)	Total Time (hr)	Accomplished During Shift Number	Possible Delays
24. Adjust pressure to expose fuel		1	8	
25. Provide cocked safety		1	8	
26. Attach 50 extension rods on fuel	4	3	8	2 shifts
*27. Attach 80 extensions on rods		5	8 and 9	
28. Check and photograph		1	9	Verify 1 shift
29. Raise sodium level		1	9	
30. Refueling - select subassembly	1			
- lower gripper	3-1/2			manipulator maintenance
- grab extension	1/2			
- raise	1/2			stuck fuel assemblies
- swing	1			
- lower into transfer tank	1/2			
- release	1/4			
- remove extension and add new assembly	1			
- grab new sub	1/2			
- raise from TT	1/4			
- swing over core	1			
- lower into core	3-1/2			
- remove extension	1/2			
- raise extension to storage rack	1			
min.	15			4/hour
Exchange 10 fuel assemblies		2-1/2	9	
Exchange 20 fuel assemblies		4-1/2	10	Contingency of 5 days
Exchange 20 fuel assemblies		4-1/2	11	
31. Locate and remove leakers and test subs			12-14	5 days
*32. Remove and control rod extensions		5	15	

*Possible to eliminate

TABLE 2.7.2.1 (Continued)

	Unit Time (min)	Total Time (hr)	Accomplished During Shift Number	Possible Delays
33. Remove cocked safety			16	
34. Equalize pressure			16	
35. Check location and level			16-17	verify 2 days
36. Lower shield plug			18	
37. Cool refueling room		8	18	
38. Enter cell in fresh air masks			19	
39. Seal shield plug		3	19	1 shift
40. Bolt reactor vessel head		3	19	1 shift
41. Exit cell		1	19	
42. Enter cell in air		1	20	
43. Couple leads		4	20	
44. Exit cell		1	20	
45. Couple rod drives			21	
46. Check rod drives			21	
47. Double check lead continuity, etc.			21	1 shift
		Total	21	

'Possible to eliminate

2.7.3 REACTOR MAINTENANCE

2.7.3.1 General Philosophy

A high plant availability is a major design goal for the proposed single-unit, 1000-MWe fast reactor station. The proposed biannual refueling operations are expected to hold the plant availability to 80-90 percent and it thus becomes very desirable that all major plant maintenance be scheduled as a separate operation to be accomplished during the refueling shutdowns wherever possible.

Vulnerable equipment pieces are positioned for easy access to accommodate either prompt repair or replacement. When failures occur remote equipment replacement is required because of the inherent coolant system activity and efficiency loss of awaiting decay. Personnel access is not attempted at this point; replacement of either spare or dummy components is made, and the plant made available for power operation with minimum delay. Check valves provided in the primary loops permit power operation at part power using a reduced number of coolant loops if necessary.

The facilities and equipment designed to implement this philosophy have been described briefly in Section 2.2. The primary facilities for maintenance are removal equipment, three storage pits, a separate radioactive maintenance area in the nuclear steam supply building, and an interconnecting accessway.

Equipment design and component layout in the single-tank design dictates close integration of shielding with the components. Component removal is accomplished by pulling the shield plug that has the component attached, maintaining an inert atmosphere over the tank sodium using the bagging technique, and placing the used component in an available storage pit. The replacement part is then bagged into position using the reverse procedure.

The failed or damaged component will be left in the storage pit in an inert gas atmosphere until decay or decontamination permits contact maintenance. When activity levels permit, the component will be removed to a separate maintenance area which will be equipped for sodium removal, further decontamination, and repair. Upon examination the economics of repair as compared to discard will be evaluated in an atmosphere relieved of plant startup urgency.

This technique of large-component replacement and maintenance places emphasis on component accessibility and reliability. On the other hand, primary sodium must be shielded heavily, and space within the shielded primary tank is at a premium. The primary-system components have been grouped together so that replacement of any component can be accomplished with a convenient number of shield plugs. The arrangement is based on the component function and the ease of access required. Seventeen shield plugs contain all major removable components. These are six primary heat exchangers, six primary sodium pumps, two sodium purification units, one main reactor access plug, one fuel transfer plug, and one primary sodium tank instrumentation plug.

2.7.3 REACTOR MAINTENANCE (Continued)

2.7.3.1 General Philosophy (Continued)

The equipment mounted on shield plugs will be designed so that some maintenance in place is possible. Whenever possible, motors, journal bearings, seals, instrumentation, etc., will be made accessible above the shield plug. Preventative maintenance of these components will be facilitated in this manner.

In general, the in-tank equipment will be designed to operate over the entire plant life. Failure will normally require replacement rather than repair. One sodium-purification unit will be designed for plant startup with the intent of replacing it with a unit designed for the remainder of the plant life.

Table 2.7.3.1 lists the major reactor equipment and planned maintenance procedures.

2.7.3.2 Remote Component Replacement

The equipment and procedures for component removal have been conceived as similar to those used in the Enrico Fermi Power Plant system. The equipment required is illustrated in Figure 2.7.3.2. The equipment consists of a flexible gas lock, gates, seals, and a crane.

Spool pieces, each containing one gate, are first placed above, and sealed to both the shield plug flanges of the component to be removed and the storage pit. The gas lock containing a second gate is then sealed to the spool (on installation, this may be completed as a single operation). With both gates open, the component can be lifted from the primary tank into the flexible bag without contaminating the environs or the sodium system. Control of the cover gas system pressures during removal will assist this operation.

Once the gates are clear they are closed and the unit separated between the gates. A spool piece with a gate similar to that on the equipment flange was also placed on the equipment storage pit so that an inert atmosphere could be maintained in this system as well.

The component in the lock is then swung into position over the storage pit. Once the lock is sealed in position the gates are opened and the component lowered into the pit. A component is returned to the system by the reverse procedure. The height of the containment building is determined by the handling requirements of the heat exchangers.

Two equipment storage pits are located within the containment vessel to receive replaced components. Special structural supports will permit receipt of every major component. The pits will be tanks and will comprise an integral part of the cover gas system. The atmosphere within the tanks will be controlled through appropriate valving associated with the cover gas system.

