CORE DESIGN AND CHARACTERISTICS

FOR THE

CONSOLIDATED EDISON REACTOR

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INTRODUCTION

The general purpose of this report is to present the major characteristics and design of the Consolidated Edison Thorium Reactor (CETR) core.

Four major design changes have taken place since the last report:

1) The cladding material has been changed from Zircaloy-2 to stainless steel.

2) The use of a combination of burnable and soluble poison is contemplated for additional reactivity control.

3) A shift from a single zone, uniformly loaded core, to a multizone core to obtain better power flattening characteristics is planned.

4) The fuel element fabrication technique has been changed from a tube sheet type structure to a brazed assembly with ferrule spacers, and the power level has been increased from 500 MW to 585 MW. The full power core reactivity life is computed to be 600 days.

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2.1 REACTOR DESCRIPTION

2.1.1 Reactor Internals

The Consolidated Edison reactor is a thorium converter type, fueled with a mixture of thorium oxide and fully enriched uranium oxide contained in stainless steel tubes. The active portion of the reactor core is composed of 120 box-type fuel elements, each approximately 6 inches square.

The primary system flow rate is 53.9×10^6 lbs. of water per hour at a pressure of 1500 psig; average reactor water temperature at the inlet is 485 F and at the outlet is 521 F at 585 MW full power. Approximately 87% of the total flow is available for heat transfer from the fuel elements. The remaining flow is diverted to cool the control rods, and thermal shields. A conical shaped lower plenum baffle with properly sized orifices insures correct flow distribution to the core section and thermal shields. The upper plenum baffle is also orificed for the discharge of the coolant from the core to the reactor outlet nozzles.

The composition of the core is determined by the requirements of heat transfer, structural integrity, excess reactivity for adequate life, and the need for an adequate negative temperature coefficient. Approximately 76 per cent of the core volume is required for coolant flow and the inventory of approximately 18,000 kilograms of urania-thoria mixture. The remaining core volume is stainless steel and Zircaloy-2. The total fuel loading is about 950 kilograms of highly enriched uranium-235. (93.5% U-235)

Three alternate layers of steel and water are arranged between the core and pressure vessel to serve as a neutron reflector, and as thermal shields to reduce the effects of neutron bombardment and gamma radiation on the pressure vessel walls. The pressure vessel which contains the entire assembly is a cylindrical shell of carbon steel having an inside diameter of 9 ft. - 9 in., with a minimum wall thickness of 6.95 in. The inside wall of the pressure vessel is lined with 0.109 inch of stainless steel.

The core layout showing placement of fuel elements, thermal shields and reactor vessel wall is shown in Figure 1. There are 28 load support tubes which provide additional hold-down force when the primary loop is cold and the primary pumps are operating.

2.1.2 Fuel Element Design

The reactor core consists of 120 fuel elements. Flow control and distribution and structural support is obtained by use of end transition pieces and cans. Each element is nominally 135-5/8 inches long and 6.139 inches square. A cross section of the element is shown in Figure 2 and an over-all view in Figure 3.

The fuel is in the form of high density pellets of close dimensional tolerance, contained in stainless steel tubing. Each tube is closed at the end with a suitable end cap. Insulating pellets are placed at the end of the active fuel to reduce thermal stresses in the end caps. The annulus between the pellets and the tubing is filled with helium. Each subassembly contains 206 tubes, spaced by ferrules (short, thin tubular sections) in a square array, with an offset for the control rod channel (refer to Figure 2). The ferrules are located with approximately eight inches between centers in planes normal to the longitudinal axis of the element. A thin strap or band is wrapped around the bundle at each plane of ferrules to support the outer row of tubes, and to space the fuel rods from the can wall. The entire subassembly is brazed as a unit.

The fuel element can is made of Zircaloy-2. The can is an envelope which surrounds the tube subassemblies. It is fitted at each end with transition pieces. A spring in each fuel element locks in the top transition piece, provides fuel element hold-down against hydraulic forces, and allows for thermal expansion of the subassemblies.

2.2 THERMAL AND HYDRAULIC DESIGN

2.2.1 General Design Considerations

Major thermodynamic characteristics of the CETR core are determined by the requirements of heat transfer, hydraulics, mechanical and optimum nuclear performance for steady state operation. Pertinent design data are shown in Table 1.

The thermal design of the core includes a comprehensive analysis of temperature distributions, local boiling, two phase pressure drops, and burnout characteristics for the hot channel. A hot channel is defined as that channel which has the radial and axial power distribution factors as well as manufacturing tolerance factors applied, where as only power distribution factors are included in the analysis of a nominal channel.

Burnout characteristics of the hot channel limit the maximum safe power at which the reactor can operate. When two-phase flow develops in a channel, the mass velocity is reduced in this channel because of parallel channel pressure drop effects. At a power substantially higher than the rated power of the reactor, the combination of increased heat flux and decreased flow resulting from steam formation can cause a failure of the heat transfer mechanism resulting in burnout.

Existing data and correlations resulting from burnout tests performed at various universities, national laboratories and other AEC sponsored facilities are used in determining the burnout heat flux. Scattering of experimental data, with respect to both local boiling and boiling with net steam generation, makes it difficult to predict the exact burnout heat flux. Based on the most conservative data available and with the most adverse combination of all power distribution and manufacturing tolerance factors, it has been determined that burnout will not occur before 131 per cent full power is attained.

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TABLE 1

REACTOR DESIGN AND PERFORMANCE CHARACTERISTICS
THERMAL, HYDRAULIC AND MECHANICAL

I. Fuel Element Materials

a) Fuel	ThO_2 - UO_2 mixture
b) Cladding	304 Stainless Steel-boron modified
c) Transition pieces	304 Stainless Steel
d) Springs	Inconel-X
e) Can	Zircaloy-2
f) Ferrules	304 Stainless Steel

II. Geometry (all dimensions are nominal)

Fuel rod O.D. (in.)	0.3125
Clad thickness (in.)	0.020
Fuel pellet (in.)	0.2690
Fuel rod pitch (square) (in.)	0.3810
No. of fuel rods per element	206
Active fuel length per rod (in.)	95.25
Total active fuel length (ft.)	196,215
Ferrule O.D. (in.)	0.2260
Ferrule I.D. (in.)	0.1960
Ferrule length (in.)	0.500
No. of ferrule planes	14
Strap width (in.)	0.500
Strap thickness (in.)	0.015
Element size (in.)	6.135
Element pitch (in.)	6.3225
Can thickness (in.)	0.180
No. of elements per core	120
Configuration	Offset
Control rod pitch (in.)	12.645
Control rod blade width (in.)	7.50
Control rod blade thickness (in.)	0.3125
Max. core diameter (in.)	84.547

III. Fluid Flow

Total reactor flow (lb./hr.)	53.9×10^6
Leakage flow (lb./hr.)	7.1×10^6
Velocity (ft./sec.) (inside elements)	21.4
Total area inside element (in.^{2}/element)	30.53
Flow area (eff.) (in.^{2}/element)	14.73
Flow area in ferrule region (in.^{2}/element)	12.57
Total eff. flow area thru elements (ft^2)	12.28
Equivalent diameter for pressure drop (ft.)	0.0218
Equivalent diameter of unit cell (ft.)	0.0232
Pressure drop (grid plate to grid plate) (psi)	35.9
Pressure drop (fuel region) (psi)	24.5
Operating pressure (psig)	1500

TABLE 1 (Cont'd)

IV. **Heat Transfer**

Reactor power (MW)	585
Heat flux (average B/hr-ft^2 @ 585 MW)	124,376
Heat transfer coefficient (B/hr-ft^2-F)	6860
Heat transfer area (ft^2)	16,053
Reactor inlet temperature + 585 MW (F)	484.5
Average core Δ T (F) (@ 585 MW)	36.52

Maximum Temperatures (calculated)

	100% Power (585 MW)	125% Power (732 MW)
Nominal Channel		
Bulk coolant temp. (F)	552	569
Clad surface temp. (F)	582	605
Fuel pellet temp. (F)	4044	4490
Hot Channel		
Bulk coolant temp. (F)	572	596
Clad surface temp. (F)	605	606
Fuel pellet temp. (F)	5056	5780

V. **Nuclear**

Equivalent core diameter (cm)	197.297
Total core volume (cu.in) (97.15 in. length)	466,091

Material	Volume (cu. in.)	Volume Fraction
Water	0.2227×10^6	0.4779
Fuel	0.1338×10^6	0.2872
Stainless Steel	0.0459×10^6	0.0986
Zr-2	0.0487×10^6	0.1045
Helium	0.0039×10^6	0.0084
Inert Pellets	0.0007×10^6	0.0014
Control Rod	0.0102×10^6	0.0220

Metal-Water Ratio (Homogeneous) 1.093

A detailed analysis of the thermal limitations is presented in the Appendix A.

