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U.S. ARMY ENGINEER REACTORS GROUP
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MH-1A CORE 3 THERMAL HYDRAULIC DESIGN ANALYSIS

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A brief description of the MH-1A Floating Nuclear Power Plant is given, followed by the safety system setpoints and operating limits of the reactor. The safety criteria which form the basis for safe operation and protection of the core are presented. A parametric analysis is made of the thermal-hydraulic capabilities of the MH-1A Core 3 during steady-state operation. This includes calculation of peak centerline fuel temperatures at various powers, the maximum linear heat rate, DNBR, and maximum hot channel enthalpy rise. Also included are analyses of transient conditions accompanying uncontrolled rod withdrawal in the power range and during source level startup, secondary system steam line rupture, and loss of reactor coolant flow.

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I. INTRODUCTION

This report presents the analysis of the thermal and hydraulic performance of Core 3 for the MH-1A Floating Barge Nuclear Power Plant. The primary purpose of this study is to assure that the reactor core can meet normal steady-state and transient performance requirements without exceeding acceptable fuel damage limits. The analyses presented herein define the limits of safe operation for the core in terms of measurable reactor parameters: Reactor power, primary system pressure, coolant temperature, and flow rate. During steady-state operation, the reactor, control and protection system provides automatic action, outside the normal operating range, to scram the reactor and to assure that the reactor safety limits are not exceeded. Analysis of postulated reactor transients caused by possible operator error or equipment malfunction assures that the core safety criteria are not violated during transient situations which have a reasonable probability of occurence.

A. Scope

Section II of this report defines the safety system setpoints and the operating limits of the reactor.

Section III describes the safety criteria which pertain to the MH-1A, and on which the safety limits of the reactor core are based.

Section IV describes the analysis of the thermal hydraulic characteristics of the core during steady-state operation. Included in this section are the calculation of peak centerline fuel temperatures at various power levels, and the determination of the maximum linear heat rate. A parametric study of the hot channel enthalpy rise and DNBR (departure from nucleate boiling ratio) is covered, and a determination of the reactor core safety limits for steady-state operation is presented.

Section V covers the analysis of transients initiated from the normal operating limits. Included in this section are analyses of uncontrolled rod withdrawal in the power range, secondary system steam line rupture, rod withdrawal during source level startup, and loss of reactor coolant flow.

Section VI presents the analysis of transients initiated from the design limits and the reactor scram setpoints.

B. Brief Description of the MH-1A

The MH-1A Floating Nuclear Power Plant is designed to serve the United States Army as a mobile source of power at any site accessible by waterways. The MH-1A is to be utilized as a source of electrical power for a variety of missions. It will be towed to a mooring site, secured, and put into service. The reactor will not be operated during towing or associated transitory operations.

The nuclear power plant is installed in a floating mount. This is a Liberty Ship hull, modified by the addition of a new midbody. All of the original propulsion equipment has been removed, and the nuclear reactor and associated equipment have been located in the new midbody. The design of the floating mount is in keeping with the best commercial marine standards consistent with the applicable rules and regulations of the U. S. Coast Guard and the American Bureau of Shipping. For added protection, a collision barrier and a deeper inner bottom are provided.

The electrical generating system is used to generate the electricity required for plant operation as well as the 10MWe for shore distribution. Either 50- or 60-cycle alternating current can be supplied (the 50-cps operation allows a maximum net output of 8.05 MWe).

The heart of the nuclear system is 45 MWt pressurized water reactor. The nuclear core utilizes low-enriched pellets of uranium dioxide in tubular fuel elements, boron steel control rods with borated followers, and a single pass coolant flow.

The MH-1A reactor is cooled by a single reactor coolant loop with two reactor coolant pumps operating in parallel. Coolant enters the reactor vessel through a single inlet nozzle located in a plane above the core, flows down in flow paths between the vessel and core barrel, and then makes a 180 degree turn. The coolant then flows through the lower orifice plate which reduces any maldistribution caused by the inlet conditions. The flow then passes through a lower grid plate which distributes the flow to the fuel elements, control rods, and inner thermal shields. After passing through the core, exit plenum, and hot leg piping, the primary coolant flows through the tubes of a vertical U-tube steam generator where heat is transferred from reactor coolant to the secondary water on the shell side of the steam generator where steam is produced for the secondary plant.

Fuel is contained within free-standing, fully-annealed type 348 stainless steel tubing. Adequate radial pellet clearances are provided to assure pellet assembly and freedom for fuel swelling. A fission gas plenum is provided in each fuel tube for the collection of released fission gases.

Normal operation will allow a power swing from zero electrical power output (plant load only) to full power without moving the control rods. This mode of operation is possible because of the negative doppler and reactor coolant coefficients of reactivity.

The Type I modified control rods were assumed for the analyses in this report. These control rods have no boron in the followers. The result is a lower bank position at peak reactivity and, therefore, higher peaking factors. These control rods then are the most conservative to use in the analysis.

Table I-1 lists the important characteristics of the overall MH-1A power plant. The general thermal and hydraulic performance characteristics of Core 3 are summarized in Table I-2.

TABLE I-1

MH-1A PLANT DESCRIPTION

	(ft)	(in)
1. Hull Characteristics		
Length overall	441	6
Length between perpendiculars	416	0
Beam, molded	65	0
Depth, to upper deck at side	37	4
Draft	17	10
Displacement at 17 ft. 10 in. draft	9400 tons	
2. Electrical Generation Characteristics		
Turbogenerator capacity		
60 cycles, 0.85 power factor, kw	11,500	
50 cycles, 0.58 power factor, kw	9,583	
Voltage, kv	13.8	
Net output to shore		
60 cycles, kw	10,000	
50 cycles, kw	8,051	
Main transformer, kva	15,000	
Taps, kv	13.8/66/44/33/22.9	

TABLE I-1 (cont'd)

3. Reactor Core Characteristics	
Equivalent dia. (in.)	45.2
Active hgt. (in.)	36.0
Fuel element data	
Fuel elements (No.)	32
Fuel pins/fuel element (No.)	104
Fuel pin pitch (in.)	0.654
Fuel pin OD (in.)	0.507
Fuel clad material	SS 348
Fuel clad thickness (in.)	0.023
UO_2 pellet OD (in.)	0.4565
UO_2 density (gm/cm^3)	10.35
Radial helium gap (cold), in.	0.00225
Control Rod Data	
Control Rods (No.)	12
Shape	Cruciform
Blade width (in.)	
Inner 4 rods	10.78
Outer 8 rods	10.20
Absorber matrix composition	Boron stainless steel
Absorber thickness (in.)	0.250
Clad material	SS 348
Clad thickness (in.)	0.050
Absorber length (in.)	36.75
4. Secondary System Characteristics	
Feedwater temperature, °F	345
Feedwater flow rate, lb/hr	170,960
Continuous blowdown rate, lb/hr	1,700
Steam temperature, °F	430
Steam pressure, psia	
Full power	342
Low power	815
Steam flow rate - full power, lb/hr	169,260
Steam quality to turbine, %	99.75
Steam safety valve setting, psig	885
5. Containment Vessel Design Characteristics	
Maximum internal pressure, psia	155
Maximum internal temperature, °F	350
Maximum external pressure, ft of water	150

TABLE I-2

GENERAL THERMAL AND HYDRAULIC CHARACTERISTICS
OF MH-1A, CORE 3

Rated power level, MW	45
Btu/hr	1.53×10^8
Heat generated in the fuel, %	96.6
Pressure	
Nominal, psia	1400
Minimum, steady-state	1352
Coolant flow (nominal)	
Total flow, lb/hr	4.40×10^6
Coolant flow to inner thermal shields,%	2.3
Total coolant leakage flow, %	8.8
Effective flow rate for heat transfer lb/hr	4.01×10^6
Core flow area, ft^2	5.22
Average coolant mass velocity, lb/hr ft^2	$.769 \times 10^6$
Primary loop pressure drop, psi	32.6
Coolant temperature	
Nominal inlet temperature, °F	476
Maximum inlet, steady-state, °F	483
Average rise in vessel, °F	30.5
Nominal outlet of vessel, °F	506
Nominal core bulk outlet temperature,°F	509
Heat transfer (at rated power)	
Active heat transfer area, ft^2	1323
Core average heat flux, Btu/hr ft^2	0.112×10^6
Average linear heat rate of rod, kw/ft	4.35
Maximum linear heat rate, kw/ft	18.6
Maximum fuel temperature, °F	4106
Average core enthalpy rise at rated power Btu/lb	38.3
Hot channel (normal operating limits)	
Reactor power, MW	48.2
Maximum heat flux, Btu/hr ft^2	0.514×10^6
Peak linear heat rate, kw/ft	19.9
Maximum UO_2 temperature, steady-state,°F	4402
Maximum clad surface temperature, °F	612
Hot channel outlet temperature, °F	562
Hot channel outlet enthalpy, Btu/lb	564.2

TABLE I-2 (cont'd)

Reactor outlet temperature, °F	517
Pressure, psia	1352
Inlet temperature, °F	483
DNB ratio (W-3 correlation) steady-state	1.77
Maximum quality, %	-4.8

II. DETERMINATION OF SAFETY SYSTEM SETPOINTS

This section discusses the selection of the safety system setpoints and the margins which these setpoints assure with respect to the safety limits. Table II-1 summarizes the protective system settings for Core 3, including the alarm setpoints and reactor trip setpoints and the effects of associated instrument error. Subsequent paragraphs discuss safety limits and summarize the conclusions of the steady-state and transient analyses.