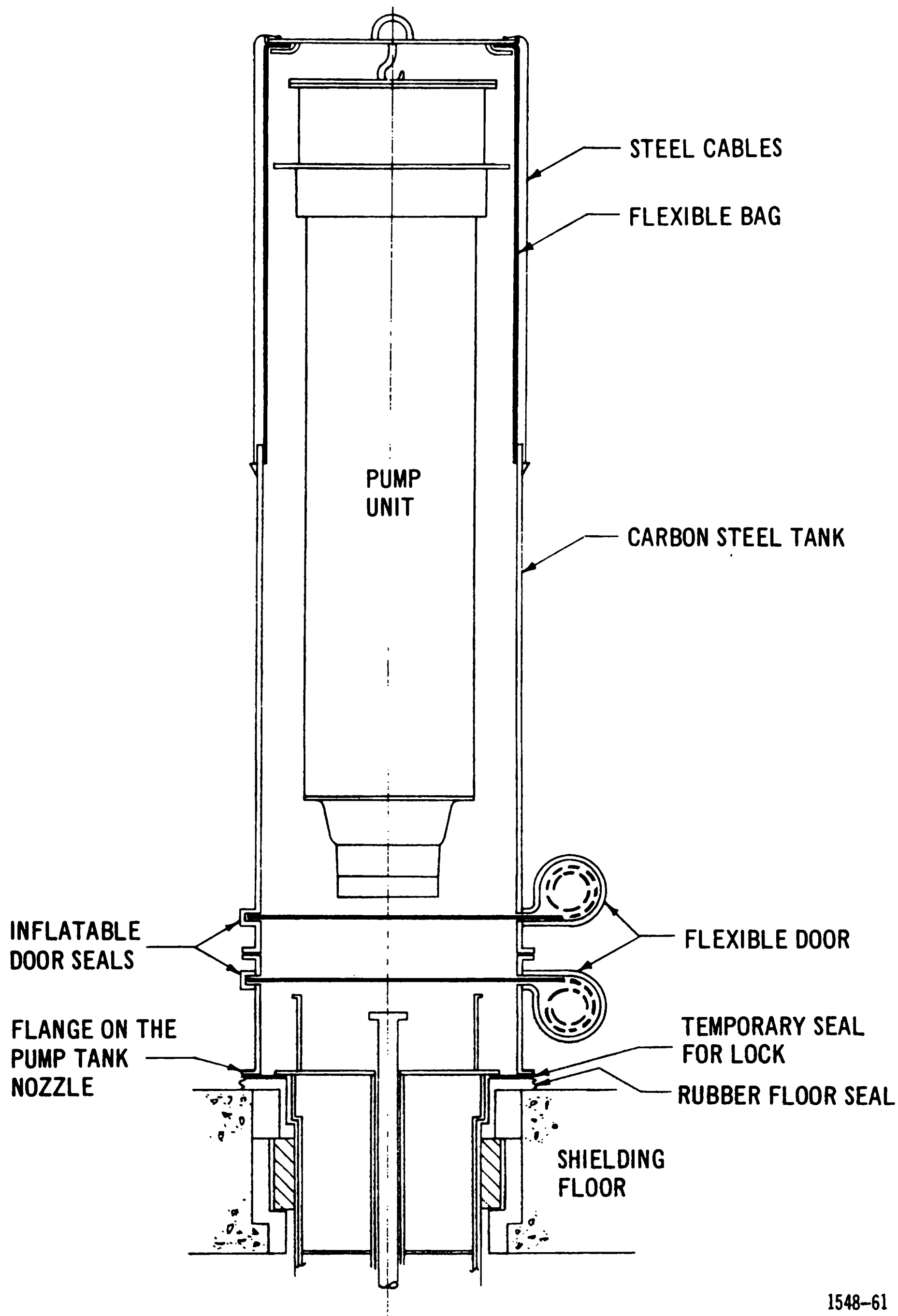


Figure 2.7.3.2. Typical Primary Loop Component Removal Means

2.7.3 REACTOR MAINTENANCE (Continued)

2.7.3.2 Remote Component Replacement (Continued)

The design criteria for the building crane would be based on the capability of replacing the heaviest and largest component in the system. The crane would also be capable of loading equipment onto flatcars for removal from the building.

2.7.3.3 Component Removal and Repair

After decay in the storage pits, component removal and repair will be accomplished using the building crane, a flatcar on rails, a maintenance area crane, and facilities for receiving, cleaning, decontaminating, disassembling, and repairing the component.

The large equipment will be placed in a storage pit in a pit-liner tank which contains the piece completely. Upon removal, the building crane will lift the equipment in its container. In this manner an inert-gas atmosphere can be maintained, and radioactive contamination can be controlled during the transfer.

The equipment in its container can then be placed in a sodium-removal facility. Conceptually this facility is a mineral oil tank equipped for ultrasonic vibration and mineral oil recirculation and cleanup.

2.7.3.4 Small Component Maintenance

Certain of the smaller radioactive components will require special maintenance procedures. Examples are the primary loop check valves, control rod drives, reactor instrumentation, and the refueling system.

Check valves in the primary loops will be made accessible through the refueling room. These valves will be designed so that all maintainable internal parts are attached to the cover flange. The cover flange will be designed so that remote removal and replacement is possible using the fuel handling manipulators.

Control rod drives will present a special problem because they will be situated in the refueling room. Contact maintenance probably will be performed by men in fresh air masks during the refueling shutdown.

The refueling system will be designed for contact maintenance of the equipment in the refueling room. Whenever possible, drive mechanisms, etc., will be placed on the accessible side of the room. The remote manipulators will be designed for either easy repair or replacement of the in-tank portions.

TABLE 2.7.3.1

MAJOR EQUIPMENT MAINTENANCE

Equipment	No. Required	Operation or Failure	Type of Maintenance Planned	Planned Maintenance or Replacement Frequency
Primary Heat Exchangers	6	a. Plug tubes.	a. Contact maintenance in repair facility after sodium removal.	a. Infrequent.
		b. Shell and tube sheet failure.	b. Disposal.	b. Design life same as plant life.
		c. Replace temperature instrumentation.	c. Replacement in-place.	c. During refueling or annual shutdown.
Primary Sodium Pumps	6	a. Repair motor, shaft seal and associated equipment (accessible without pulling shielding plug).	a-1. Preventative maintenance. a-2. Replacement in place.	a-1. During refueling or annual shutdown. a-2. Infrequent.
		b. Repair pump shaft, bearings, impeller (rotating members).	b. Contact maintenance in repair facility after sodium removal <u>or</u> disposal.	b. Design life same as plant life but inspection every 5 years is probable.
		c. Repair support structure for shaft seal bearings, wear rings, seal rings, and diffuser section for discharge flow.	c. Contact maintenance in repair facility after sodium removal <u>or</u> disposal.	c. Infrequent.
Sodium Purification Unit	2	a. Replace oxide collection portion of cold trap.	a. Remote replacement and disposal of entire unit.	a. Once after startup, once during plant life.
		b. Replace or repair sodium pump motor, shaft seal, and associated equipment instrumentation, coolant circulation system coolers, motors, etc. (accessible without pulling shielding plug).	b-1. Preventative maintenance. b-2. Replacement in-place.	b-1. During refueling or annual shutdown. b-2. Infrequent (possible during power operation).
		c. Replace economizer pump moving parts, plugging indicator, flow control equipment, shield plug and casing.	c. Remote replacement and disposal of entire unit.	c. Once after startup, once during plant life.
Reactor Access Plug	1	a. Replace core thermocouples.	a. Replaced with plug in place.	a. As required.
		b. Replace control rod drives and couplings.	b. Replaced with plug in place.	b. During refueling or annual shutdown.
		c. Replace core ion chambers.	c. Replaced with plug in place.	c. As required.
		d. Replace control rod blades.	d. Replaced with plug removed.	d. Inspection and replacement during annual shutdown.
		e. Replace seal.	e. Replaced on each removal.	e. Inspection and replacement during annual shutdown.
Fuel Transfer Plug	1	a. Replace seal.	a. Replaced on each removal.	a. During refueling.
Axial Core Ion Chambers	1 or more	a. Replace ion chamber.	a. Replaced from top of shield.	a. As required.
Primary Sodium Tank Instrumentation	1	a. Replace level, temperature, or activity monitors.	a. Plug replaced on failure.	a. Infrequent.
Sodium Removal Equipment - Primary Tank	1	a. Install.	a. Replace other plug for temporary installation.	a. Not normally required in life of plant.
Steam Generator	6	a. Plug tubes.	a. Plugged with unit in place.	a. During annual shutdown.
		b. Major sodium-water reaction.	b. Replace unit.	b. Not expected.
Turbine Generator	1	a. General overhaul.	a. Preventative maintenance.	a. Annual shutdown.