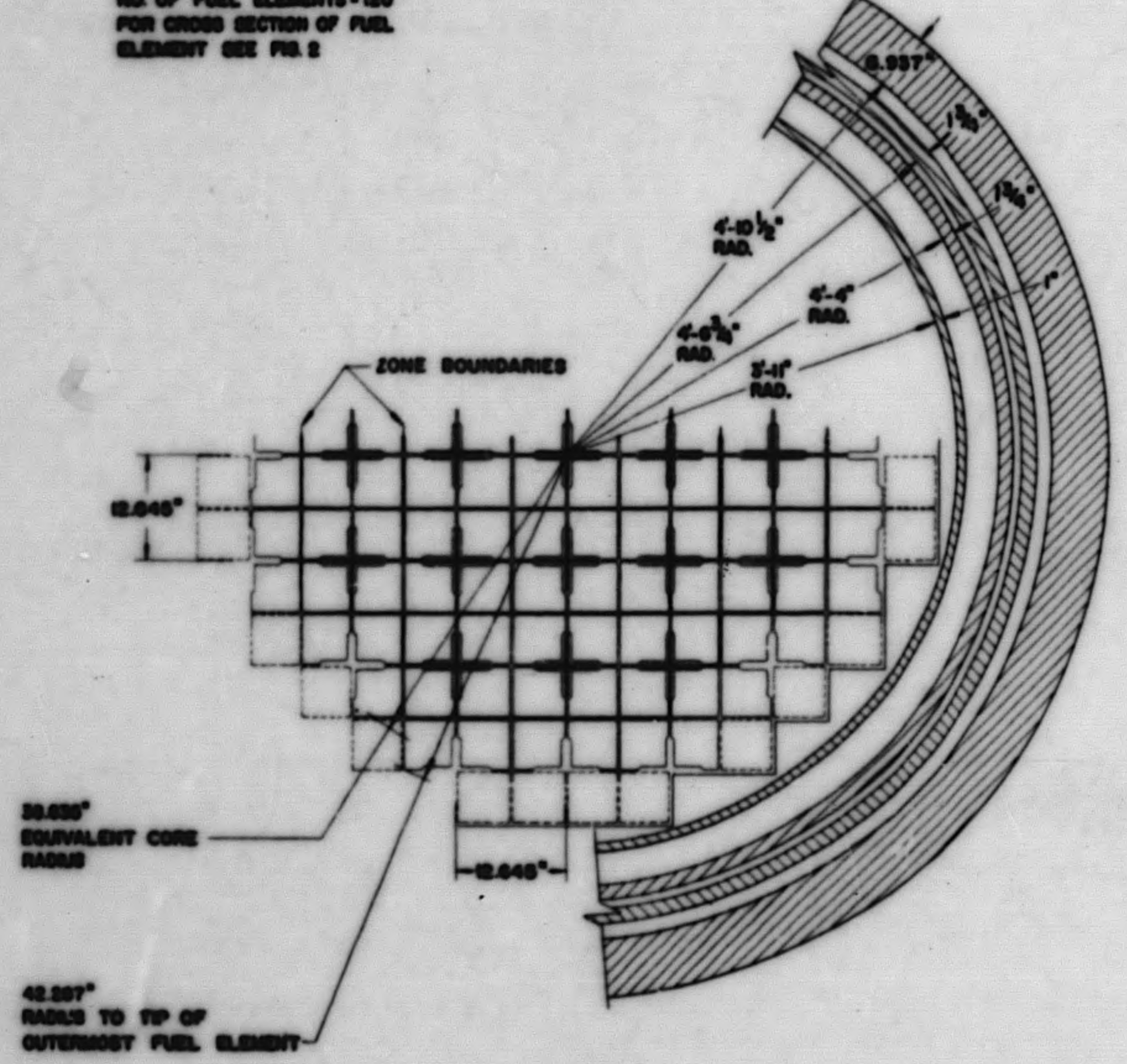
The following design criteria are established for the core:

a) Heat transfer surface burnout lies above the maximum steady state or transient power permitted by the control and safety systems.

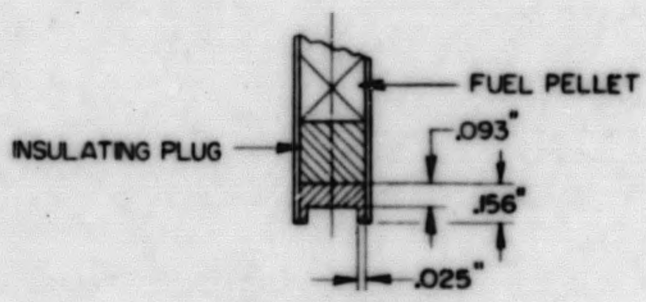
b) The fuel should not melt at the maximum power level permitted by the control and safety systems.

These criteria limit the bulk water temperature, clad surface temperature and internal fuel pin temperature. A temperature chart based on nominal dimensions is shown in Figure 4.

NOTE:
NUMBER OF CONTROL RODS = 21
NO. OF FUEL ELEMENTS = 120
FOR CROSS SECTION OF FUEL
ELEMENT SEE FIG. 2



LAYOUT OF CORE FOR CONSOLIDATED EDISON REACTOR
SHOWING PLACEMENT OF FUEL ELEMENTS, THERMAL SHIELDS,
AND REACTOR VESSEL WALL
FIG. 1