A. Basis

The basis for the determination of safety system setpoints is that given in AEC General Design Criteria 6:

> "The reactor core with its related controls and protection systems shall be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits which have been stipulated and justified. The core and related auxiliary system designs shall provide this integrity under all expected conditions of normal operation with appropriate margins for uncertainties and for specified transient situations which can be anticipated".

The concept of a safety limit is further described in the Guide On Content of Technical Specifications for Nuclear Reactors, also an AEC document.

> "The safety limit is a value of the chosen variable at which one can say with confidence that no serious consequences will occur. If the value of the variable were to be at this limit and all other variables at the upper bound of their operating range, and if all uncertainties in technical knowledge of the process were resolved unfavorably, no hazard to the public would exist".

The limiting safety system settings are established so as to assure that no core damage will occur during all normal operation and expected transient conditions by preventing reactor parameters from approaching or exceeding the safety limits. The basis for determination of reactor core safety limits is the protection of the fuel element cladding - the first barrier to the unsafe release of fission products. The integrity of the fuel cladding is assured by (1) preventing fuel centerline temperatures from reaching the melting point under all operating conditions, and (2) preventing departure from nucleate boiling (DNB) or undamped oscillations in the core which can lead to DNB. A limitation of + 2 percent quality has been established

TABLE II-1

MH-1A CORE 3 SUMMARY OF PROTECTIVE SYSTEM SETTINGS

	Power (%)		Flow (gpm/psi)		High Pressure (psia)		Low Pressure (psia)		Outlet Temperature (°F)	
	Real	Meas	Real	Meas	Real	Meas	Real	Meas	Real	Meas
LSSS (Scram or Safety valve)	115	$\leq 110^{(1)}$	$\frac{9250}{25.6}$	$\leq 27.0^{(1)}$	1632 Safety	$\leq 1600^{(1)}$ Valve	1265	$\geq 1287^{(1)}$	534	$\leq 530^{(1)}$
Adverse Limit Operating Band	111	$\leq 106^{(2)}$	$\frac{9750}{28.6}$	$\leq 30.0^{(2)}$	1477	1450	1352	$\geq 1375^{(2)}$	534 516	$\leq 530^{(2)}$ 513 Average
Nominal	105	100	$\frac{10,200}{31.2}$	$\frac{10,800}{32.6}$	1423	1400	1387	1400	510 495	506 491 Average

(1) Limiting Safety System Setpoint
(2) Limiting Alarm Setpoint; note that alarm on temperature is not required since reactor trip occurs at 534°F.

for steady-state operation. No undamped oscillations have been observed at qualitites $\leq$ 2 percent. The occurence of DNB must be prevented because of the resultant large decrease in heat transfer capability at the cladding surface, and the possibility of cladding failure due to high temperature.

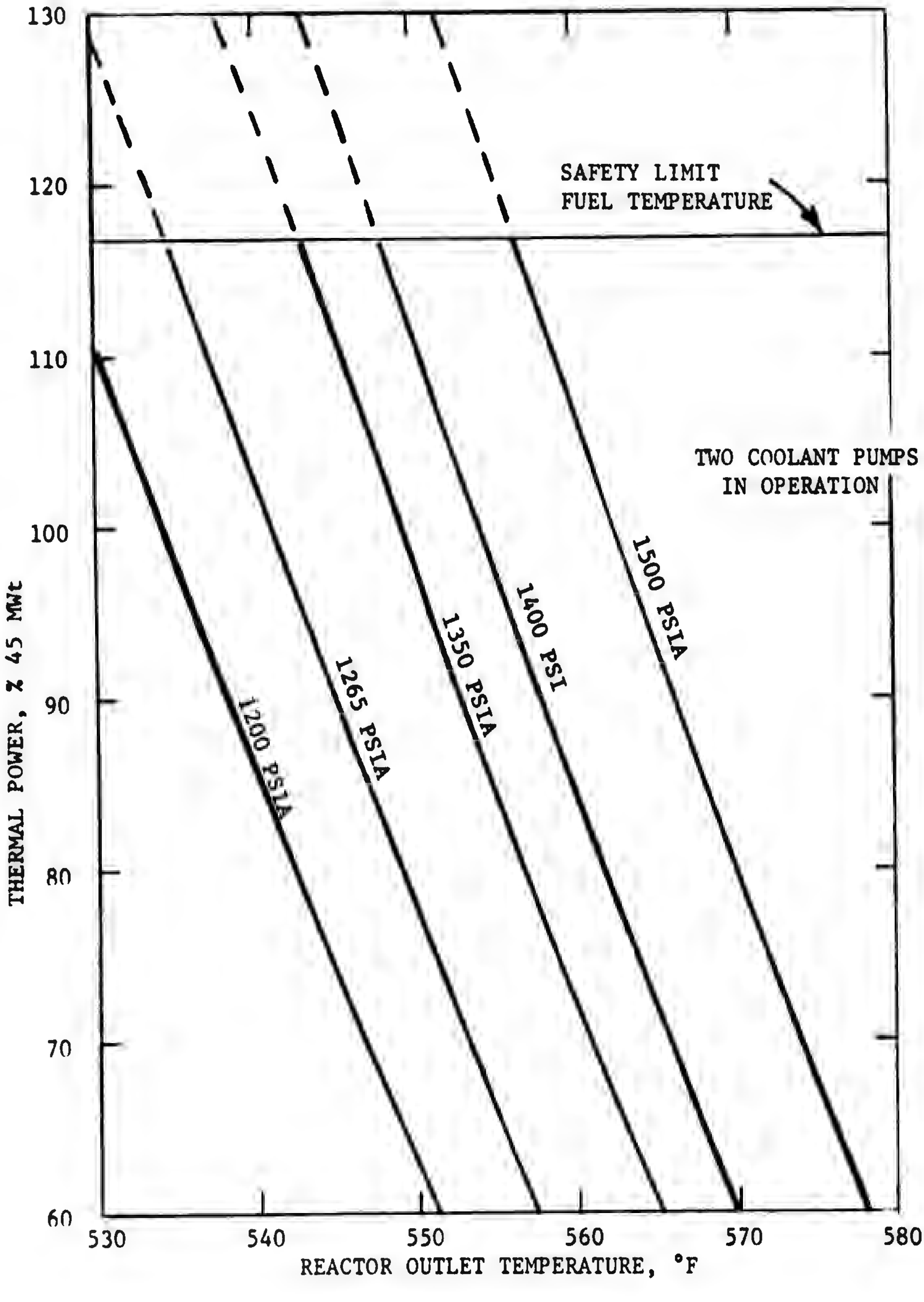
Safety limits are developed, based on the thermal hydraulic analysis, for system parameters used in monitoring the behavior of the reactor plant. Figure II-1 shows the reactor core safety limits for reactor power, coolant temperature, and pressure with two primary coolant pumps in operation. The safety limit is exceeded if the point defined by the combination of reactor outlet temperature and power level is at any time above the appropriate pressure line. In addition, a safety limit on reactor power is established to specifically prevent fuel centerline melting due to excessive heat generation in the fuel. The pressure curves shown in Figure II-1 represent the locus of points of reactor power and coolant temperature for which the hot channel exit quality is equal to 2 percent. The actual departure from nucleate boiling safety limit is reached subsequent to these limits.

The safety limits represented by these curves should be interpreted properly as the maximum allowable instantaneous rate of heat transfer from the fuel to the coolant (thermal power). Note that the rate of heat generation (neutron flux level) is equal to the rate of heat transfer to the water only if the neutron flux is not varying with time. In the case of power excursion, the rate of heat transfer to the coolant can exceed the safety limit only if the nuclear flux level remains in excess of it for a sufficient length of time. Analysis of specific transient conditions determins whether an actual safety limit has been violated.

It should be noted that these safety limit curves are not a basis for normal operation. Normal plant operations are well within the limits specified by these curves. For adverse operating conditions, the reactor protection system is designed to actuate a reactor trip before a safety limit is reached to insure a hot channel exit quality of less than 2 percent, departure from nucleate boiling ratio of greater than 1.30, and a fuel centerline temperature of less than melting. The following section discusses the normal operating range of each of the reactor parameters.

B. Normal Operating Limits

The normal operating band is defined by limits above and below the nominal full power value of each prime system variable. The prime system variables for the MH-1A are: Reactor power, coolant temperature, coolant flow, and coolant pressure. The total margin between the operating limit and the nominal value is based on observed deviations during normal plant transients (i.e., load changes), permitted deviations during steady-state operation, and instrument error band.



MH-1A CORE 3 SAFETY LIMITS FOR DESIGN FLOW
FIGURE II-1

1. Thermal Power Level

The nominal thermal power level corresponding to 100 percent power is 45 MWt. Operating data shows that the indicated power level is generally 99 percent, but rarely as much as 101 percent during "full power" operation. By defining the indicated normal full operating band as between 95 percent and 102 percent, sufficient margin is available to account for normal fluctuations in load demand. In the event of a partial or complete loss of load, the power decreases and leaves this operating range. However, this is in the safe direction as far as steady-state operation is concerned, and transients (such as rod withdrawal) are analyzed from various initial power levels to account for transients during startup or less than full power operation. The instrument error associated with power readings is $\pm$ 5 percent; therefore, 107 percent is considered the maximum power obtained during steady-state operation.