2.7.3 REACTOR MAINTENANCE (Continued)

2.7.3.4 Small Component Maintenance (Continued)

Reactor instrumentation will be accessible from the refueling room. Thermocouples, which monitor channel outlet temperatures, and most of the ion chambers* will extend from the large refueling plug in guide tubes or conduits.

Neutron shielding elements, which are placed around the reactor vessel, can be removed and replaced from the refueling room. The elements will be placed in a circular array which can be rotated so that each element is accessible.

*Some ion chambers placed for axial flux monitoring will be placed in conduits leading into the reactor from the side. These chambers will be maintained by replacement from outside the refueling room.

2.8 ECONOMICS DATA

2.8.1 FUEL CYCLE COST

~~2.8.1.1 Economic Ground Rules and Physics Input~~
~~2.8.1.2 Cost Estimates~~
~~2.8.1.3 Additional Considerations~~

2.8.2 PROCURED COST OF REACTOR EQUIPMENT

2.8.3 ESTIMATE OF TOTAL POWER PLANT CAPITAL COSTS

2.8.4 OPERATING AND MAINTENANCE COSTS

2.8.5 TOTAL POWER COSTS

2.8 ECONOMICS DATA

Although the principal incentive for fast reactor development lies in the fuel cycle costs attainable, the capital costs, particularly in the larger ratings, are not expected to provide an undue burden. These two aspects of cost are examined in this report, the first in some detail and the second in detail only with respect to the procurement cost of a number of specialized reactor items. A broad estimate is given of the balance of plant construction costs and of the Operating and Maintenance costs as well. It is, of course, understood that even the fuel cycle costs, as presented, are preliminary only. A final cost evaluation must take into account the interrelationship between fuel and capital cost items, and detailed optimization in a number of important areas including coolant temperature, steam conditions, thermal efficiency, reactor power density, and fuel management schedules is still ahead.

2.8 ECONOMICS D   :ontinued)

2.8.1 FUEL CYCLE COST

The estimated equilibrium fuel cycle cost is shown in Table 2.8.1.1. These results were obtained using the economic ground rules specified in Table 2.6.1.2. and physics data slightly different ~~more pessimistic~~ than those finally computed and presented in Section 2.5. A simplified block diagram flow path of the equilibrium fuel during the cycle is shown in Figure 2.8.1.1.

The core is power flattened radially by providing three equal volume zones of varied enrichment. These zones are costed separately in the analysis, and the results added together to obtain the results in Table 2.8.1.1.

TABLE 2.8.1.1

Fuel Cycle Costs, Mills/eKwhr

	Core	Axial Blanket	Radial Blanket	Reactor Total
Fabrication	0.248	0.052	0.044	0.344
Depletion	0.322	-0.430	-0.080	-0.188
Inventory	0.191	Negligible	Negligible	0.191
Recovery	0.070	0.093	0.030	0.193
Interest on Fuel Fabrication Costs	0.019	0.004	0.006	0.029
Total	0.850	-0.281	0	0.569

Unit Price, dollars/kg U + Pu

	Core	Axial Blanket	Radial Blanket	Reactor Average
Fabrication	298	41	75	119
Depletion	397	−350	-138	-82
Inventory	238	<1	0	61
Recovery	84	74	52	68
Interest of Fuel Fabrication Costs	23	3	11	11