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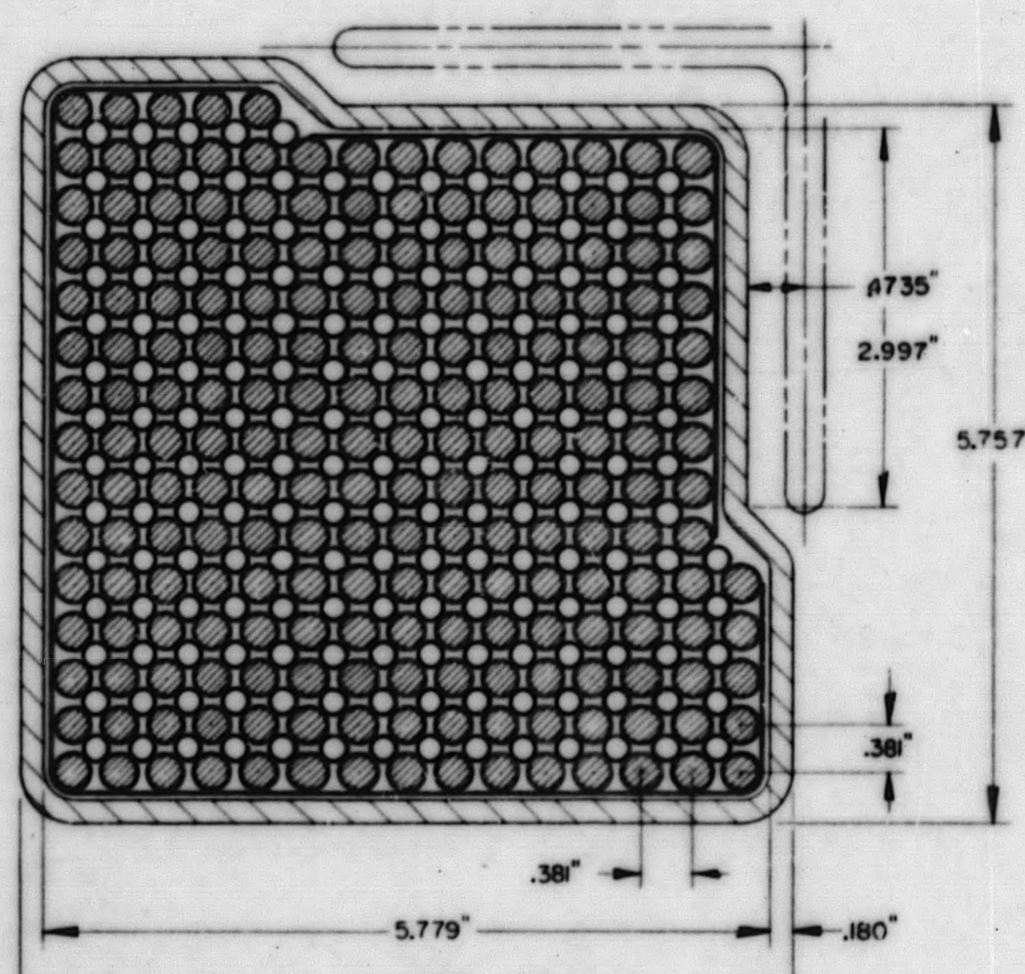
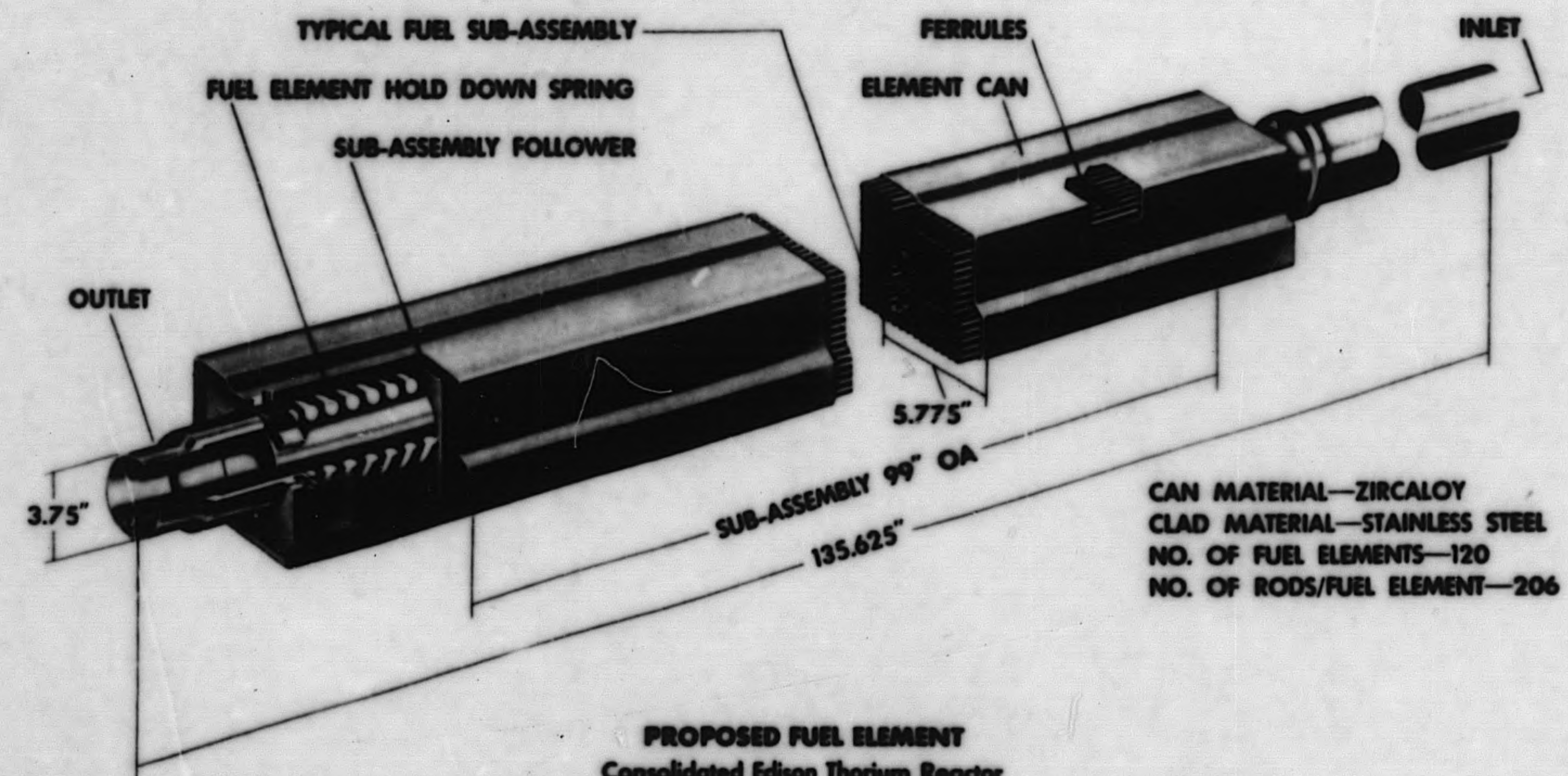


CROSS SECTION OF FUEL ELEMENT
FOR
CONSOLIDATED EDISON REACTOR
FIG.2



PROPOSED FUEL ELEMENT
Consolidated Edison Thorium Reactor

FIGURE No. 3

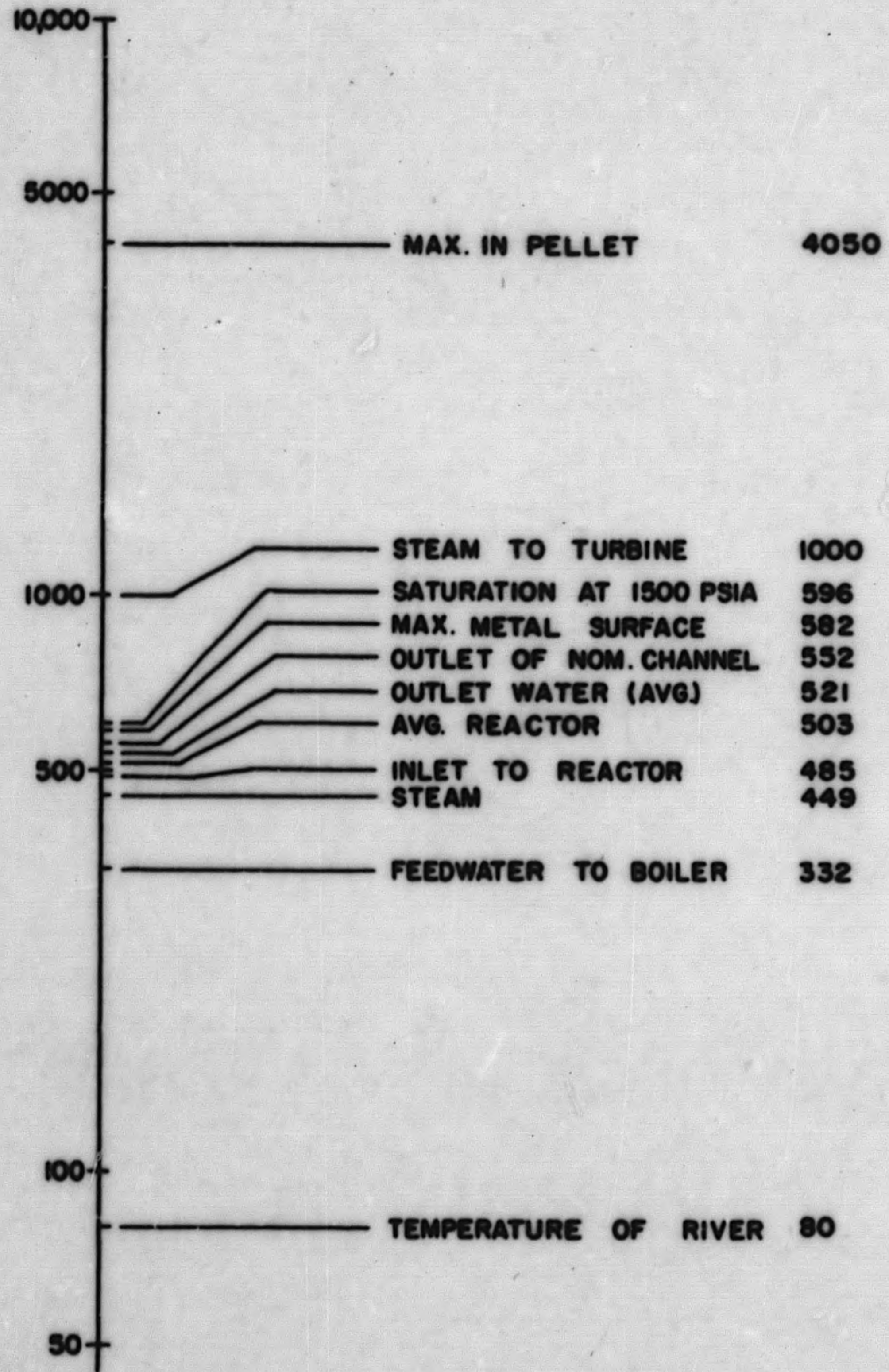


FIGURE 4

CONSOLIDATED EDISON REACTOR
TEMPERATURES (F) AT 585 MW
BASED ON NOMINAL DIMENSIONS

2.3 NUCLEAR DESIGN

2.3.1 General Design Considerations

The change from Zircaloy-2 to stainless steel as a cladding material, and the results of critical experiments have led to major revisions in the nuclear design of the core.

The principal effect of the use of stainless steel is to increase the critical mass, and thus the initial loading of U-235 to obtain the design life of 600 full power days. Information from the critical experiments indicates that the control rod worth is insufficient to fully compensate for the necessary excess reactivity, and that the desired maximum to average radial power factors in the uniformly fueled core would be exceeded for some rod configurations. Consideration of these factors has made it necessary to utilize zoned loading of U-235 and to provide additional means of reactivity control. The available supplementary control methods are burnable poisons and soluble poisons.

A burnable poison is a highly absorbing isotope such as boron-10 which is initially incorporated in the fuel element. This isotope acts as a poison to supplement the control rods in holding excess reactivity and gradually burns out by neutron absorption.