2. Primary System Pressure

The normal primary system pressure of the MH-1A is 1400 psia (1385 psig). The pressure is controlled automatically by pressurizer spray and three banks of heaters. One bank of heaters is in continuous operation to offset heat losses. Bank number 2 is energized at 1360 psig and de-energized at 1385 psig. Bank number 3 is energized at 1335 psig and de-energized at 1385 psig. The capacity of banks 1, 2, and 3 are 15, 75, and 150 kw respectively. The pressurizer spray valves open at 1435 psig. Allowing 5 psi for overshoot, and considering that during normal operation heater bank number 3 is not activated, the operating band on indicated pressure is defined as between 1360 psig and 1440 psig; the instrument error is $\pm$ 22.4 psi.

3. Core Inlet Temperature

The nominal full power inlet temperature is 476°F. Allowing a $\pm$ 2°F control band, a 5 percent uncertainty in power, and an instrument error of 3.6°F, results in an operating band between 468°F and 483°F.

4. Primary Coolant Flow

The nominal coolant flow (10,800 gallons per minute) is based on: (1) Byron-Jackson test data (ref. 4) on the as-fabricated pumps, (2) extrapolation by Byron-Jackson of this data to 470°F, and (3) plant differential pressure measurements. The accuracy of (1) and (2) combined is assumed to be $\pm$ 5 percent; the accuracy of the differential pressure measurements is $\pm$ 1.4 psi. This places an uncertainty on nominal flow of between 10,350 gpm and 11,200 gpm. Small changes in temperature do not affect flow significantly.

A variation in flow, (due to roll and pitch, a normal plant transient) or due to flow decreases with lifetime of ± 150 gpm is assumed, resulting in operating band limits of 10,200 gpm and 11,350 gpm. Since flow is not controlled by the control system, no instrument error is added when determining the normal operating limits; but since the principal flow scram is on pump differential pressure, the instrument error (± 1.4 psi) is considered in determining the scram point.

Table II-2 shows the normal operating limits for MH-1A including the effects of instrument error.

TABLE II-2

MH-1A NORMAL OPERATING BAND LIMITS

	Nominal	Operating Band Limit ± Instrument Error
Reactor power, % 45 MW_t	100	107
Pressure, psia	1400	1352
Inlet temperature, °F	476	483
Outlet temperature, °F	506	517
Flow rate, gpm	10,800	10,200

C. Design Band Limits

The operating limits defined for reactor power, coolant temperature, and pressure are sufficient for normal steady-state operation. However, under certain operating conditions, it is possible that one or more of the reactor parameters may be temporarily outside of the normal operating band. In the event that such a condition develops in an adverse direction, alarm setpoints are established for the purpose of alerting the operator to take corrective action to return the plant to the normal operating range and prevent a reactor trip. Sufficient margin is provided between the nominal conditions and the alarm setpoints to prevent unnecessary alarms during plant operations. The alarm setpoints are also referred to as adverse limits of the operating band and are used as design conditions in the thermal hydraulic analysis.

Table II-1 shows the alarm setpoints for each parameter and the effect of instrument error on each. It should be noted that the reactor coolant outlet temperature alarm setpoint may be set at any appropriate value up to the reactor trip setpoint in order to prevent unnecessary alarms due to the rise in average temperature following a turbine trip. The alarm setpoint on reactor coolant flow is based upon pump head and primary loop characteristics. The measured pump differential pressure is 32.6 psi and the alarm setpoint is specified as 30.0 psi. The primary coolant flow corresponding to the alarm setpoint, plus instrument error, is conservatively used as the design flow in the thermal hydraulic analysis. Since the output of the coolant pumps is not controllable and the loss of one pump results in a loss of flow trip, the actual steady-state flow will be greater than the assumed design flow unless a loss of flow transient occurs as analyzed in this report. The alarm setpoints, plus instrument error on other parameters, are conservatively assumed as design values in the steady-state analysis and as initial conditions for the analysis of transients.

D. Safety System Setpoints

Based upon the results of the steady-state thermal hydraulic and the transient analyses, a set of safety system setpoints was selected. The limiting value of the setpoint and the effects of instrument errors is shown in Table II-1. The setpoints were selected to (1) assure that none of the safety limits established in the steady-state thermal hydraulic analysis are violated during plant operations, (2) assure that safety limits are not violated during postulated reactor transients, and (3) to give the reactor plant operational flexibility by providing adequate margin between normal operating conditions and the safety system setpoints.

E. Margins to Safety Limits

The following paragraphs discuss the reactor trip setpoints and the margins that these setpoints insure with respect to the safety limits.

1. Steady-State Summary

The thermal hydraulic analysis indicates that core exit quality and fuel melting considerations are more limiting than actual DNB caused by high heat flux. The maximum overpower capability with respect to fuel centerline melting is 117 percent of rated power. The safety limit on fuel melting is not significantly affected by other operating parameters. The limits stated below for reactor parameters are the values which would result in reaching a safety limit on fuel temperature, DNB, or exit quality, assuming that the remaining parameters are at the adverse limit of the operating band including instrument error.

(a) Reactor power

The limiting reactor power levels with respect to fuel centerline melting (4,800°F), and to a DNB ratio of 1.3 are 117 percent and 132 percent respectively. The high reactor power setpoint of 110 percent provides a steady-state margin of 7 percent reactor power before the most limiting safety limit is reached. Five percent of this margin accomodates instrument and calibration error, so that the minimum safety margin is 2 percent reactor power.

(b) Reactor Coolant Outlet Temperature

The limiting reactor coolant outlet temperature with respect to steady-state core exit quality is 545°F, and the limiting temperature with respect to DNB is greater than 560°F. The maximum fuel centerline temperature with a coolant outlet temperature of 545°F is less than 4570°F. Thus, a high coolant outlet temperature setpoint of 530°F provides a steady-state margin of 15°F before the hot channel exit quality exceeds 2 percent; 4°f of this margin accomodates instrument and calibration error, so that the minimum safety margin is 11°F.

(c) Primary System Pressure

The limiting primary system pressure with respect to exit quality is 1240 psia. The limit with respect to DNB is less than 1000 psia. The maximum fuel temperature at a pressure of 1240 psia is 4540°F. The setpoint on primary system low pressure of 1287 psia provides a margin of approximately 47 psia; 22.5 psi of this margin accomodates instrument and calibration error.

(d) Primary Coolant Flow

The limiting primary coolant flow rate with respect to the exit quality is 8350 gpm. From a DNB ratio standpoint, the limit is less than 7000 gpm. The maximum fuel temperature is not significantly affected by changes in flow rate.

Low Reactor Coolant Pump Δp. The primary system flow rate at 490°F is 10,800 gpm based on the measured Δp of 32.6 psi and the pump characteristic curve. Including an instrument error of 1.4 psi in the Δp measurement and assuming the loop characteristic curve varies as $(w/w_0)^2$, the scram setting of 27.0 psi corresponds to a flow rate of 9500 gpm. The margin of 6.0 psi to the safety limit flow rate of 8350 gpm or 21 psi is sufficient to accomodate the 1.4 psi instrument error and a 4.0 psi error for transmitter roll (15 degrees) and pitch (5 degrees).

(e) High Primary System Pressure

The limit on high primary system pressure of 1760 psia is a mechanical limit established to insure the integrity of the primary coolant system. The basis for this limit is 110 percent design pressure for the reactor pressure vessel under the ASME Code, Section III (Reference 1). The analysis shows that the fuel centerline temperature is relatively insensitive to an increase in system pressure, and the limiting pressure in this regard is well beyond the safety limit of 1760 psi. Margins to other safety criteria are not affected adversely by high system pressure.

2. Transient Summary

A summary of the analysis of reactor transients and the protection provided by the safety system settings is presented in Table II-3. The limits stated are for the most severe transient in each case initiated from the adverse limit of the operating band including instrument error.

It should be noted that during transient conditions, safety limits are given in the form of curves rather than by single numerical values. This approach is based on the fact that certain limits are related to interdependent variables such as (1) reactor coolant outlet temperature, (2) heat flux (reactor power), and (3) reactor coolant pressure. The use of safety limit curves aids in the determination of whether a particular limit has been approached or exceeded.

(a) Power Range Rod Withdrawal Transient

Uncontrolled rod withdrawal in the power range was analyzed for conditions of both fast reactivity insertion and slow reactivity insertion. The high outlet temperature trip protects the plant against slow reactivity insertion rates that would cause system pressure and temperatures to increase significantly before the plant is scrammed on high power. The nuclear trip channels, on the other hand, will respond first for rapid reactivity insertion rates before any significant change in system process variables is detected.

The analysis shows that all transient conditions which could occur as a result of uncontrolled rod withdrawal are constrained by a combination of high power and high outlet temperature reactor trips. The nuclear high power setting provides a boundary that assures that the reactor will be scrammed at or before a power level of 115 percent is reached. The most limiting transient caused by a fast reactivity insertion rate results in a maximum thermal flux of 112 percent and a maximum outlet temperature of 535°F, which do not exceed the allowable safety limit boundaries. The corresponding maximum fuel temperature and minimum DNB ratio are 4564°F and 1.55 respectively. The hot channel exit quality is less than 0 percent at all times during the transient.