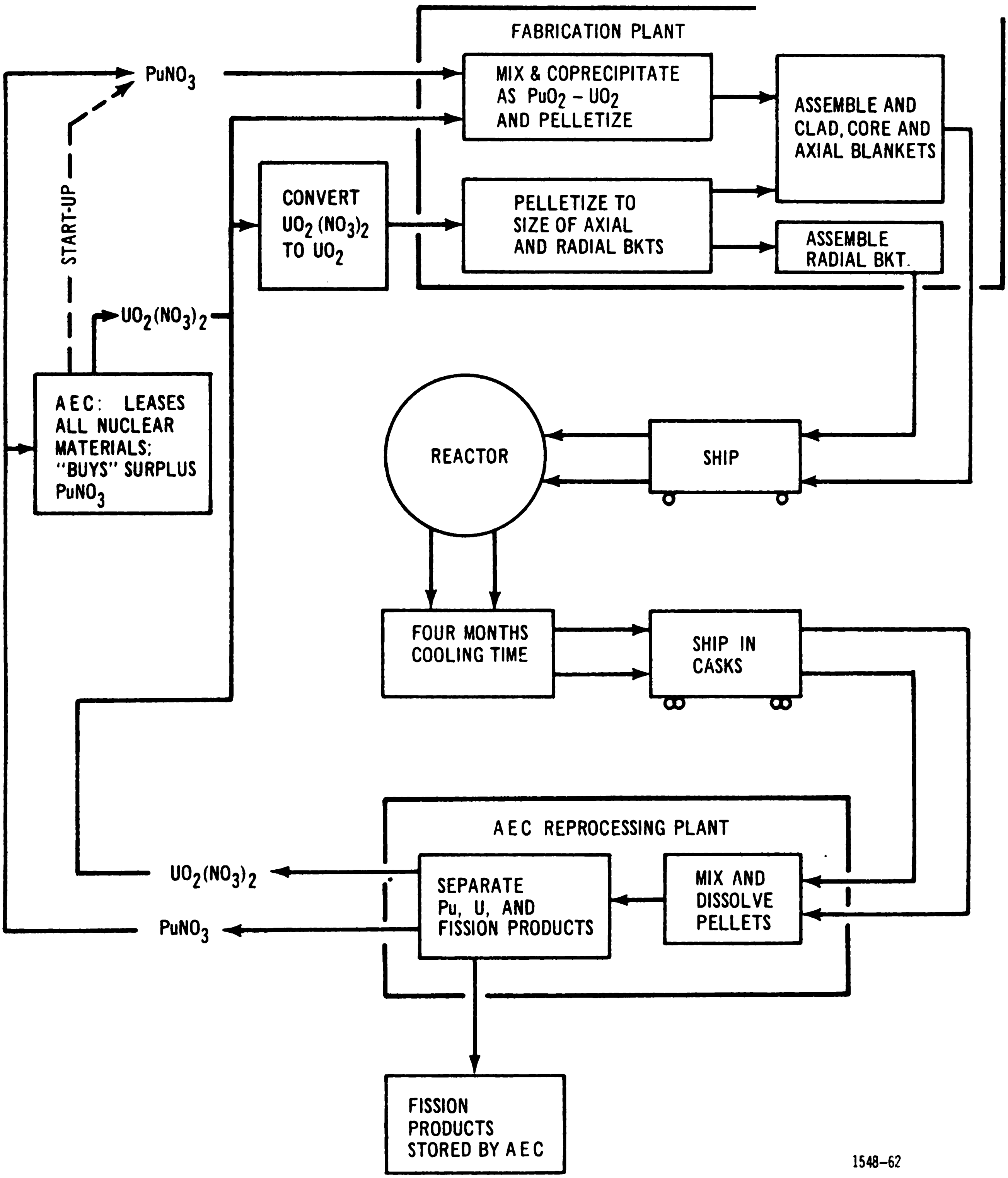


Figure 2.8.1.1. Fuel Cycle

TABLE 2.8.1.2

ECONOMIC GROUND RULES ~~AND PHYSICS INPUT~~

Fabrication - Estimate based on fabrication plant described in AEC Report GEAP-3824. Plant fabricates mixed UO_2-PuO_2 by cold pressing and sintering process. Cost includes conversion of fuel nitrate to oxide, pelletizing and encapsulation, materials, and a 1 percent loss of nuclear materials. The cost of the BeO rods is also included here in the fuel cycle as they are assumed to be replaced as frequently as the fuel rods with no salvage value.

Inventory (or use charge) - Based on AEC ownership of all fissile and fertile material. Annual interest rate of 4.75 percent is charged on material leased from AEC. Linear depreciation from initial to final core enrichment values is assumed. No inventory charge is made on Pu bred in blankets.

Depletion - Value of plutonium in form of nitrate is 10.00 dollars/gm of contained Pu-239 and -241. Value of depleted uranium (0.3 percent U-235 and 99.7 percent U-238) is 5.00 dollars/kg U as oxide and 3.00 dollars/kg U as nitrate (i.e., 2.00 dollars/kg U for conversion).

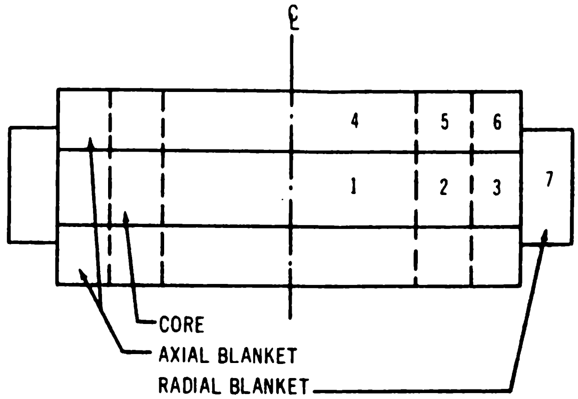
Recovery - Includes cost of shipping spent fuel in casks by rail from St. Louis to Augusta. Reprocessing done in AEC plant described in TID-7025 with plant rental of 17,000 dollars/day. The criticality limit for plutonium bearing fuels is assumed to be 67 percent of the concentration for U-235-enriched fuels because of the lower critical mass of plutonium. Nuclear material losses of 1 percent are incurred during reprocessing.

Interest on Fuel Fabrication Cost - Interest rate of 6 percent per annum is charged on book value of fabrication cost. Book value indicates a linear depreciation to zero during core residence.

2.8 ECONOMICS DATA

2.8.1 FUEL CYCLE COST (Continued)

The various zones of the core and blankets are assigned numbers as shown in the following sketch:



The reactor design data used to calculate the fuel cycle cost are summarized by zone in Table 2.8.1.3.. The load factor of 70 percent includes down time for refuelling. The power shown for each zone is the average distribution at mid-cycle (i.e., after 55,000 MWD/MT in the core). The average fuel lifetime is then estimated from the above load factor and average power.

Table 2.8.1.4 shows the calculation by zone of the fuel cycle cost. The results obtained for each category (fabrication, depletion, etc.) are in units of dollars/kg U+Pu in each zone. To convert dollars/kg U+Pu into mills/eKwhr, the following "factor" is applied:

$$\left[\text{dollars/kg}\right]_{\text{Zone}} \times \left[\frac{1000}{24} \cdot \frac{1}{(\text{MWe})_{\substack{\text{Total}\\\text{reactor}}}} \cdot \left(\frac{\text{MW}}{\text{MWD/MT}}\right)_{\text{Zone}}\right] = \left[\text{mills/eKwhr}\right]_{\text{Zone}}$$

The units to the above factor:

$$\left[\left(\frac{1000\ \text{mills}}{\$}\right)\left(\frac{\text{Day}}{24\ \text{hrs}}\right)\left(\frac{1}{\text{MWe}}\right)\left(\frac{\text{MWt}\quad\text{MT}}{\text{MWt}\quad\text{Days}}\right)\left(\frac{\text{MWe}}{\text{eKW}} \times \frac{\text{kg}}{\text{MT}}\right)\right]$$

2.8 ECONOMICS DATA

2.8.1 FUEL CYCLE COST (Continued)

The results may be summarized by zone as follows:

	dollars/kg						Mills/eKwhr					
Zone	Fab	Dep	Inv	Rec	IFF	Factor	Fab	Dep	Inv	Rec	IFF	Total
1	295	287	191	83.4	20.3	0.000303	0.0894	0.0870	0.0579	0.0253	0.0061	0.2657
2	297	401	207	84.5	19.7	0.000315	0.0936	0.1263	0.0652	0.0266	0.0062	0.3179
3	302	502	315	85.8	29.1	0.000216	0.0652	0.1084	0.0680	0.0185	0.0063	0.2664
4	41.2	-327	0.5	74.1	2.8	0.000458	0.0189	-0.1498	0	0.0339	0.0013	-0.0957
5	41.3	-319	0.5	74.0	2.7	0.000469	0.0194	-0.1496	0	0.0347	0.0013	-0.0942
6	41.3	-404	0.7	74.9	4.0	0.000323	0.0133	-0.1305	0	0.0242	0.0013	-0.0917
7	75	-138	1.1	52.2	11.3	0.000581	0.0436	-0.0802	0	0.0303	0.0066	0.0003
Avg. →	119	-82	61	68	11	Total →	0.343	-0.1884	0.1911	0.1935	0.0291	0.5687