Soluble poison reactivity control is obtained by dissolving a highly absorbing poison such as boric acid in the coolant water and removing it chemically, or by dilution, instead of by nuclear burnout.

The current reference design for the CETR utilizes a combination of burnable and soluble poison for supplemental control. Burnable poison will be used to supplement control rods in compensating for burnup of fuel and buildup of fission product poisons. This is accomplished by alloying natural boron with the stainless steel used for fuel element cladding.

To assure that the core will produce the specified power without risk of burnout in the hot channel, the ratio of maximum to average power in any channel must be held within a certain value. The over-all power peaking factor is made up of three individual factors: radial power peaking, axial peaking, and localized peaking near control rod channels and between fuel element cans. Uniformly loaded cores exhibit maximum to average radial power ratios which alone approach the over-all allowance. This has been demonstrated on the CETR plate type critical experiments. To obtain more favorable radial power distribution, radial zone loading of fuel has been adopted for the CETR. Zone loading is accomplished by dividing the core into radial regions with different U-235 concentrations. The lowest concentration is at the core center, increasing to the highest concentration around the outside.

In the reference core there are three radial zones. The central zone has 32 fuel elements, and the intermediate and outside zones each have 44 fuel elements. The ratios of U-235 concentrations (atoms/cc) in the three zones are about 3/4/5.

Some of the more pertinent nuclear characteristics of the core are listed in Table II.

TABLE II

REACTOR DESIGN AND PERFORMANCE CHARACTERISTICS - NUCLEAR

Initial Concentrations	atoms/cc x 10^{-20} Inner zone	Middle zone	Outer zone	kg (total)
Uranium-235 (fuel)	2.35	3.14	3.92	950 (as metal)
Thorium Oxide (fertile material)	58	58	58	16,860
Water (moderator & coolant)	126.2	126.2	126.2	
Zircaloy-2 (structural)	54.3	54.3	54.3	
Stainless Steel (cladding)	83.9	83.9	83.9	

Concentrations at End of 600 days of Full Power Operation	Inner zone	Middle zone	Outer zone	kg (total)
Uranium-233	.57	.57	.42	150
Uranium-234	.036	.035	.016	8.3
Uranium-235	1.13	1.70	2.7	556
Uranium-236	.23	.28	.25	75.9
Protactinium-233	.06	.06	.04	15.2
Thorium Oxide	57.1	57.1	57.4	16,630

Average thermal flux at 585 MW 2×10^{13} n/cm^2-sec.

Over-all conversion ratio 0.5

2.3.2 Reactivity Requirements and Reactivity Control

In addition to the cold clean critical mass enough fuel must be placed in the reactor to provide reactivity for temperature rise, equilibrium xenon and samarium, doppler effect and the consumption of fuel and buildup of fission products. Enough control must be provided for these reactivity effects plus an additional amount of control for shutdown to insure that the reactor can be made several per cent subcritical in its most reactive condition (cold, clean, fully loaded).

A breakdown of various reactivity components in terms of per cent excess reactivity is shown in Table III.

TABLE III

REACTIVITY REQUIREMENTS

Component	$\frac{\Delta k_{eff}}{k_{eff}}$ (%)
1. Shutdown	3
2. Equilibrium Xenon and Samarium	3
3. Temperature (Inc. Doppler to 500 F)	5.9
4. Doppler Effects (From hot zero power to full power	1.5*
5. Shim for Lifetime	9.7**
	23.1

*Based on an average oxide temperature of 1115 F and a coefficient of .013% change in the thorium resonance integral per degree F.

**Includes complete transient xenon override for about 540 days

The distribution of the reactivity requirements between control rods, soluble and burnable poisons will be determined after the completion of the experimental program.

2.3.3 Reactivity Coefficients

a) Temperature Coefficient

The over-all temperature coefficient of reactivity, exclusive of the Doppler coefficient, has been calculated to be

$$-1.8 \times 10^{-4} \ (\pm 1 \times 10^{-4}) \quad \Delta k/F \quad \text{at } 500 \text{ F}$$

Considerable experimental work is being planned to evaluate the uncertainties in some of the parameters used to compute the temperature coefficient.

b) Power Coefficient

The power coefficient of reactivity occurs as a result of the Doppler broadening of thorium resonances as the fuel pin heats up to operating conditions. This coefficient has been calculated to be

$$-2.6 \times 10^{-5} \ (\pm 1.5 \times 10^{-5}) \quad \Delta k/F$$

c) Pressure Coefficient

The pressure coefficient of reactivity at the operating temperature and pressure of 500 F and 1500 psi is about $+2 \times 10^{-6}$ Δk/psi.

APPENDIX A

THERMAL AND HYDRAULIC ANALYSIS

1. Description of Power Distribution

Average reactor power is 585 MW and appropriate factors for radial and axial power distributions are considered in the design calculations. A reactor over-all maximum to average power distribution ratio of 3.0 is used to allow for non-uniform power distribution in the core. This factor is the product of an axial maximum to average ratio of 1.5 and a radial maximum to average ratio of 2.0. The radial maximum to average factor includes a local neutron flux peaking factor and applies to non-uniform loading of fuel.

Experiments presently being performed at the critical experiment laboratory will establish final design power distribution factors. The possibility of local power peaking at the fuel element boundaries is under study as well as power peaking due to control rod positioning.

2. Design Criteria

Thermal aspects of the core design must insure that no damage to the reactor will occur as a result of steady state or transient operation. This design is analyzed on the basis of the worst expected combination of power distribution, manufacturing tolerances, empirical uncertainties, heat balance and instrumentation inaccuracies. These effects are included in the thermal calculations as hot channel factors.

Basic thermal design criteria established for this design are as follows:

a. No steady state or transient heat transfer surface burnout.

b. No oxide fuel melting at 125 per cent of rated power.

3. Thermal Calculation Methods

In the analyses of the reactor core, calculation methods are established to account for the following effects on the thermal and hydraulic design:

a. Dimensional deviations from the nominal design of the fuel element resulting from manufacturing tolerances.

b. Non-uniform power generation.

c. Deviations from ideal flow conditions.

d. Uncertainties in empirical correlations.

Effects of fuel concentration, pellet eccentricity and growth, rod bowing and flow distribution, on heat flux and maximum temperatures are included in the analysis. A probability of one has been assigned for the combination of these adverse effects occurring simultaneously in a specific channel which is defined as the hot channel.

Initially the clearance between the oxide fuel and tube wall is filled with helium; however, during operation fission gases decrease the thermal conductivity of the gaseous medium. Results of recent tests and reported information are factored into the maximum fuel temperature calculation and a value of 0.01 B/hr-ft-F, is used for the gap thermal conductivity. The thermal conductivity of pure helium is 0.16 B/hr-ft-F.

Variation of the oxide thermal conductivity with temperature is important to fuel element design, however, these data are still inconsistent. A conservative value of 1.0 B/hr-ft-F is used in these analyses. It has been estimated that the oxide thermal conductivity lies in the range of 1.1 to 1.2 B/hr-ft-F.

4. Design Data

The maximum surface heat flux at 585 MW is approximately 464,000 B/hr-ft^2 and the subcooled burnout heat flux calculated by the Jens and Lottes* correlation is 1.6 x 10^6 B/hr-ft^2. This represents a burnout safety factor of 3.45 for rated power operation in the hot channel.

*Jens and Lottes, ANL 4627

At 585 MW the maximum oxide fuel temperature in the hot channel is approximately 750 F below the melting point.

Core thermal analysis cannot be limited to rated power operation but must include effects of heat balance and instrumentation inaccuracies. Depending upon the magnitude of these inaccuracies, the reactor could very well be operated briefly in steady state at a power level of approximately 110 per cent of rated power. For this design, positive scram level is set at approximately 115 per cent of rated power.

Hot channel studies consider reactor operating levels up to 125 per cent of rated power and include reduced flow effects due to local boiling and two phase flow. At this level of operation, the maximum oxide fuel temperature is below the melting point.

The hot channel surface is still below surface burnout for power overshoots (above 125 per cent rated power) resulting from rod withdrawal transients.

Reactor power would have to attain a steady state value of 766 MW (131 per cent of rated power) before surface burnout is reached based on the more conservative two-phase flow burnout correlation. The safety system restricts steady state operation to power levels below 125 per cent of rated power, and even in accidental transients such as accidental rod withdrawal or a cold water accident, the maximum power does not exceed 130 per cent rated power.

5. Burnout Analysis

The core is designed for steady state rated power operation without bulk boiling. At this condition, the maximum surface heat flux is well below the burnout heat flux as predicted by the forced convection correlation of Jens and Lottes.

The possibilities and effects of transient boiling have been investigated. Hot channel pressure drop with bulk boiling (two-phase flow) is greater than the non-boiling pressure drop in that channel for the same mass velocity. For this condition, the margin to burnout is decreased because of the decreased flow and also because burnout heat fluxes are lower for two-phase flow at given mass velocities.