A transient caused by a slow reactivity insertion, resulting in a significant increase in primary system pressure and temperature, is terminated by high outlet temperature scram. The most severe transient of this nature, in which the power levels off just below the scram setpoint and the temperature increases at the maximum rate, results in a temperature overshoot of 14°F. The primary system pressure reaches 1395 at the time of scram and continues to 1430 psi where it is limited by the pressurizer spray system. The high outlet temperature protection prevents the reactor parameters from exceeding the safety limit boundary defined by hot channel exit quality. The minimum DNB ratio occuring during this transient is 1.45 and the maximum hot channel exit quality is +2.0 percent. The maximum fuel centerline temperature is 4734°F corresponding to a steady-state power level of 115 percent.

The remaining safety system setpoints on low primary system pressure and low coolant flow rate are not significantly influenced by the power range rod withdrawal transient.

(b) Main Steam Line Rupture Transient

The effect of rapid cooldown of the primary system resulting from a main steam line rupture was analyzed for the most severe conditions with respect to reactivity addition and initial steady-state conditions. The analysis shows that the high power reactor trip is sufficient to prevent violation of the core safety limit boundaries. The maximum fuel temperature and minimum DNB ratio resulting from the main steam line rupture transient are 4564 °F and 1.54 respectively. Neither the reactor coolant temperature nor coolant flow rate safety setpoints are influenced by this transient. A low pressure setpoint of 1265 psia was used throughout the analysis of this transient.

(c) Loss of Coolant Flow Transient

Loss of reactor coolant flow is detected by direct measurement of the flow in the coolant loop and by loss of pump power. The flow rate is measured by differential pressure tap across the pumps. The analysis shows that the core is protected with a setpoint of 24 psi (8650 gpm) and a delay time between pump failure and initial rod insertion of 0.500 seconds. Loss of flow from either of the two pumps initiates a low flow reactor trip. Coolant flow, after a reactor trip, is maintained momentarily by the flow coastdown due to the inertia of the pumps. Analysis of the loss of flow transient initiated from the most adverse limit of the operating band plus instrument error shows that the minimum DNB ratio is greater than 1.40 and that there are no sustained flow oscillations during the transient. Since any possible increase in measured coolant outlet temperature occurs after the low flow scram, and since there is no reactor power increase in the transient, the setpoints of these variables are not influenced. While the primary system pressure might increase during the transient, the pressure is conservatively taken at its minimum operating value of 1352 psia.

(d) Source Range Rod Withdrawal Transient

Protection against rapid approach to criticality caused by continuous control rod withdrawal from a subcritical condition is provided by the high power reactor trip. Although very conservative assumptions were used, the analysis indicates that the reactor core is fully protected from the source range rod withdrawal transient. Because of the rapidity of the transient and the long fuel heat transfer time constant, fuel temperature is the most limiting safety criterion. There is a large margin to DNB during the transient, since the rod surface heat flux remains low. The maximum fuel temperature during the transient is less than 960°F. Since the transient is initially terminated by Doppler effect, the fuel temperature is relatively insensitive to reactor power setpoint. The other reactor system setpoints are not influenced by this transient.

III. SAFETY CRITERIA

The thermal design of the MH-1A is based upon protection of the core from any transients or operating conditions which might result in loss of the first line of containment - the fuel pin cladding. In order to insure that the clad integrity is not violated, certain design criteria are defined. These constitute limits on minimum DNB ratio, maximum fuel centerline temperature, and a hydraulic stability criteria.

A. DNBR

While the occurence of DNB does not necessarily lead to clad failure, protection against DNB combined with protection against high clad temperature does insure against clad failure.

In order to insure that sufficient margin to DNB always exists during steady-state and transient operation, a minimum DNBR of 1.30 was established. DNB ratio is defined as:

$$DNBR = \frac{q''DNB}{q''act}$$

where q"DNB is the heat flux which results in departure from nucleate boiling for given flow rate and core inlet temperature conditions, and where q"act is the actual heat flux for the same flow rate and inlet temperature conditions.

The empirical correlation used to calculate DNB heat flux was Westinghouse W-3 correlation developed by Tong (Ref. 3). The correlation is:

$$\underline{q''DNB,EU} = [(2.02 - 0.004302p) + (0.1722 - 0.0000984p)$$

$$\exp(18.177 - 0.004129p)X] \times [(0.1484 - 1.596X$$

$$+ 0.1729X|X|)G/10^6 + 1.037] \times (1.157 - 0.869X)$$

$$[0.2664 + 0.8357 \exp(-3.151D_e)] \times [0.8258 +$$

$$0.000794 (H_{sat} - H_{in})].$$

where:

D_e = hydraulic diameter

p = pressure

X = quality

G = mass flow rate

H_{sat} = saturation enthalpy

H_{in} = inlet enthalpy

The heat flux is given in Btu/hr-ft^2 and the units and the ranges of parameters of the data used in developing this correlation are:

P = 1000 - 2300 psia

G = .37 x 10^6 - 5.0 x 10^6 lb/hr-ft^2

D_e = 0.2 - 0.7 in.

X_{loc} = 0.15 - +0.15

H_{in} = $\geq$ 400 Btu/lb

channel length = 10 - 144 in.

The correlation is extended to channels with nonuniform axial flux distribution by

$$q''_{DNB,N} = q''_{DNB,EU}/F,$$

where:

$q''_{DNB,EU}$ = equivalent uniform DNB flux, and

$$F = \frac{C}{q''_{local}[1 - \exp(-C\ell_{CNB,EU})]}$$

$$\times \int_0^{\ell_{DNB}} q''(z) \exp[-C(\ell_{DNB,N} - z)]dz$$

$$C = \frac{0.15 \ \frac{(1-X_{DNB})^{4.31}}{(G/10^6)^{0.478}}}{} \ \text{in.}^{-1}$$

where ℓ_{DNB} = length along channel to the point where DNB occurs.

B. Centerline Fuel Melting

While it has been proven possible to operate safely with centerline fuel melting (ref. 14), a limit of no centerline fuel melting is placed on the MH-1A to preclude the possibility of clad damage due to fuel slump. This limit, coupled with the limit on DNBR and the possible range of coolant temperatures, also precludes the possibility of the clad temperature getting high enough to cause either clad melting or a metal water reaction. This limit of no centerline fuel melting is expressed as a maximum permissible linear heat rate, which is primarily a function of the thermal conductivity of the fuel.

The thermal conductivity of UO_2 is based on the Lyons correlation presented in ref. 4:

$$K(T) = \frac{0.6131}{129+T} + 7.685 \times 10^{-15}\ T^3$$

where K(T) is the fuel conductivity in Btu/hr-ft°F and T is the fuel temperature in °K. The above correlation results in an integral value of K(T)dT from 0°C to 2800°C of about 90 w/cm.

Fuel melting is conservatively assumed to occur at 4800°F for irradiated UO_2 (ref. 13).

C. Flow Stability

Periodic oscillations or instabilities could, if they occurred, lead to coolant conditions that would initiate premature burnout. Instabilities presently postulated depend on the large density changes inherent in a violently boiling environment. During normal operation, the MH-1A does not experience any bulk boiling in the hot channel. A steady-state safety limit of 2 percent on exit quality was established, which results in a maximum steady-state void fraction of 0.30. This limit is presently being used by several commercial power reactors and is considered conservative in preventing flow instabilities. The computer codes used in the analysis of transients are able to detect and analyze density and friction-factor-driven instabilities. Again, the proper choice of safety system setpoints will preclude any unstable flow oscillations.

IV. STEADY-STATE THERMAL HYDRAULIC ANALYSIS

The purpose of this section is to describe the thermal hydraulic characteristics of the MH-1A core during normal steady-state operation, and to determine the margins between normal operating conditions and those necessary to cause possible fuel element damage. The steady-state thermal performance of the core is measured with respect to the safety criteria discussed in Section III of this report. These criteria constitute limits on minimum DNBR, maximum fuel centerline temperature, maximum channel exit quality, and hydraulic stability.

In the analysis, peak steady-state fuel centerline temperature is calculated using the Lyon's correlation for thermal conductivity of UO_2 given in ref. 4 as a function of reactor power. An in-house computer program CHEETAH described in ref. 5 is used to perform these calculations. DNBR and maximum quality are determined using the computer code COBRA (ref. 6) to calculate the channel coolant enthalpy and velocity conditions. COBRA is modified to calculate DNBR using Westinghouse W-3 correlation described in Section III.

The maximum steady-state power to melting for MH-1A Core 3 is determined to be 117 percent of full power. Centerline fuel temperature represents the most limiting criterion on the over power capability of Core 3. The most limiting criterion on other operating parameters is the hot channel exit quality of 2 percent. The steady-state analysis shows that the values of operating parameters necessary to reach the quality safety limit of 2 percent (assuming that the remaining raviables are at the alarm setpoint plus instrument error) are: (1) Outlet temperature of 545°F, (2) coolant flow rate of 8350 gpm, and (3) system pressure of 1240 psia.