TABLE 2.8.1.3

REACTOR DATA FOR FUEL CYCLE

Plant Parameters				
Net Electric Rating	MWe		1000	
Efficiency, Thermal	%		40	
Load Factor	%		70	
Core Fuel		Zone 1	Zone 2	Zone 3
Exposure, average	MWD/MT		110,000	(All zones)
Power, at mid-cycle MWt		800	830	570
Total Weight (uranium and plutonium only)	MT	4.25	4.25	4.25
Initial fissile content (Pu-239+241)	a/o	15.70	17.67	20.46
Final fissile content (Pu-239+241)	a/o	12.58	13.66	15.01
Cladding-O.D.	in.	0.25	0.25	0.25
Cladding-wall thickness	in.	0.015	0.015	0.015
Cladding-material			Stainless Steel	
Length - active fuel	ft	2	2	2
Length - gas plenum	ft	3	3	3
Length - Total (including blankets)	ft	8	8	8
Weight per rod	kg UO_2	0.138	0.138	0.138
Number Fuel Rods		35,250	35,250	35,250
Average Fuel Lifetime	yrs	2.29	2.21	3.21
Axial Blanket		Zone 4	Zone 5	Zone 6
Exposure, average	MWD/MT		8000(all zones)	
Power, at mid-cycle	MWt	88	90	62
Total Weight (uranium only)	MT	6.38	6.38	6.38
Initial fissile content (Pu-239+241)	a/o	--	Depleted	--
Final fissile content	a/o	3.29	3.21	4.06
Cladding - OD	in.	0.25	0.25	0.25
Cladding - thickness	in.	0.015	0.015	0.015
Cladding - material			Stainless Steel	
Length - UO_2	ft	2 × 1.5	2 × 1.5	2 × 1.5
Length - gas plenum	ft	3.0	3.0	3.0
Length - Total (including core)	ft	8.0	8.0	8.0

TABLE 2.8.1.3 (Continued)

Axial Blanket (Continued)		Zone 4	Zone 5	Zone 6
Weight per rod	kg UO_2	0.206	0.206	0.206
Number UO_2 rods		35,250	35,250	35,250
Average Lifetime in Reactor	yrs	2.29	2.21	3.21

Radial Blanket		Zone 7
Exposure, average	MWD/MT	4,300
Power, at mid-cycle	MWt	60
Total Weight, (uranium only)	MT	17.8
Initial fissile content (Pu-239+241)	a/o	Depleted
Final fissile content (Pu-239+241)	a/o	1.4
Cladding-OD	in.	0.50
Cladding-thickness	in.	0.025
Cladding-material		Stainless Steel
Length - UO_2	ft	3
Length - gas plenum	ft	1
Length - Total	ft	4
Weight per rod	kg UO_2	0.900
Number UO_2 rods		22,500
Average Lifetime in Reactor	yrs	5.0

TABLE 2.8.1.4
FUEL CYCLE COST ESTIMATE CALCULATIONS

Summary of Fabrication Plant Annual Costs in Dollars:

	Core			Axial Blanket			Radial Blanket
Fuel Zone Number:	1	2	3	4	5	6	7
Equipment Cost	105,050	108,200	74,550	2,200	2,250	1,550	28,300
Buildings Cost	51,900	53,400	36,900	5,800	6,000	4,100	13,200
Payroll Cost	350,300	360,500	248,300	21,500	22,300	15,300	276,300
O and M Materials Cost	94,700	97,600	67,300	11,000	11,400	7,900	22,000
Subtotal	601,950	619,700	427,050	40,500	41,950	28,850	339,800
G and A at 15 percent	90,290	92,960	64,050	6,100	6,290	4,330	50,970
Product Materials Cost	284,200	307,500	221,300	112,100	116,100	80,000	123,500
Shop Cost, Subtotal	976,440	1,020,160	712,400	158,700	164,340	113,180	514,270
Working Capital, R and D, Profit at 56 percent	546,810	571,290	398,900	88,900	92,030	63,380	287,990
BeO and Materials Cost	96,400	99,900	68,900	78,600	81,400	56,000	0
BeO Profit, etc. at 25 percent	24,100	24,980	17,200	19,600	20,350	14,000	0
TOTAL ANNUAL COST $	1,643,800	1,716,300	1,197,400	345,800	358,100	246,600	802,300
Annual Throughput kg U + Pu	5,570	5,770	3,970	8,360	8,660	5,960	10,700
Fabrication							
Unit Price dollars/kg U + Pu	295	297	302	41.3	41.3	41.3	74.9

Depletion

Value of Pu as nitrate 10.00 dollars/gm Pu fissile
0 dollars/gm Pu fertile

Value of depleted U as oxide - 5.00 dollars/kg U

Value of depleted U as nitrate - 3.00 dollars/kg U

TABLE 2.8.1.4 (Continued)

	Core			Axial Blanket			Radial Blanket
Fuel Zone Number:	1	2	3	4	5	6	7
Depletion (Continued)							
Initial fissile content (percent Pu fissile)	15.44	17.67	20.03	0	0	0	
Final fissile content (percent Pu fissile)	12.57	13.66	15.01	3.29	3.21	4.06	1.40
Initial value of Pu (dollars/kg U+Pu)	1,544	1,767	2,003	0	0	0	0
Final value of Pu (dollars/kg U+Pu)	-1,257	-1,366	-1,501	-329	-321	-406	-140
Initial value of U (dollars/kg U+Pu)	2	2	2	5	5	5	5
Final value of U (dollars/kg U+Pu)	-2	-2	-2	-3	-3	-3	-3
Depletion Unit Price (dollars/kg U+Pu)	287	401	502	-327	-319	-404	-138

Inventory or "Use Charge"

	1	2	3
Initial value, Ci, dollars/kg U Pu	1,546	1,769	2,005
Final value, Cf, dollars/kg U Pu	1,259	1,368	1,503
Tf-Te-Tr Yrs.	$\frac{1}{4}$-2.29-$\frac{1}{2}$	$\frac{1}{4}$-2.21-$\frac{1}{2}$	$\frac{1}{4}$-3.21-$\frac{1}{2}$
$\frac{1}{2}$(Ci)Tf =	193	221	250
$\frac{1}{2}$(Ci+Cf)Te =	3,210	3,465	5,630
(Cf)Tr =	629	684	751
Total	4,032	4,370	6,631
$\times$ 0.0475 = Unit Price, dollars/kg U+Pu	191	207	315

(No inventory is charged on Pu bred in blankets. Inventory on depleted U is negligible.)

TABLE 2.8.1.4 (Continued)

	Core			Axial Blanket			Radial Blanket
Fuel Zone Number:	1	2	3	4	5	6	7
<u>Recovery</u>							
Daily Plant Charge:	$17,000						
Daily Plant Throughput (mixed core R. Blanket)	570 kg U+ Pu						
Plant Rental, dollars/kg	29.8	29.8	29.8	29.8	29.8	29.8	29.8
Losses, dollars/kg	12.6	13.7	15.0	3.3	3.2	4.1	1.4
Shipping, dollars/kg	41.0	41.0	41.0	41.0	41.0	41.0	21.0
Unit Price, dollars/kg	83.4	84.5	85.8	74.1	74.0	74.9	52.2
<u>Interest on Fuel Fabrication Cost</u>							
Cf-Fabrication Unit Price, dollars/kg	295	297	302	41.3	41.3	41.3	75
Te - Core residence time, yrs.	2.29	2.21	3.21	2.29	2.21	3.21	5.00
Unit Price = ($\frac{1}{2} \times Cf \times Te \times 0.06$), dollars/kg	20.3	19.7	29.1	2.8	2.7	4.0	11.3