Since the Jens and Lottes burnout correlation is not applicable to bulk boiling, a two phase flow burnout relationship was determined from existing data and correlations. All available burnout data were plotted and a burnout design curve for zero quality and zero subcooling (Figure A-1) was drawn below all measured burnout points. Quality burnout data were also analyzed and correlated as a design curve. (Figure A-2). The mass velocity correlation is the more conservative of the two relationships. These data are results of burnout tests performed at various universities, national laboratories and other AEC sponsored facilities.

Figure A-3 shows the hot channel axial temperature distribution of the bulk coolant (T_W) and clad surface (T_S) for 585 MW operation.

Local boiling is a form of nucleate boiling which occurs when the subcooled liquid (at a temperature below saturation) is brought into contact with the clad surface that is at a temperature ($T_{surface} = T_{sat} + \Delta T_{sat}$) sufficient to cause boiling at the interface. The Jens and Lottes local boiling correlation is used to determine ΔT_{sat}. As shown in Figure A-3 local boiling begins at a distance of 51 inches from the core inlet and continues to a point 81 inches from inlet. Since the coolant is below saturation temperature there is no bulk boiling in the hot channel at 585 MW operation.

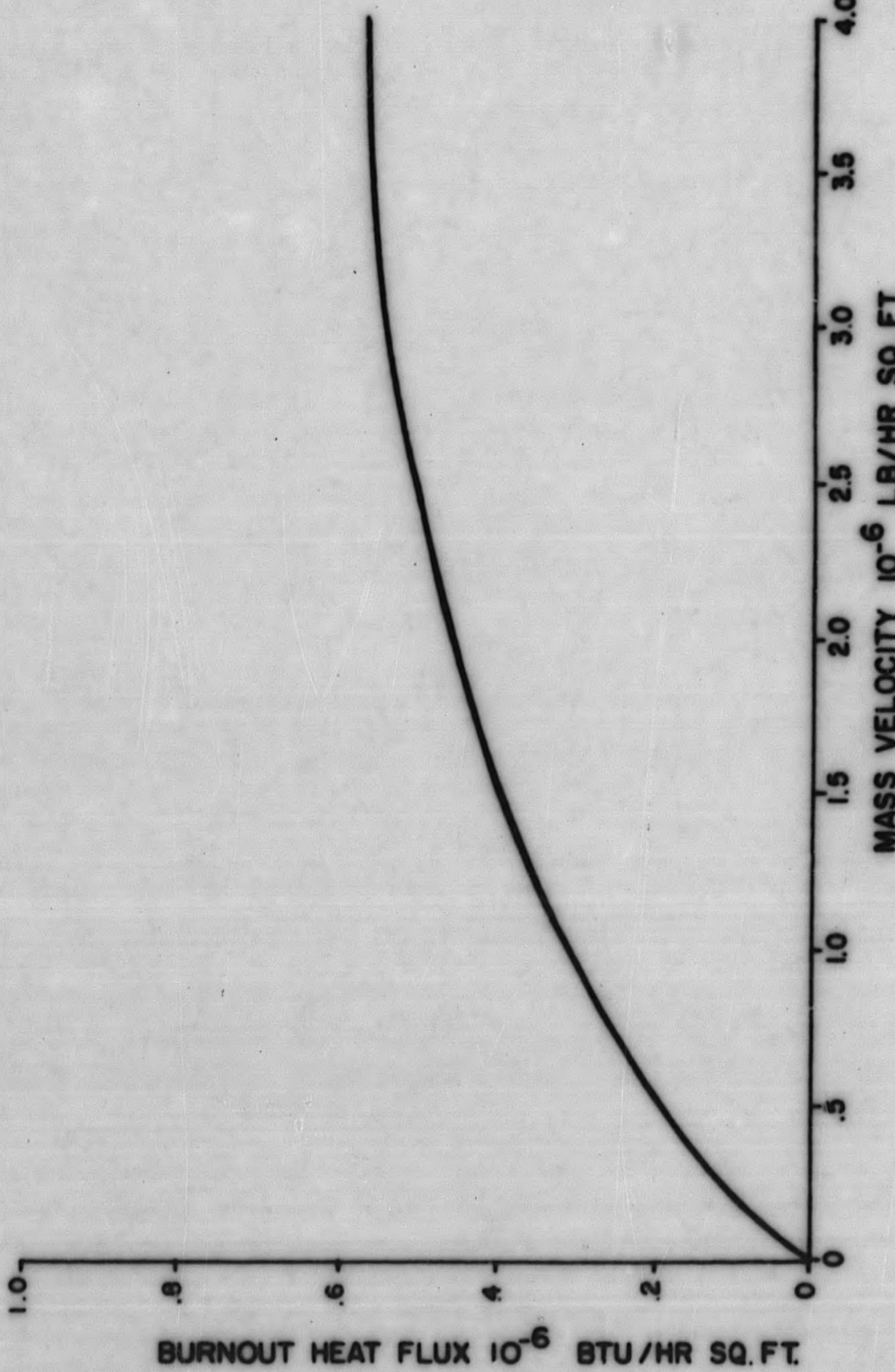
At 122 per cent of rated power (714 MW) and with full flow (4.68×10^7 lb/hr in fuel region) the coolant is still below saturation temperature.

This condition is shown in Figure A-4. An important effect of local boiling is to increase hot channel pressure drop thereby reducing flow in that channel (Figure A-6). The reversal in slope of the curves in this figure is ascribable to density changes in the fluid and contraction and expansion losses in the fuel rod spacers. Since pressure drop is a multi-valued function of flow the lowest value of mass flow rate or mass velocity is used in this analysis. Accounting for this reduction in flow (4.43×10^7 lb/hr or 3.6×10^6 lb/hr-ft^2) results in coolant bulk boiling at a point 95.25 inches from the core inlet.

For 122 per cent of rated power (714 MW) burnout heat flux based on the mass velocity correlation is 5.7×10^5 B/hr-ft^2. At the axial position in the hot channel where bulk boiling begins (zero quality, zero sub-cooling) the maximum surface heat flux is 0.4×10^5 B/hr-ft^2. This results in a margin to bulk boiling burnout of 5.3×10^5 B/hr-ft^2.

Results of a comprehensive analysis establish that the region of possible steady state surface burnout is entered at 131 per cent of rated power (766 MW) based on the mass velocity correlation. With the core at 766 MW and a reduced mass velocity of 2.8×10^6 lb/hr-ft^2 bulk boiling (zero quality) begins at a point 63 inches from the core inlet as shown in Figures A-5 and A-6. At this axial position in the hot channel the surface heat flux is 5.4×10^5 B/hr-ft^2 (Figure A-8) and burnout heat flux is 5.4×10^5 B/hr-ft^2. This is equivalent to a zero margin to burnout or a safety factor of 1.0.

Based on the quality correlation steady state burnout occurs at approximately 142 per cent of rated power.