A. General

The steady-state thermal hydraulic analysis is used to define all those conditions which might reasonably lead to violation of the safety criteria, and to assure that all normal operation occurs within these safety limits with appropriate margins for transient overshoot and instrument error. Normal steady-state operation is assumed to occur within the operating band limits, plus or minus appropriate instrument error as discussed in Section II. Safety limits are developed by varying one or more of the operating parameters (reactor power, coolant flow, temperature, and pressure) beyond the operating band limits until the most limiting safety criterion is reached.

In analyzing the reactor core, the concept of the hot channel is used. This concept assumes that the most adverse fuel properties, and channel or fuel rod dimensions, occur at the location of the most limiting heat flux and coolant flow. This results in one or two channels in the core being most limiting. The thermal hydraulic performance of the core is measured from the behavior of these hot channels.

B. Operating Parameters

The total primary coolant flow through the reactor at nominal operating pressure and temperature is 4.40×10^6 lb/hr. A portion of this flow passes through the inner thermal shields and control rod coolant channels, and is considered coolant leakage flow. The fraction of the coolant flow effective is removing heat from the core is 0.912 of the total flow or 4.01×10^6 lb/hr at nominal conditions. Coolant flow area associated with the effective flow is 5.22 ft^2, resulting in an average core mass velocity at nominal conditions of 0.769×10^6 lb/hr-ft^2. The average core mass velocity at design conditions is $.677 \times 10^6$ lb/hr-ft^2.

Normal primary system pressure is 1400 psia. Operating band and instrument error associated with pressure result in range in actual pressure between 1352 psia and 1477 psia. The nominal full power core inlet temperature is 476°F. Maximum coolant inlet temperature is 498°F corresponding to an outlet temperature of 534°F. Table IV-1 shows the operating parameters for the nominal and design (alarm setpoint plus error) conditions used for the steady-state analysis of Core 3.

TABLE IV-1

MH-1A CORE 3 THERMAL HYDRAULIC OPERATING PARAMETERS

	Nominal	Design
Reactor power, %	100	111
MW	45	50.0
Primary system flow rate, gpm	10,800	9750
lb/hr	4.40×10^6	3.87×10^6
System pressure, psia	1400	1352
Coolant inlet temperature, °F	476	498
Core average coolant mass velocity, lb/hr-ft^2	0.769×10^6	0.677×10^6

C. Hot Channel Factors

In evaluating the thermal-hydraulic performance of the core, the local increase of heat generation and enthalpy rise with respect to the core average value is accounted for by the use of hot channel factors. Two types of hot channel factors are considered: Nuclear hot channel factors which describe the neutron flux distribution in the core and engineering hot channel factors, which account for local variations in fuel rod properties due to fabrication tolerances. Sources of engineering hot channel factors are tolerances on fuel pellet diameter, enrichment, and density and fuel rod loading. The influence of variations in rod diameter, pitch, channel area, and rod bowing are accounted for by direct use of adverse dimensions in the analysis. Hot channel factors on heat flux or linear power depend on local properties (hot spot), and factors on enthalpy rise consider the integrated effect along the length of the channel (hot channel). The overall hot channel factors on heat flux or enthalpy rise are the product of the appropriate individual subfactors.

1. Engineering Hot Channel Factors and Dimensional Considerations.

Hot channel engineering factors used in the analysis of Core 3 are derived from the appropriate tolerances in the fuel rod specifications (ref. 7). The engineering factor on enthalpy rise is taken as the ratio of the maximum U-235 loading to the nominal for a fuel rod (1.030). Engineering factors for local power or heat flux are determined from the tolerances on the UO_2 fuel enrichment, density, and fuel diameter. When determining local fuel rod heat flux, an additional factor is included to account for a decreased heat transfer area caused by a possible minimum fuel rod diameter of 0.504 inches. The derivation of engineering hot channel factors used in the analysis is shown in Table IV-2.

The effect of variations in channel dimensions on enthalpy rise and local DNB heat flux are incorporated directly in the analysis. When calculating the hot channel enthalpy addition, the minimum flow area is used to obtain the highest channel enthalpy rise. However, when determining DNB heat flux using the W-3 correlation (Section III), the maximum enthalpy rise combined with an increased local hydraulic diameter is used to give the most adverse DNB heat flux.

TABLE IV-2

ENGINEERING HOT CHANNEL FACTORS
ON ENTHALPY RISE, LOCAL HEAT FLUX, AND LINEAR HEAT RATE

Fuel Variation	Specification	Engineering Factor
<u>Local Power</u>		
Fuel enrichment, %	4.65 ± .068 w/o	1.015
Fuel density, gm/cm	10.35 ± 0.15	1.014
Fuel pellet diameter, in	0.4564 ± .0005	1.001
Factor on Linear Heat Rate		1.030
Fuel clad outside diameter, in	0.507 ± .003	1.006
Factor on Heat Flux		1.036
<u>Enthalpy Rise</u>		
Fuel rod U^{235} loading, gms	40.968 ± 1.23	1.030
Factor on Enthalpy Rise		1.030

Channel flow area and hydraulic diameter are determined from the fuel rod pitch. The effective minimum fuel rod pitch depends on several factors which include the tolerances on the base plate holes and end caps, as-assembled fuel rod bowing t 'erance, and possible thermal bowing due to a temperature gradient cross the fuel rod. The maximum reduction of the fuel rod pitch from these sources was found to be 0.615 inch. The corresponding minimum flow area and hydraulic diameter are 0.1763 in^2 and 0.4427 inch. Table IV-3 shows the nominal and hot channel characteristics for MH-1A Core 3.

TABLE IV-3

NOMINAL AND HOT CHANNEL CHARACTERISTICS

	Nominal Channel	Hot Channel
Fuel rod diameter, in	0.507	0.510
Fuel rod pitch, in	0.654	0.615
Clad thickness, in	0.023	0.024
Fuel diameter, in	0.4565	0.4570
Channel area, in^2	0.2258	0.1763
Hydraulic diameter, in	0.5670	0.4427
Inter-channel spacing, in	0.1470	0.1080

2. Power Distributions

The radial power peaking factors used in the analysis corresponds to the beginning of life (BOL) power distribution determined in the nuclear analysis of Core 3 (ref. 11) using the computer code TURBO*. Figure IV-1 shows the individual rod power factors for the innermost fuel element which is the highest flux fuel element in the core and contains the hot channel. These values represent the unrodded region of the core for hot condition 490°F and are higher than the power peaking factors in the rodded region of the core. The maximum BOL radial power peaking factor for Core 3 is 1.92.

The axial power distribution for the BOL is shown in Figure IV-2. This flux shape corresponds to a twelve-rod bank position of 10.6 inches at 490°F, and is based on data generated by the one dimensional CNCR-2 code. The curve, as used in the analysis, has a normalized value of unity. Use of the BOL flux distribution is conservative because the axial peak will decrease during the life of the core and more than compensate for any slight increase in radial factors shortly after BOL. For normal operating conditions (490°F) the maximum BOL axial power peaking factor is 1.957.

D. Methods of Analysis

The maximum steady power level with respect to fuel centerline melting is calculated using the in-house computer code CHEETAH (ref. 5). This code calculates the fuel temperature distribution in the hot pellet, taking into account the flux suppression in the center of the fuel in terms of the neutron diffusion length in the fuel.

The upper limit on the coolant temperature is taken as the saturation temperature at given system pressure. The temperature increment between the bulk coolant and clad wall is given by the conservative Thom nucleate boiling correlation (ref. 12):

$$T_{wall} = T_{sat} + .072\ (q'')^{0.50}\ (e^{-P/1260}) \qquad (1)$$

where:

T_{wall} = cladding surface temperature, °F

T_{sat} = saturation temperature for system pressure, °F

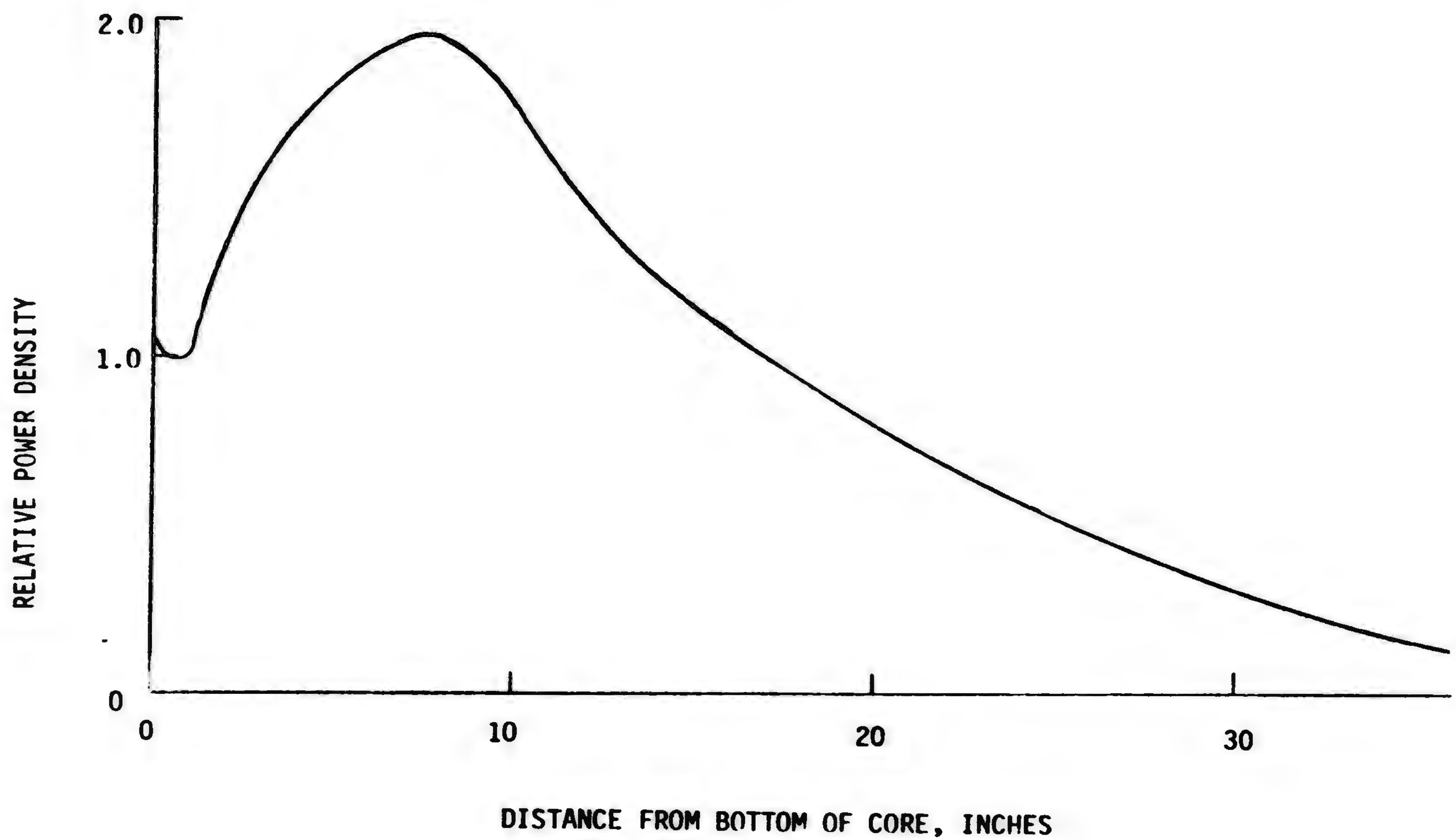
q'' = local surface heat flux, Btu/hr-ft^2

P = system pressure, psia

									1.51	1.21
	1.92	1.77	1.76	1.75	1.73	1.70	1.66	1.64	1.60	1.33
	1.81	1.61	1.60	1.59	1.57	1.55	1.52	1.52	1.51	1.58
	1.81	1.61	1.59	1.58	1.57	1.55	1.52	1.51	1.51	1.62
	1.81	1.60	1.58	1.58	1.57	1.54	1.52	1.51	1.51	1.63
	1.80	1.59	1.58	1.58	1.56	1.53	1.52	1.52	1.52	1.64
	1.805	1.59	1.59	1.58	1.56	1.53	1.53	1.53	1.53	1.67
	1.80	1.59	1.59	1.58	1.56	1.54	1.53	1.53	1.53	1.68
	1.78	1.59	1.59	1.58	1.56	1.54	1.53	1.53	1.54	1.69
	1.76	1.60	1.59	1.58	1.57	1.55	1.53	1.54	1.54	1.69
	1.45	1.66	1.68	1.68	1.68	1.67	1.64	1.65	1.65	1.68
Source	1.22	1.65								

FIGURE IV-1 RADIAL POWER DISTRIBUTION FOR CENTRAL FUEL ASSEMBLY. EACH FUEL PIN REPRESENTED (BOL, HOT)

FIGURE IV-2 MH-1A CORE 3 DESIGN AXIAL POWER DISTRIBUTION, BOL
FOR HOT CONDITIONS (490°F)
ROD BANK 10.63 INCHES WITHDRAWN



Heat conduction through the clad and gap is calculated to give the fuel surface temperature as a function of power level. The fuel centerline temperature is then determined based on the local heat generation in the fuel and the temperature dependent UO_2 thermal conductivity.

The heat conduction equation for cylindrical co-ordinates is solved using a modified zero order Bessel function to represent the heat generation shape in the fuel. The solution to the heat conduction equation as programmed into the CHEETAH code is written as:

$$4\pi \int_{T_{surface}}^{Tr} k(T)\,dT = q' \frac{2\ell}{R} \left(\frac{Io\,(R/\ell) - Io\,(r/\ell)}{I_1\,(R/\ell)} \right) \quad (2)$$

where: q' = peak linear heat rate

r = radial distance from the fuel centerline

R = radius of the fuel

$k(T)$ = thermal conductivity of UO_2

$T_{surface}$ = fuel surface temperature

Tr = fuel temperature at radius

Io = modified zero order Bessel function

I_1 = modified first order Bessel function

ℓ = neutron diffusion length in fuel

The program calculates the fuel temperature distribution based on equation (2) for various steady-state power levels, and determines the linear heat rate at which a specified fuel melting temperature is reached at the center of the fuel.

The basic method used in the analysis of the MH-1A hot channel enthalpy rise and DNB ratio conditions is the COBRA digital computer program (ref. 6). The COBRA program calculates coolant flow and enthalpy in the subchannels of an arbitrary rod bundle. The program has the ability to consider both single and two phase flow, and accounts for the flow redistribution and thermal mixing between adjacent flow channels resulting from turbulent cross flow and diversion cross flow. An arbitrary heat flux distribution can be input by specifying the axial flux distribution, relative rod power, and the fraction of rod power to the adjacent subchannels.

Differences in hydraulic resistance due to local boiling cause flow redistribution which may increase the hot channel enthalpy rise. COBRA calculates the flow distribution in the subchannels by forcing the pressure drop in each channel to be identical. Turbulent mixing tends to reduce the hot channel enthalpy rise caused by local power peaking and adverse mechanical dimensions. The effect of turbulent mixing has been previously considered for the MH-1A in Reference 8. A value for the turbulent mixing parameter of .0075 was determined to best represent MH-1A subchannel conditions where the correlation for β is written:

$$\beta = 0.0062 \frac{D}{S} (R_e)^{-0.1} \qquad (3)$$

where:

D = the hydraulic diameter (inches)

S = the rod spacing (inches) and

R_e = the Reynolds number.

Extensive experimental data (ref. 9) have verified the accuracy of COBRA in predicting subchannel conditions in a multirod geometry.

COBRA is modified to calculate DNB ratios utilizing the W-3 DNB correlation and calculated channel conditions. The W-3 correlation includes the effect of a nonuniform heat flux distribution, and takes into account both local and upstream channel properties in determining the local DNB heat flux.

E. Fuel Temperature Power Distribution

The maximum fuel temperature occurs in the highest power fuel rod in the core. The maximum BOL axial peaking factor is 1.957 and is located in the unrodded region. The maximum radial factor is 1.92 and the total engineering factor on local power is 1.030. Thus the overall peaking factor on local power, including a 5 percent calculational uncertainity in the axial and radial direction, is $1.957 \times 1.92 \times 1.030 \times (1.05)^2 = 4.267$. In determining the heat generation in the fuel, an additional factor of 0.966 is applied to account for the fraction of the power generated outside of the fuel.

F. Bulk Boiling and DNBR Hot Channel

The hot channel, from the standpoint of enthalpy rise, was determined to be near the corner of the centermost fuel element. The channel configuration used in COBRA is shown in Figure IV-3. In order to calculate the highest heat addition for the hot channel, the four fuel rods surrounding the channel are increased by the engineering factor of 1.030 so that they contribute an increased heat flux into the hot channel. The channel area and hydraulic diameter are reduced to the minimum in the hot channel. The radial power peaking factors are increased by a factor of 1.05 to account for uncertainties in the calculation of the power distribution. Uncertainties in the axial direction tend to average out to zero along the length of the channel, and the factor in this direction is unity when calculating channel enthalpy rise.

The minimum DNB ratio coincidently occurs in the same channel as the maximum enthalpy rise (Figure IV-3). When calculating the maximum local heat flux, the average channel heat flux from COBRA is increased by the ratio of the hot spot engineering factor to the channel averaged factor, and the ratio of the highest radial power peaking factor to the average radial factor for the channel. The heat flux is reduced by 0.966 to account for the fraction of power generated directly in the coolant. For conservatism, the hydraulic diameter is increased to the nominal value when used in the W-3 DNB correlation to account for possible local widening of the channel. Factors of 1.05 in the axial and radial direction are included to allow for the uncertainties in the calculation of the power distribution. The axial power shape for BOL has a peak of 1.957, and the highest radial peaking factor is 1.92 corresponding to the unrodded region of the core where the maximum heat flux occurs. The overall power peaking factor on heat flux is $1.92 \times 1.957 \times 1.036 \times (1.05)^2 = 4.292$.

A summary of the hot channel factors for enthalpy rise and heat flux is given in Table IV-4. These factors do not include the effects of adverse channel dimensions or flow distribution and mixing which are calculated directly in the analysis.