2.8 ECONOMICS DATA

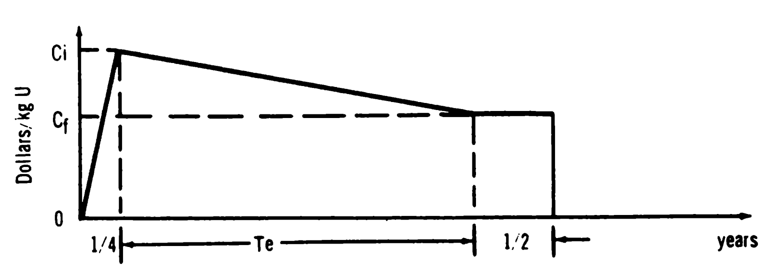
2.8.1 FUEL CYCLE COST (Continued)

EXPLANATION OF TABLE 2.8.1.4

The fabrication cost estimate is based on the model plant described by Collins.[2] The estimate is based on the annual cost of running the plant which is designed to fabricate mixed-oxide fuel rods to supply 3000 MWe of nuclear power. Thus, the 1000-MWe fast breeder reactor will utilize one-third of the plant's total capacity. A breakdown of the fabrication plant annual cost is included in Table 2.8.1.4. The cost of the BeO, its cladding, additional hardware, and profit are estimated separately (BeO at 20.00 dollars/pound) and added to the total cost before the fabrication unit price is calculated. Thus the cost of the BeO rods is included in the fabrication cost in a manner that assumes their replacement as the fuel is replaced. If the BeO rods are later designed to be re-usable, a saving of 22 dollars/kg U+ Pu of the core and 12 dollars/kg U+ Pu of the axial blanket fabrication unit prices may be realized.

For fuel depletion estimation, the value of plutonium as nitrate is fixed at 10.00 dollars/gm of contained Pu-239 and -241. There is no value attached to the fertile plutonium, thus the value of the nitrate does not depend on the fissile Pu fraction. The $PuNO_3$ is converted to PuO_2 in the fabrication plant. For depleted uranium, a value of 5.00 dollars/kg U is assumed for UO_2, and 3.00 dollars/kg U as $UO_2(NO_3)_2$. The fuel depletion charge is calculated as shown in Table 2.8.1.4, and is seen to include the cost of converting the uranium from nitrate to oxide for the blanket fuels. The core uranium is converted in the fabrication plant.

The fuel inventory (or use charge) is the interest due on the fuel leased from the AEC, and thus it is charged on the fuel on hand, both in- and out-of-pile. The inventory unit price is found by multiplying the interest rate (4.75 percent in this case) by the time averaged value of a batch of fuel. For the core the time for shipping, fabricating, and loading a batch of fuel is estimated as 3 months. The cooling, shipping, and recovery time is estimated to take 6 months. Both fuel acquisition and depletion are assumed linear. Thus a time-value plot of the core fuel looks like:



2.8 ECONOMICS DATA

2.8.1 FUEL CYCLE COST (Continued)

EXPLANATION OF TABLE 2.8.1.4 (Continued)

Where Ci and Cf are the initial and final values of the fuel, respectively, and Te is the average in-pile fuel lifetime in years. The inventory is then calculated for each zone as shown in Table 2.8.1.4. No inventory is charged on the plutonium that is bred in the blankets since it is not leased originally from the AEC. The inventory on the uranium in the blankets is found to total less than 0.001 mills/eKwhr, and is neglected.

The charge for fuel recovery includes the cost of shipping the exposed fuel in casks from St. Louis to Augusta, the rental fee for use of the AEC reprocessing plant, and a 1 percent loss of nuclear material throughput. The shipping cost is found by designing a lead cask to accommodate three spent core subassemblies, including their fission gas reservoirs, and estimating the total weight of the loaded cask. This is found to total over 50 tons, and includes the loading of about 1000 pounds of core uranium and plutonium. When this weight is shipped between the two points specified by the AEC at an estimated rail freight charge of 50 cents/ton mile, the shipping unit price is found to be 41 dollars/kg U+ Pu. Since a radial blanket subassembly is half as long as a core subassembly, approximately twice as much radial-blanket fuel can be shipped per cask and thus its shipping cost is almost halved.

The reprocessing rate is determined from the plot shown in Figure 2.8.1.2. One curve in the plot shows the reprocessing rate for U-235-enriched fuels as a function of their fissionable concentration as specified in TID-7025. The second curve is the assumed reprocessing rate for plutonium-bearing fuels, which is estimated as a 33 percent reduction of the criticality limitation concentration of fissionable material. The blanket and core fuel is assumed to be mixed together before separation, thus, an averaged fissionable concentration is calculated. For this purpose, the initial concentration of fissionable material in the core is added to the final concentration in the blanket regions. The cycle average is found to be 6.4 percent. The cost of the material losses is calculated from the composition of the fuels as discharged, and the dollar value in nitrate form.

The fabrication working capital charge is an allowance for the cost of the capital invested in the fabrication of fuel, which depreciates to zero during core residence. The average fuel life is used for each zone, and an annual interest of 6 percent is charged on the book value. The book value allows a linear depletion from the initial value to zero at the end of core life.

ADDITIONAL CONSIDERATIONS

The fuel cycle cost analysis presented above is based on current AEC economic ground rules. The following summarizes a number of possible changes to the total fuel cost if these ground rules were changed as indicated. The effect on fuel cost is given in mills/Kwhr.