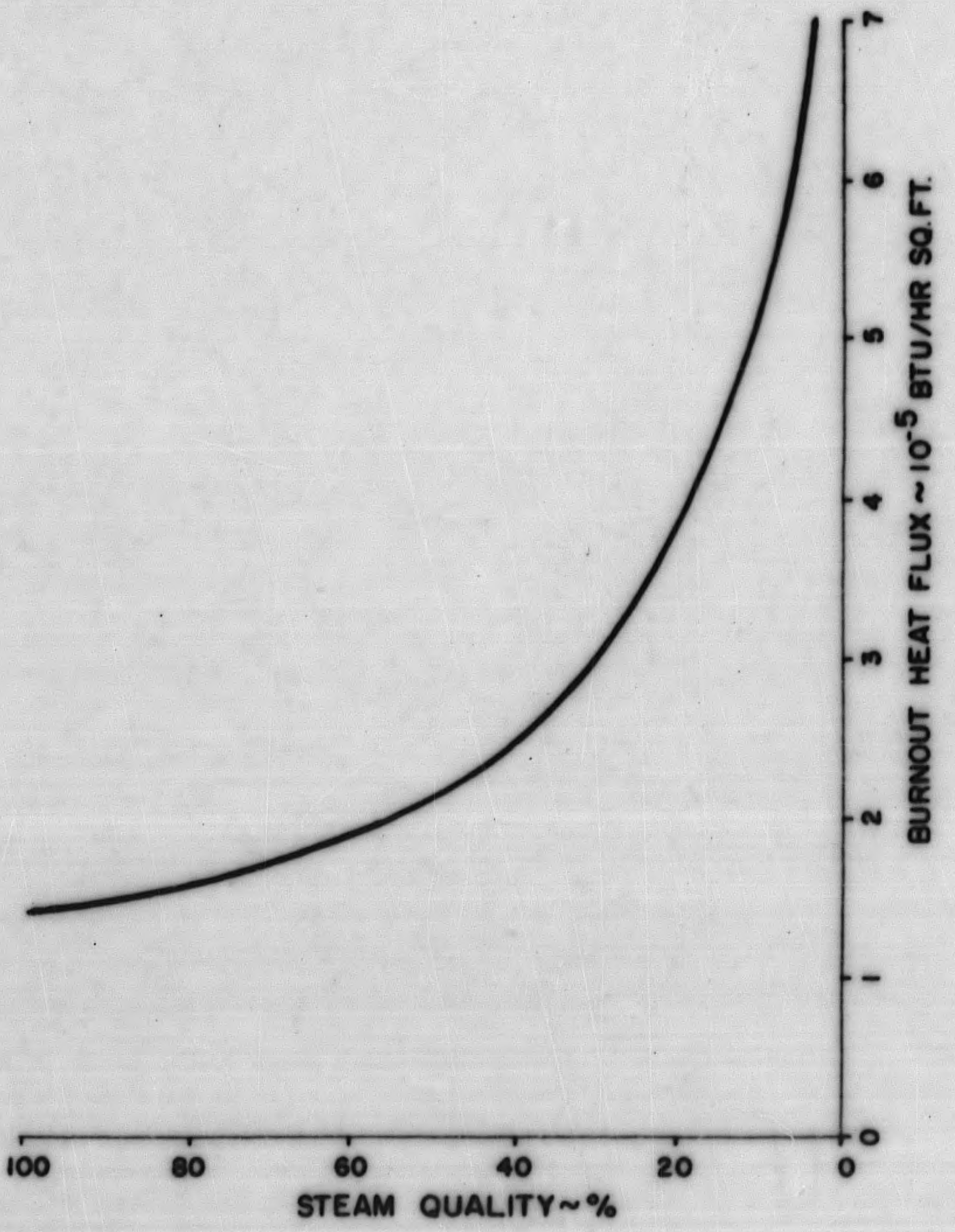


BURNOUT HEAT FLUX

VS

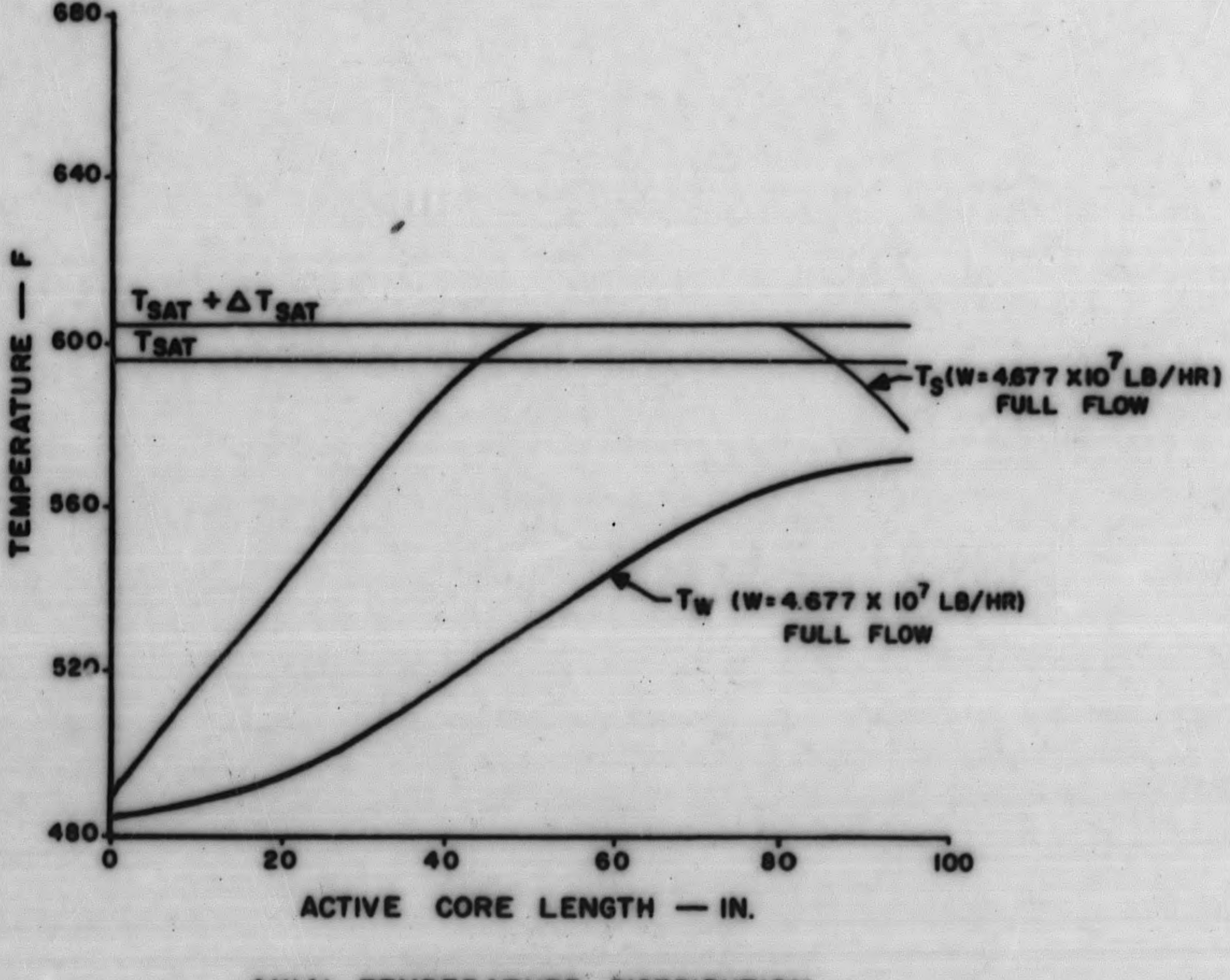
MASS VELOCITY DESIGN CURVE

FIGURE A-1



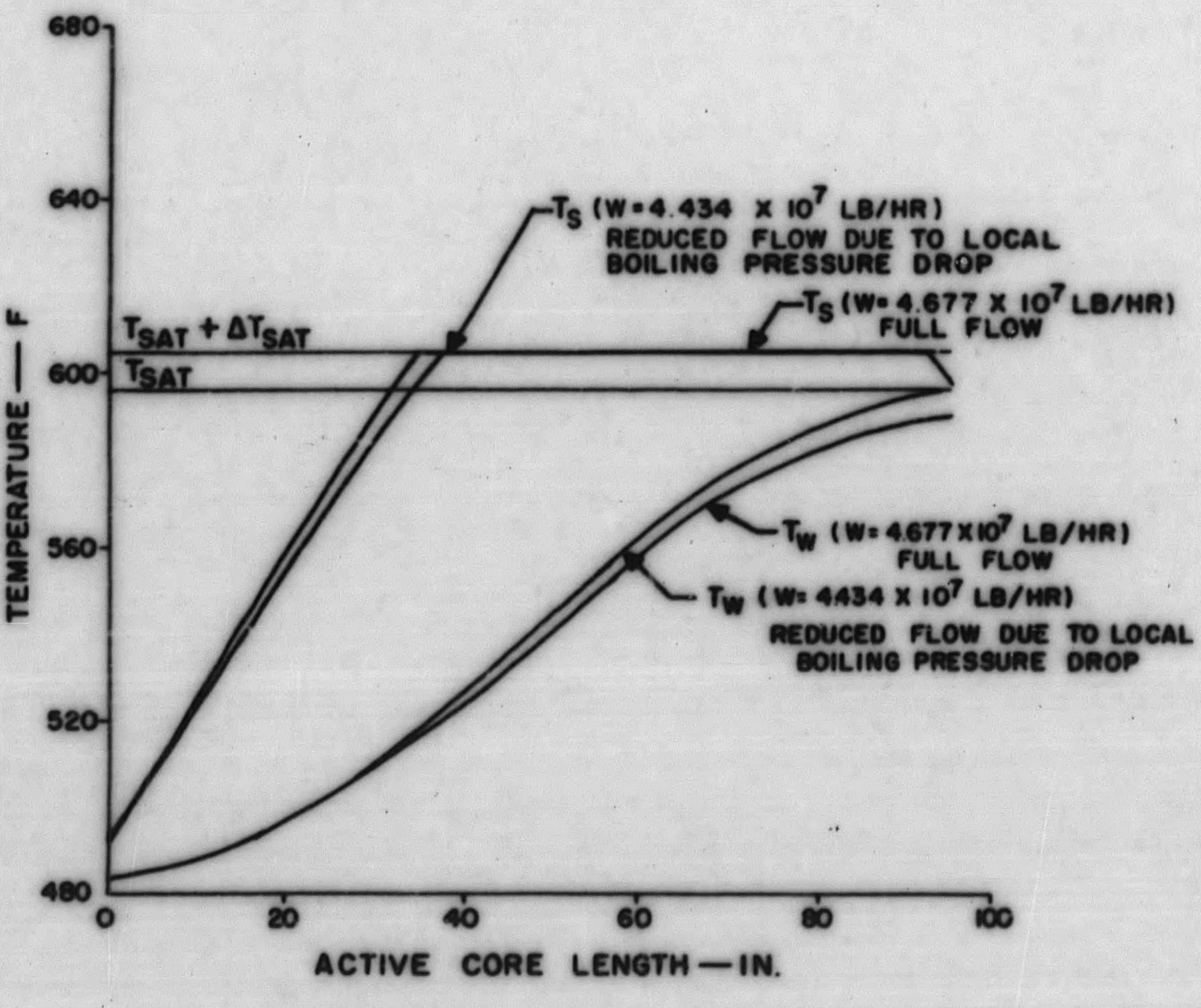
STEAM QUALITY
VS
BURNOUT HEAT FLUX DESIGN CURVE

FIGURE A–2