TABLE IV-4

HOT CHANNEL FACTORS FOR ENTHALPY RISE AND HEAT FLUX

Designation	Source of Factor	Factor
Enthalpy Rise		
Nuclear	Radial "hot channel"*	1.778
	Uncertainty	1.050
Engineering	U^{235} loading per fuel rod	1.030
TOTAL		1.923

* Defined as average peaking factor of four rods surrounding the hot channel.

TABLE IV-4 (cont'd)

Designation	Source of Factor	Factor
Heat Flux		
Nuclear	Radial "hot spot"	1.92
	Axial (BOL 490°F)	1.957
	Uncertainty	$(1.050)^2$
Engineering	UO_2 diameter, enrichment density, fuel rod diameter	1.036
TOTAL		4.292

G. Results

1. Fuel Temperature Distribution

The temperature distribution from the surface of the fuel to the centerline is primarily a function of UO_2 thermal conductivity and the local power density. The fuel surface temperature is affected by the cladding temperature and thermal conductance across the gap. The occurance of nucleate boiling results in a cladding surface temperature of less than 612°F for the maximum system pressure in the operating band (1477 psia). A gap conductance of 1763 Btu/hr-ft^2 at the hot spot is used in the analysis.

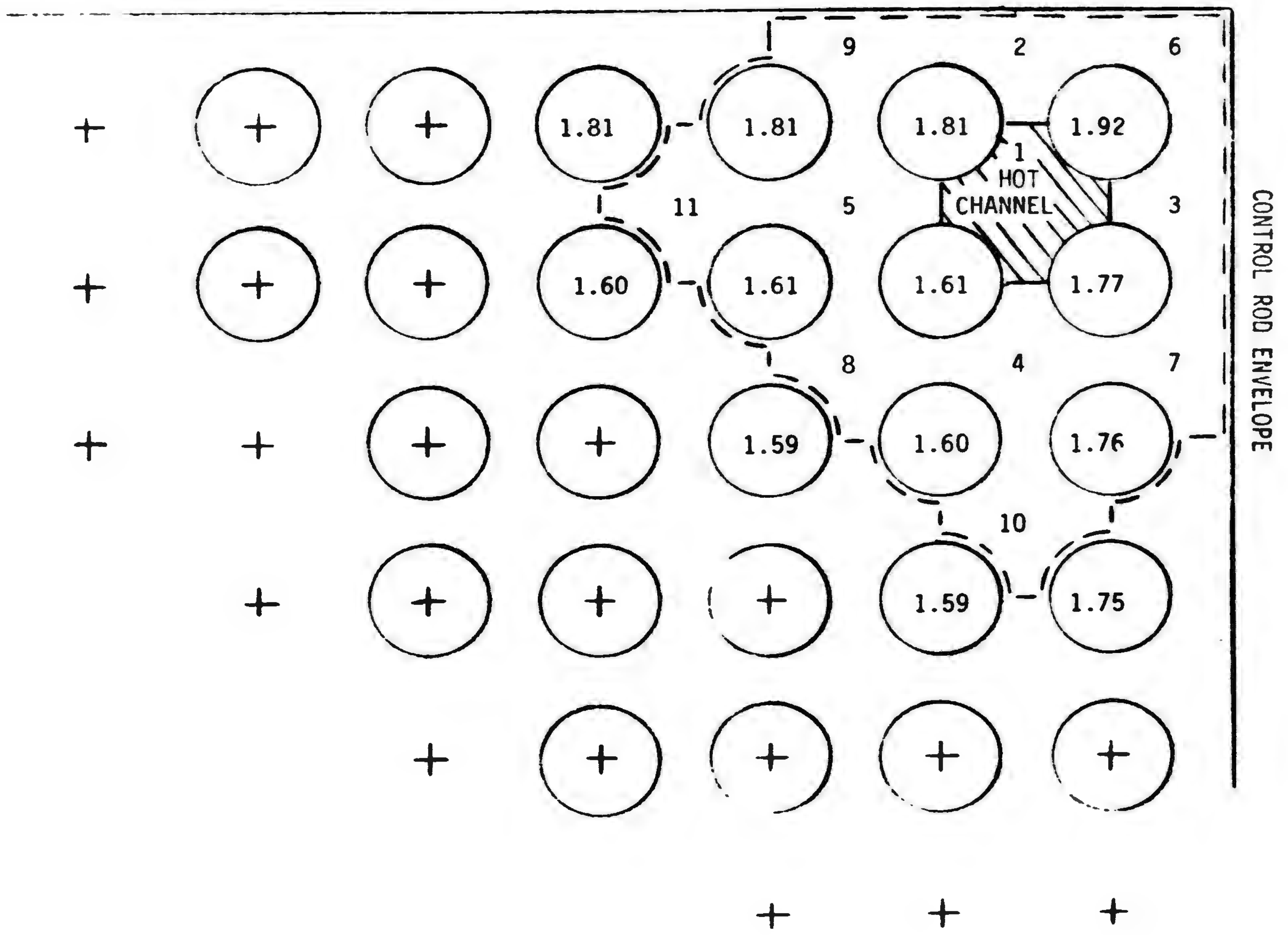
Thermal conductivity of the UO_2 was evaluated from the design equation for thermal conductivity presented in Section III of this report. This equation is the Lyons correlation given in reference 4. The area under the curve is such that the integral $\int_0^{2800^\circ C} k\,dt$ is equal to approximately 90 w/cm. This average value given by reference 4 is based on the determination of UO_2 molten boundary location produced by accurately known thermal performance conditions.

In-house calculations using the CHEETAH code have indicated that assuming a flat flux distribution across the fuel pellet and a melting point of 4800°F, a linear heat rate of 20.3 kw/ft will cause melting of the UO_2. However, due to the high fission cross section of the UO_2, the majority of the fissions occur in the outer portion

FIGURE IV-3

MH-1A CORE 3 COBRA GEOMETRY FOR HOT CHANNEL ENTHALPY RISE AND DNBR

CONTROL ROD ENVELOPE



of the fuel pellet, resulting in a flux suppression at the center of the fuel. The effect of this flux suppression in reducing the temperature at the center of the fuel is a function of U_{235} enrichment and fuel diameter as given in reference 10. For a 4 percent enrichment estimated for the irradiated MH-1A fuel pin, the flux suppression factor would equal 0.935 as shown in figure IV-4. This corresponds to a neutron diffusion length of approximately 0.200 inches as calculated by the CHEETAH computer program. The linear heat rate to centerline fuel melting is then 21.7 kw/ft.

Based on the above information, the maximum centerline temperature of the hot spot at rated power is 4106°F for conditions shown in Table IV-5. This is well below the melting point of UO_2 which is conservatively assumed to be about 4800°F. The maximum linear heat rate at rated power is calculated using the Core 3 total peaking factor of 4.267 as:

$$q'' = \frac{(45{,}000 \text{ kw})\ (.966)\ \times\ (4.267)}{(3328 \text{ rods})\ (\ 3 \text{ ft/rod})} = 18.6 \text{ kw/ft}$$

Using a linear heat rate to melting of 21.7 kw/ft, the maximum overpower capability of Core 3 is then:

$$\text{Power} = \frac{21.7}{18.6} \times 100 = 117\%$$

Figure IV-5 shows radial fuel temperature profiles determined from the fuel temperature analysis program CHEETAH based on the maximum bulk water temperature of 595°F for various steady-state power levels. The central UO_2 temperature is 4575°F at maximum steady-state design conditions (111 percent power), and 4734°F during the maximum postulated overpower condition (115 percent power), resulting from a slow rod withdrawal transient discussed in Section V-2.