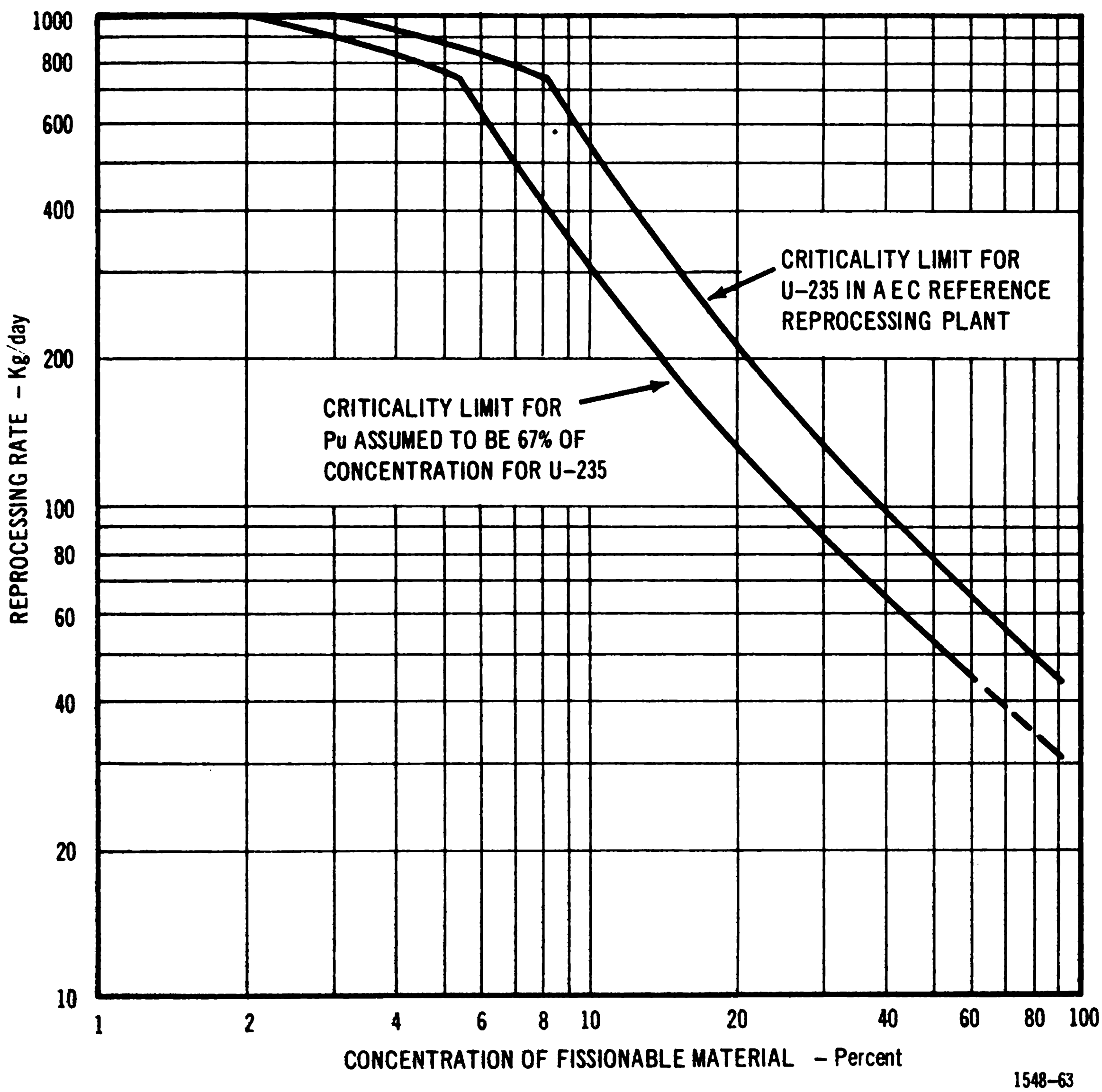


Figure 2.8.1.2. Permissible Reprocessing Rate

2.8 ECONOMICS DATA

2.8.1 FUEL CYCLE COST (Continued)

ADDITIONAL CONSIDERATIONS (Continued)

1. Reprocess each zone separately. . . . . effect: +0.048
   When the daily throughput to the AEC reprocessing plant was calculated, a smeared composition was used for the mixed fuel pellets. If it is assumed that the fuels of each zone (1 through 7) are reprocessed as individual batches, the net effect is a higher reprocess charge due to logarithmic function of Figure 2.8.1.2. However, there are alternate reprocessing plant concepts available which could result in significant economies. (1)

2. Lower value on plutonium. . . . . Depletion charge effect: +0.062
   Economic predictions of the commercial value of plutonium show that by 1975 the value of recycled reactor plutonium will not be higher than 6.60 dollars/gm plutonium fissile and 1.10 dollars/gm plutonium fertile. Since the net depletion charge was previously a credit, lower plutonium values tend to increase the depletion charge.

3. Private ownership of Fuel. effect: +0.170
   With private ownership of fuel, the interest rate for fuel inventory may be as high as 10 percent per year. The inventory charge must also be extended to include the plutonium bred in the blankets. The net effect, including 1975 plutonium values (see above), is an increase of about 0.17 mill/eKwhr.

4. Recycle BeO. . . . . maximum effect: -0.033
   The BeO in the channels was assumed to be replaced when the fuel was replaced. Further analysis may show this to be unnecessary. The maximum saving would be realized if the BeO lasted the life of the reactor. The above reflects such an estimation.

5. Reaching Equilibrium Core. . . . . effect: +0.110
   To reach a core that is in equilibrium with respect to isotopical composition, some of the initial batches of fuel may have higher compositions, and may not be burned to the average exposure of equilibrium fuel. This increases the pro rata cost of both fabrication and reprocessing.

6. Spare Fuels and Incomplete Burnup Defective Fuels. . . . . . .undetermined

Summary:
The effect of the above considerations would be to increase the total fuel cycle cost from 0.559 to 0.923 mill/eKwhr.

2.8 ECONOMICS D. ontinued)

2.8.2 PROCURED COST OF REACTOR EQUIPMENT

The procured cost of equipment items related to the reactor and the integral primary coolant recirculation loop are detailed in Table 2.8.2.1. The items covered include all of the reactor components excepting the fuel assemblies and their associated flow channels. Also included are the neutron shielding, the thermal shielding, control system, the primary coolant loop, and the enclosing sodium tank. The costs of the major stainless steel fabrications were estimated on a "dollars/pound" basis with the applicable unit cost varying with complexity and functional requirement as indicated in the table. The costs of the more elaborate hardware items were estimated on a fabrication shop basis.

TABLE 2.8.2.1
REACTOR EQUIPMENT COST SUMMARY

	Number Required	Material	Estimated Weight Tons	Dollars/Pound Fabricated	Total Direct Cost 1000 Dollars
1. Reactor Vessel					
Vessel	1	304	86	4.00	688
Downcomer	1	304	25	1.00	50
Lid and Drive Plug	1	304	29	4.00	232
Vessel Lid Bolt Extension					33
Instrumentation Access Hardware					16
Subtotal					1,019
2. Vessel Internals					
Thermal Shields	3	304	95	0.70	133
Grid Plate and Support					91
Upper Fuel Hold Down					24
Material Subtotal					248
(Installation					200)
3. Reactor Control System					
Control Drive Assemblies	85				3,003
Control Drive Guide Tubes	97				97
Control Drive Seals	85				84
Control Blade Assemblies	85				217
Control Drive Extensions					67
Subtotal (Material)					3,468
(Installation					300)

TABLE 2.8.2.1 (Continued)

	Number Required	Material	Estimated Weight Tons	Dollars/Pound Fabricated	Total Direct Cost 1000 Dollars
4. Primary Loop Structure					
Sodium Tank	1	304	192	1.00	384
Main Support Structure	1	304	130	2.00	520
Pump Discharge Pipes	6	304	7	0.75	10
Pump Check Valve Mounting	6	304	-	-	8
Intermediate Heat Exchanger Inlet Flow Plenum	6	304	20	1.00	40
Pump Wells	6	304	19	0.75	28
Intermediate Heat Exchanger Wells	6	304	145	0.75	217
Instrumentation					
Flow Ducting Support Ring	1	304	10	1.00	20
Intermediate Heat Exchanger Flexible Seals	6	-	-	-	15
Pump Flexible Seals	6	-	-	-	10
Sodium Service Unit Flexible Seals	1	-	-	-	25
Refueling Skirt	1	304	9	0.50	9
Insulation	-	-	-	$8/ft^2$	76
Heater Elements	-	-	-	-	25
Subtotal					1,387
5. Shielding (external to reactor vessel)					
Neutron Shield Support Frame					272
Neutron Shield Rods					819
Neutron Shield Frame	1	304	35	0.75	52
Neutron Shield Frame Support	1	304	5	0.75	7
Thermal Shield	3	304	120	0.50	120
Subtotal					1,270