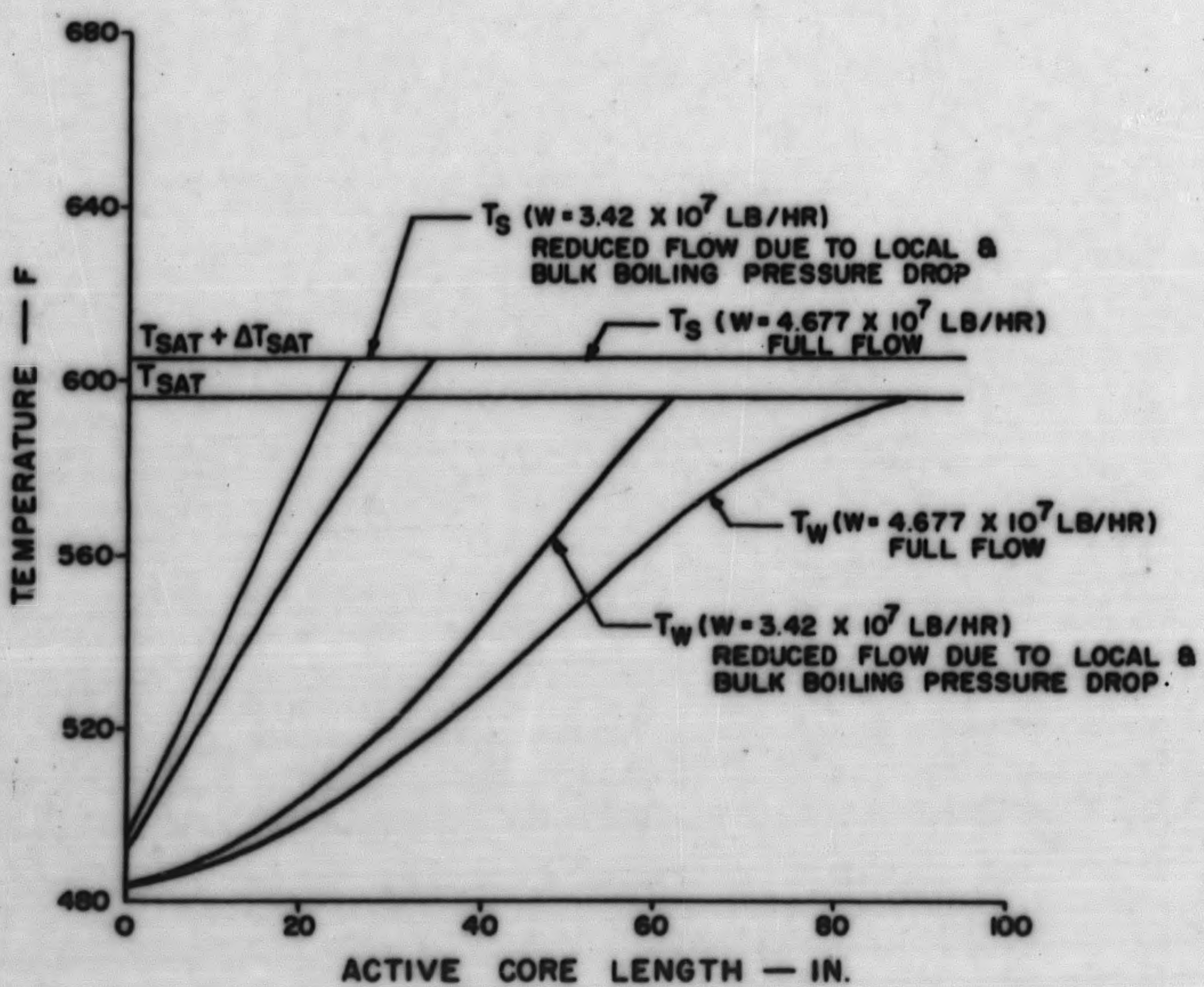
AXIAL TEMPERATURE DISTRIBUTION
IN HOT CHANNEL AT 585 MW
(100% POWER)

FIGURE A—3



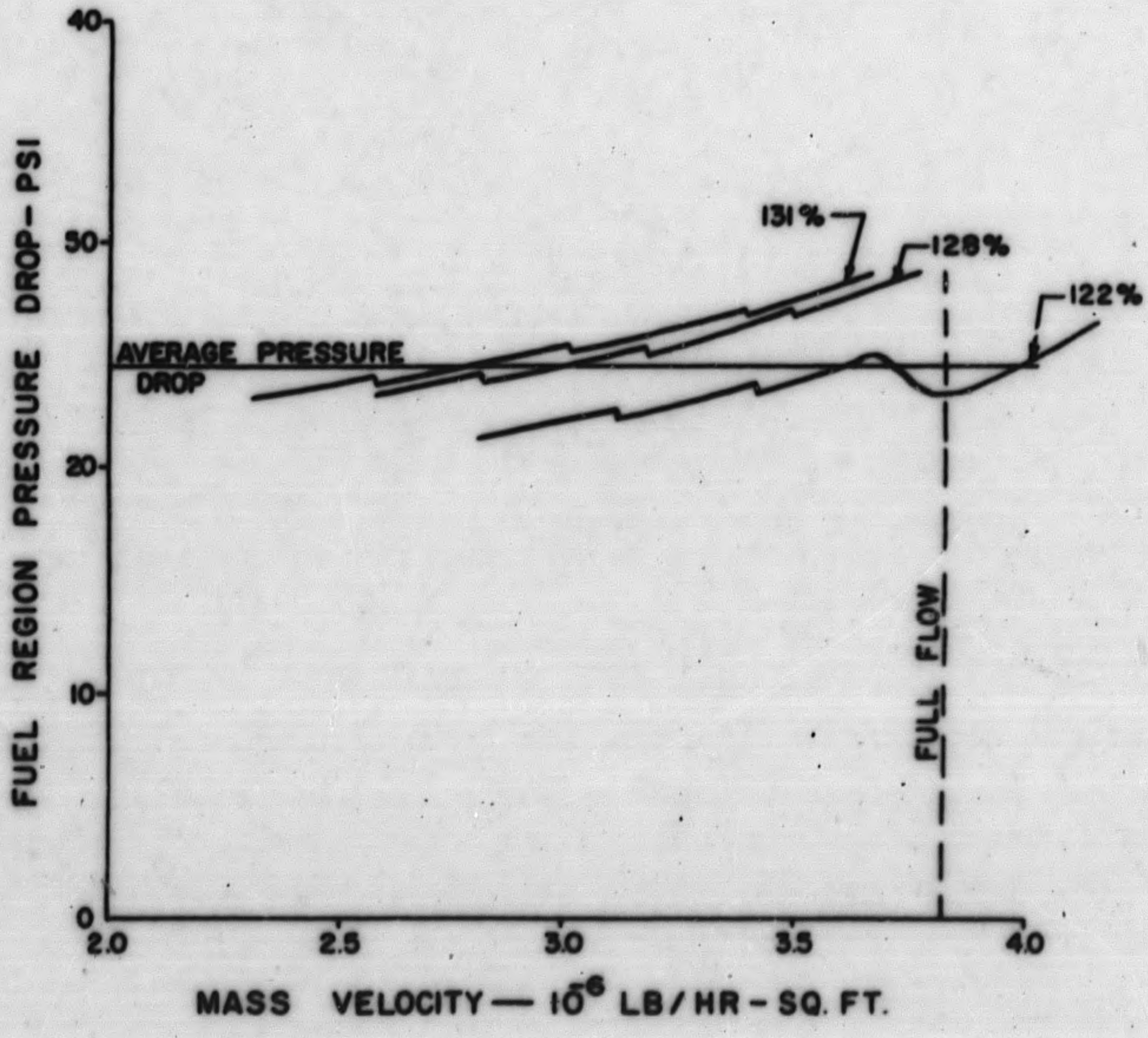
AXIAL TEMPERATURE DISTRIBUTION IN HOT CHANNEL
AT 714 MW (122% POWER)-POWER AT WHICH
NET BOILING BEGINS IN HOT CHANNEL

FIGURE A—4



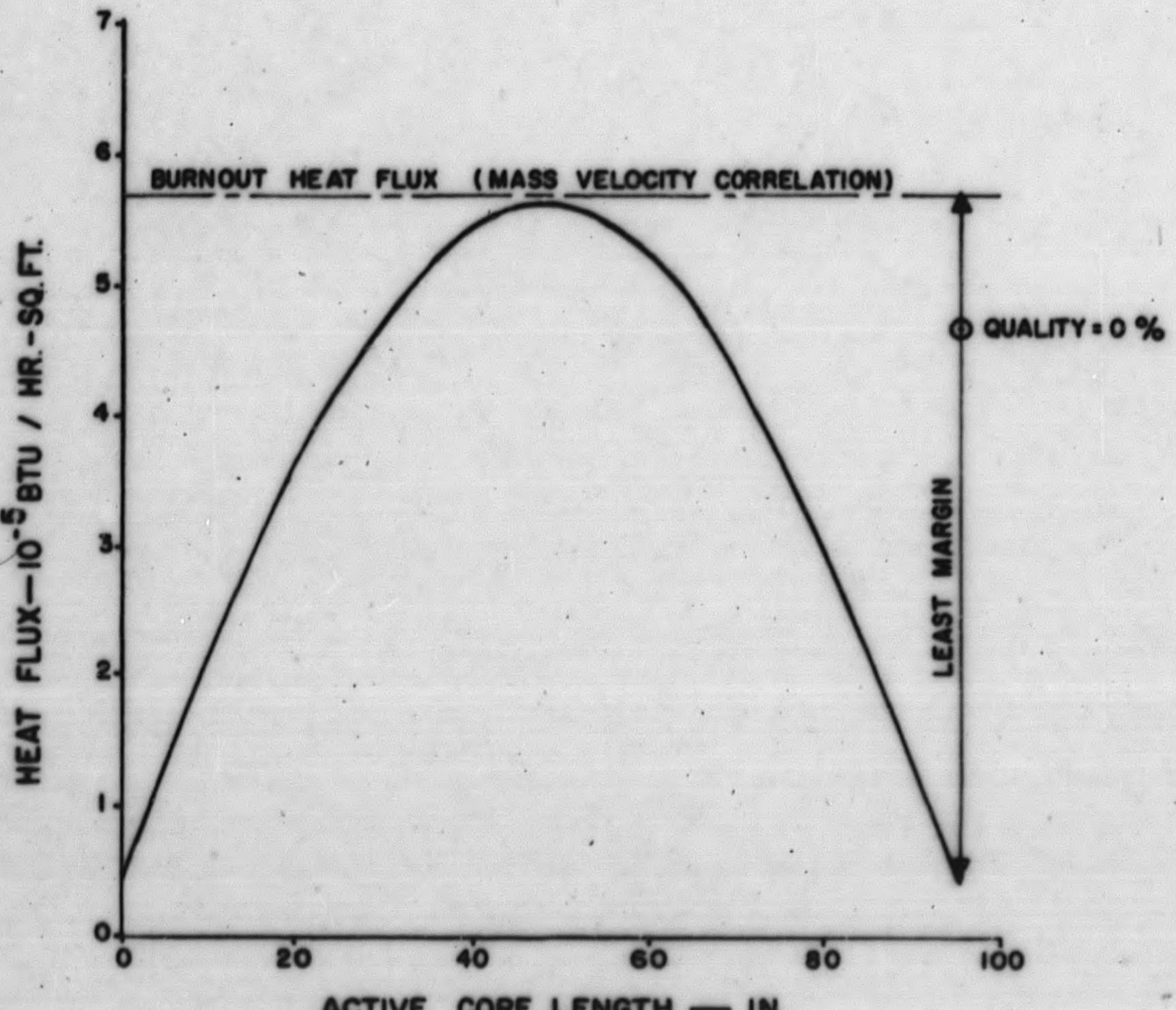
AXIAL TEMPERATURE DISTRIBUTION
IN HOT CHANNEL AT 766 MW
(131 % POWER)

FIGURE A-5



FUEL REGION PRESSURE DROP
VS.
MASS VELOCITY — FOR VARIOUS
POWER LEVELS — HOT CHANNEL

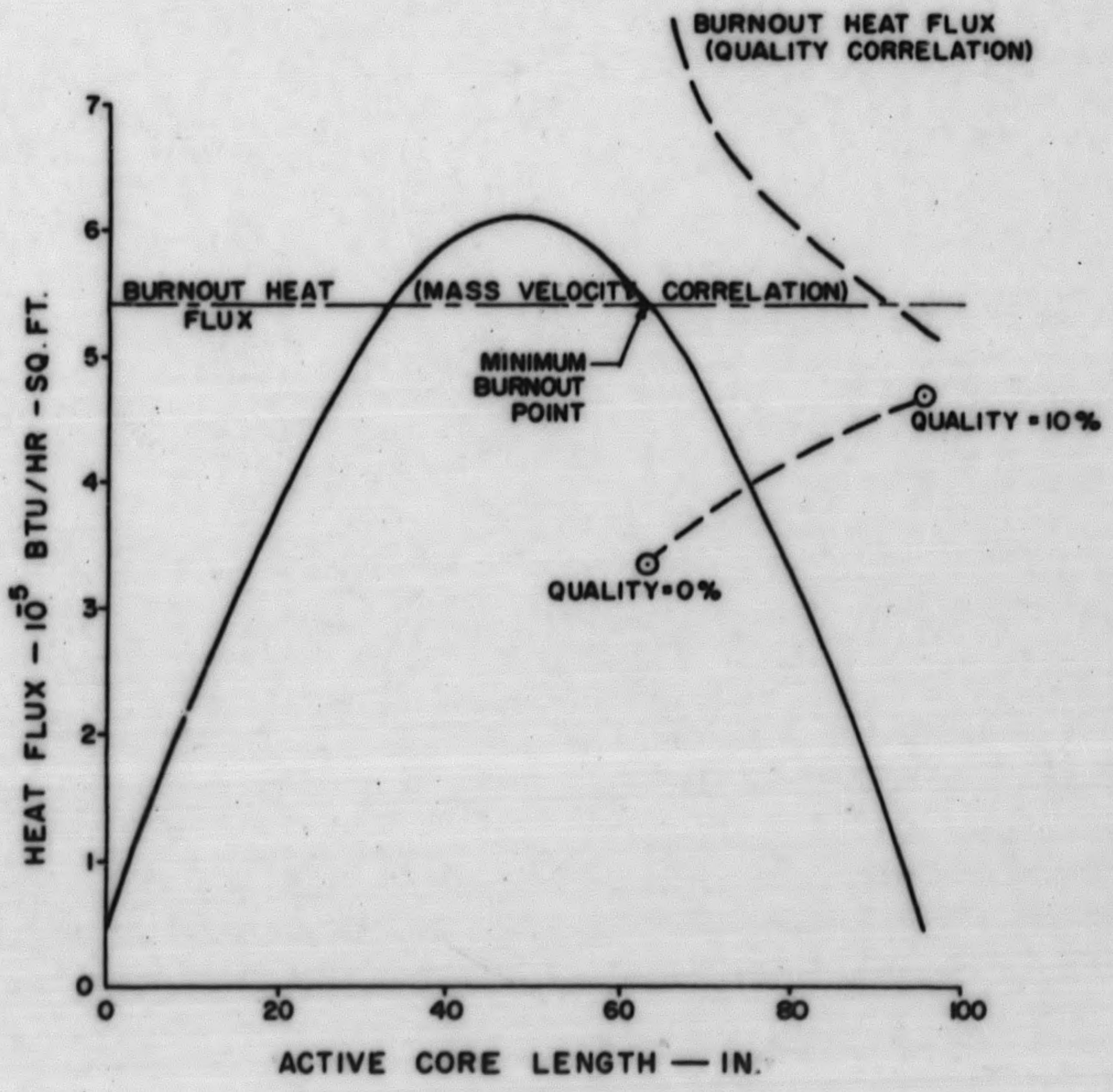
FIGURE A-6



HEAT FLUX & QUALITY DISTRIBUTION
SHOWING BURNOUT CORRELATION
AT 714 MW (122 % POWER) — HOT CHANNEL

FIGURE A-7

EN



HEAT FLUX & QUALITY DISTRIBUTION
SHOWING BURNOUT CORRELATION AT
766 MW (131% POWER)—HOT CHANNEL

FIGURE A-8

END