TABLE 2.8.2.1 (Continued)

	Number Required	Material	Estimated Weight Tons	Dollars/Pound Fabricated	Total Direct Cost 1000 Dollars
6. Fuel Handling					
Fuel Storage Drum Well	1	304	8	0.75	12
Storage Drum Support	1	304			6
Fuel Storage Exit Seal	1	-	-	-	3
Refueling Cell Windows	2				25
Refueling Cell Periscope	1				15
Refueling Cell Jib Crane (3T)	1				20
Refueling Cell Main Hoist (50T)	1				15
Refueling Cell Manipulators (1T)	3				210
Refueling Cell Miscellaneous Tools					100
Refueling Exit Room					
Refueling Exit Manipulator	1				70
Refueling Exit Window	1				10
Refueling Exit Hoists	1				10
Fuel Transfer Tube	1				13
Shipping Cask Hoist (50T)	1	-	-	-	15
Subtotal (Material)					524
(Installation					300)

Total Procured Cost of Reactor Equipment (1-6)	7,920
Additional Installation Costs	800
Reactor Engineering	3,500
Total	12,220
Constr. OH, Contingency, Interest during Constr.	3,880
Grand Total	16,100

2.8 ECONOMICS DATA (Continued)

2.8.3 ESTIMATE OF TOTAL POWER PLANT CAPITAL COSTS

The reactor equipment costs from Table 2.8.2.1 are summarized in Table 2.8.3.1 together with the direct installed costs of the remaining major plant structural and equipment items. Here the additional costs have been obtained by scaling up from the plant costs for a generally similar 565-MWe (net) plant described previously.[3] The total cost shown includes all construction, overhead and indirect costs, contingency, and interest during construction.

TABLE 2.8.3.1
CONSTRUCTION COSTS
1000-MW FAST OXIDE REACTOR

Item	
Land	$ 220
Site Improvements	3,990
Controls and Instrumentation	2,970
Turbine-Generator Building	8,200
Turbine-Generator Unit	27,000
Condenser Circulating Water System	7,500
Accessory Electrical Equipment	8,750
Feedwater System and Piping	11,800
Subtotal "nonnuclear"	$ 70,430
Reactor Equipment (Table 2.8.2.1)	16,100
Cover Gas System	850
Primary Sodium Pumps and Drives	3,280
Intermediate Heat Exchangers	4,150
Secondary Sodium Pumps and Drives	2,740
Secondary Sodium Piping and Valves	610
Steam Generators	19,800
Subtotal "nuclear steam supply"	$ 47,530
Reactor Building	7,100
Total Plant Cost ($1,000)	$125,060

2.8 ECONOMICS DATA (Continued)

2.8.4 OPERATING AND MAINTENANCE COSTS

With an operating and maintenance staff of 112 men, the annual payroll costs of the 1000-MW fast oxide reactor would approximate $1,150,000. This includes the salaries itemized in Table 2.8.4.1 together with the appropriate benefits and payroll costs in the amount of 20 percent. These payroll costs, together with an estimated $670,000 for supplies and $335,000 for nuclear insurance, give a total annual operating and maintenance cost of $2,157,000. On the basis of 6120 hours per year operation (70 percent plant operating factor), the operating and maintenance contribution to total power cost is 0.34 mill/Kwhr.

TABLE 2.8.4.1

ESTIMATED OPERATION AND MAINTENANCE EXPENSE

		Annual Cost
Supervision		
Station Superintendent		17,000
Operations Supervisor		15,000
Maintenance Supervisor		14,000
Engineering Supervisor		14,000
Services Supervisor		12,000
Operation		
Shift Supervisor	5 at $11.000	55.000
Chief Operator	5 at 10.000	50.000
Operators	20 at 9.000	180.000
Helpers	20 at 7.500	150.000
Planner/Schedule		7.000
Maintenance		
Instrument Mechanics	9 at 9.000	81,000
Electricians	5 at 8.000	40.000
Mechanics	12 at 8,500	102,000
Helpers	3 at 6.000	18.000
Engineering		
Nuclear Engineering	1 at 11.000	11.000
Electrical Engineering	2 at 11.000	22.000
Chemical Engineering	2 at 11.000	22.000
Mechanical Engineering	2 at 11.000	22.000
Junior Engineering	3 at 8,000	24.000
Technician	2 at 7.000	14.000

TABLE 2.8.4.1 (Continued)

Services		Annual Cost
Clerks	4 at 6,000	24,000
Patrolmen	5 at 6,000	30,000
Janitors	4 at 6,000	24,000
Storekeepers	2 at 6,000	12,000
		960,000
Total Base Payroll		960,000
Fringe Benefits	20 percent	192,000
		1,152,000
Operation and Maintenance Materials and Supplies		670,000
Subtotal Operation and Maintenance Cost		1,822,000
Nuclear Liability Insurance		260,000
Government Indemnification		75,000
TOTAL OPERATION AND MAINTENANCE COST		2,157,000

2.8 ECONOMICS DATA (Continued)

2.8.5 TOTAL POWER COSTS

The reference plant design has been based on a 2,500-MWt rating and an estimated plant availability of up to 92 percent based on biannual refueling shutdowns (see Section 2.7.2.1). It appears that the net electrical output of the reference plant may be nearer to 1100 MWe than the nominal 1000-MW rating. This results in approximately a 10 percent range in the values of all components of the power cost expressed in mills/Kwhr.

		Dollars per Annum
Capital Charges		
Capital cost of $125,060,000 at 14 percent per annum	=	$ 17,508,400
Operating and Maintenance Costs		
Staff of 112 plus supplies and insurance	=	2,157,000
Fuel Cycle Charges		
4-3/4 percent Carrying Charges on approximately 2,200 kgs plutonium in fuel cycle*	=	1,042,000
6 percent Carrying Charges on one-half of $\$6.5 \times 10^6$ core and blanket fabrication cost	=	195,000
SUBTOTAL FIXED ANNUAL COSTS		$ 20,902,400

Fuel Burnup Costs

Fuel processing less plutonium credit	=	105 dollars/kg
Average exposure of fuel and blanket materials	=	30,000 MWD/T
Burnup Costs	=	$\frac{105 \times 1000}{33{,}000 \times 1.1}$
	=	2.93 dollars/MWDt

Total for Plant Load Factors of:	50%	60%	70%	80%	90%
( ) - MWDt/yr $\times 10^{-5}$	(4.55)	(5.45)	(6.37)	(7.28)	(8.19)
Annual Costs 10^6 dollars/year = 20.902+	1.32	1.59	1.85	2.11	2.38
=	22.22	22.49	22.75	23.01	23.28
at 1000-MWe net output mill/Kwhr power cost =	5.08	4.28	3.71	3.27	2.96

* Core plutonium only

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3. Dodge, H. E., Kendall, W. W., Sangster, W. A., and Silvester, A. G., "Conceptual Mechanical Design of a Fast Ceramic Power Reactor," GEAP-4226, April, 1963.