FIGURE IV-4 FLUX SUPPRESSION FACTOR VS. UO_2 FUEL ENRICHMENT

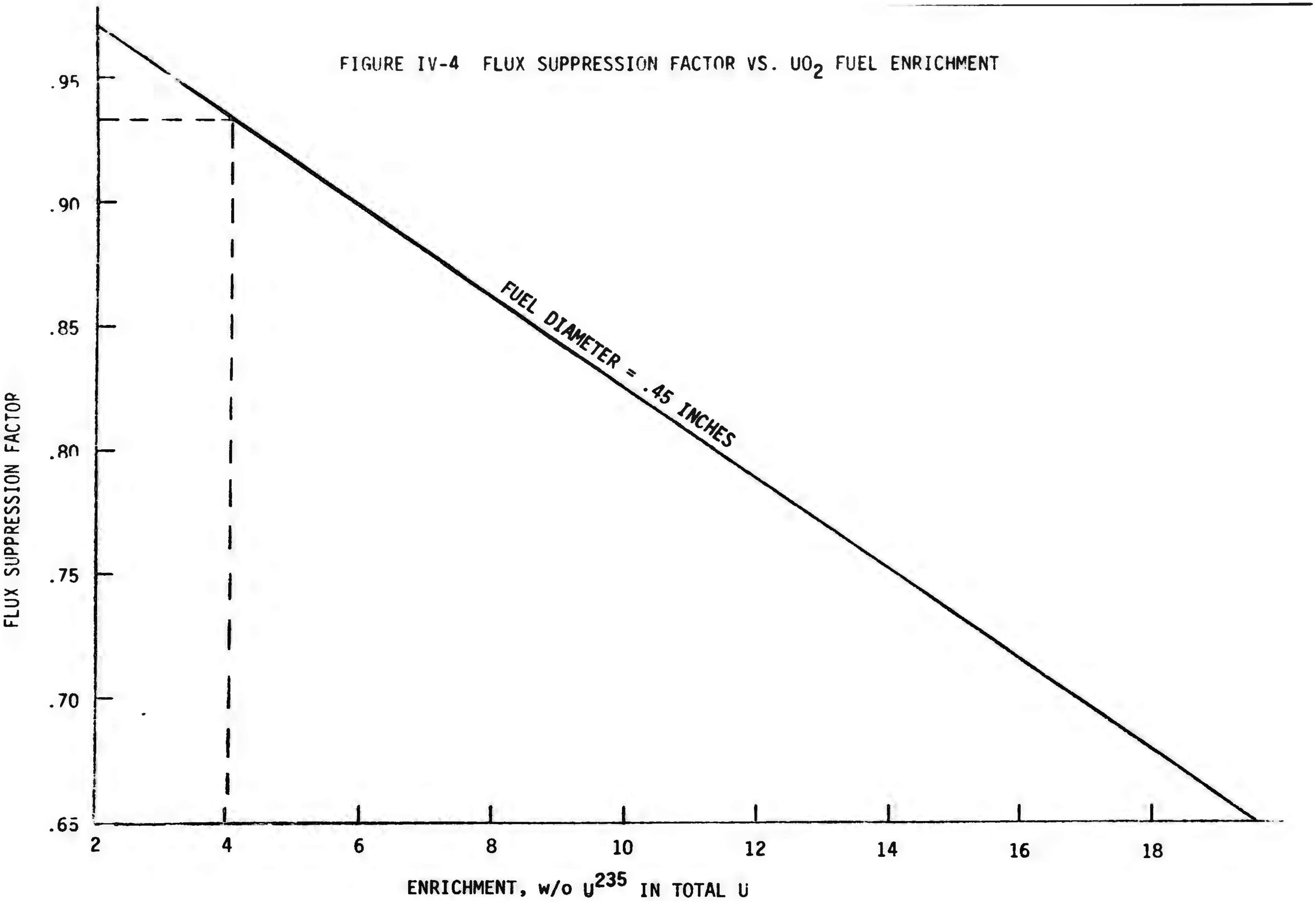


FIGURE IV-5
MH-1A CORE III STEADY STATE FUEL TEMPERATURE
DISTRIBUTIONS AT THE HOT SPOT FOR VARIOUS
REACTOR POWER LEVELS

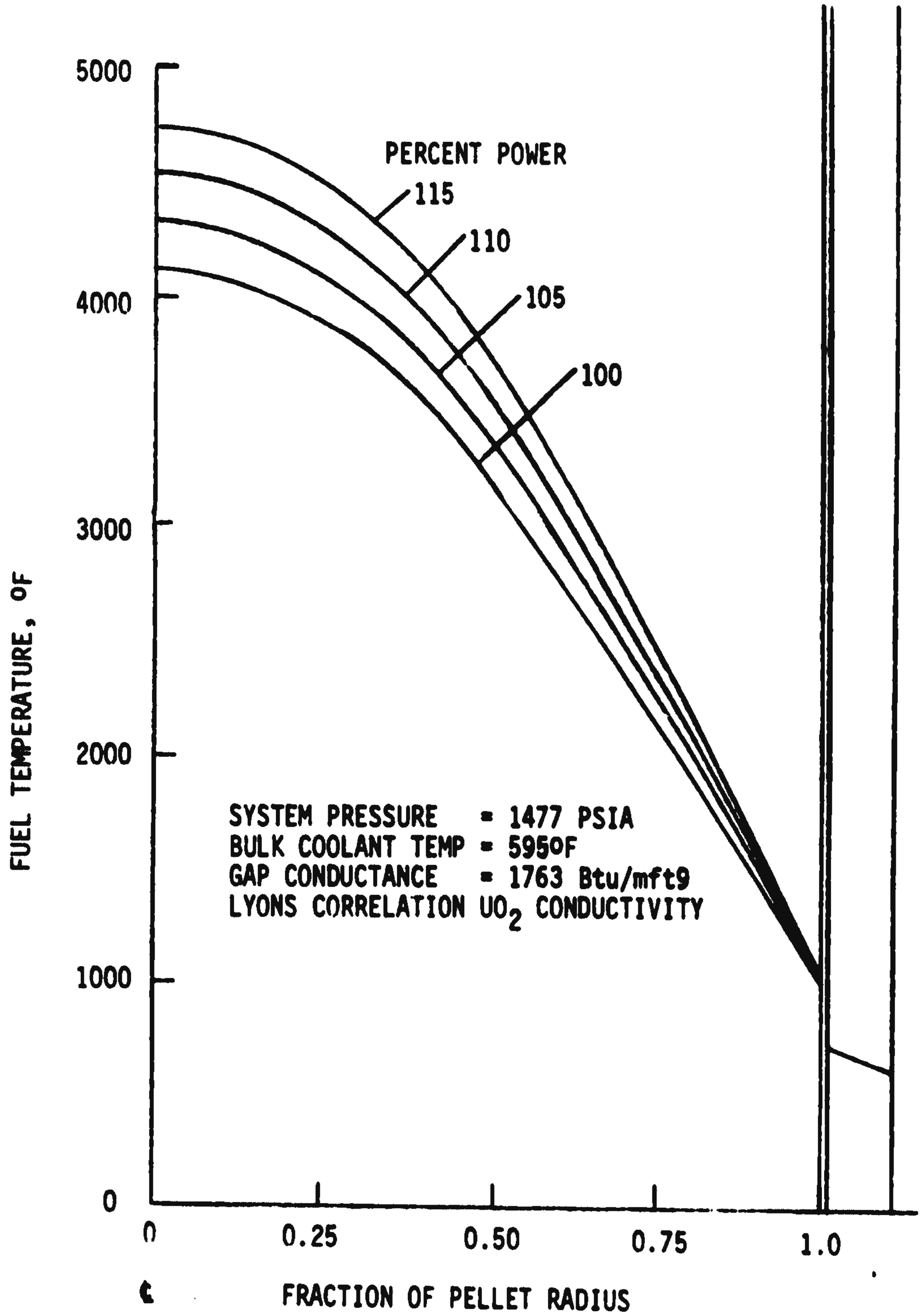


TABLE IV-5

MH-1A FUEL TEMPERATURE ANALYSIS PARAMETERS

Rated power, MWt	45	
Heat generated in fuel, %	96.6	
Maximum System pressure, psia	1477	
Coolant temperature (saturation), °F	595	
Hot Channel Factors		
Nuclear, axial	1.957	
radial	1.92	
uncertainty	$(1.05)^2$	
Engineering	1.030	
TOTAL	4.267	
Fuel diameter, in	0.4565	
Clad outside diameter, in	0.507	
Clad thickness, in	0.023	
Core length, in	36	
Number of fuel pins	3328	
Heat transfer		
Clad surface heat transfer mode	nucleate boiling, Thom correlation	
Thermal conductivity of clad, Btu/hr/ft°F	11	
Gap conductance Btu/hr-ft^2°F	1763	
UO_2 thermal conductivity	Lyons correlation	
	Rated power (45 MW)	Scram Condition (51.8 MW)
Maximum heat flux (hot spot), Btu/hr-ft^2	0.481×10^6	0.553×10^6
Max linear heat rate, kw/ft	18.6	21.4
Maximum clad surface temperature, °F	610	612
Fuel temperature maximum centerline, °F	4106	4734
Fuel temperature maximum surface, °F	996	1055

2. Enthalpy Rise and Dnb Hot Channel Characteristics

The thermal hydraulic conditions in the hot channel, calculated using the COBRA computer program, are shown in Table IV-6. Characteristics of the average channel in the core (withou· nuclear and engineering hot channel factors and constrained to nominal channel dimensions) is shown for comparison. The results indicate that during normal steady-state operation at the design limits, no bulk boiling occurs in the hot channel. The minimum DNB ratio under these conditions is 1.58. The overall effect of variations in channel dimensions, fuel rod properties, and flow distribution and interchannel mixing, is shown in Table IV-6. These effects result in an increase in the hot channel enthalpy rise compared to the nominal channel of

$$\frac{96.3 \text{ Btu/lb}}{(1.778)\ (1.05)\ \ 43.6 \text{ Btu/lb}} = 1.18$$

The effect of turbulent mixing on the hot channel enthalpy rise is calculated by the COBRA computer program based on the turbulent mixing parameters and the fuel assembly power distribution. The results of the calculation may be expressed as a flow mixing factor defined as the ratio of the hot channel mixed to unmixed enthalpy rise. The effects of mixing and flow redistribution taken separately result in a reduction of hot channel enthalpy rise of about 0.86. Flow mixing partially compensates for the increase of hot channel enthalpy rise resulting from uncertainties in the local heat generation and adverse channel dimensions.

Behavior of the hot channel at conditions other than the operating band limits was investigated using the COBRA program. The variation of hot channel exit quality and minimum DNB ratio as a function of reactor power, outlet temperature, flow, and pressure are shown in Figures IV-6, IV-7, IV-8, IV-9, respectively; in each case, the remaining operating parameters were assumed to be at either the design limit (alarm setpoint plus error) or the normal operating limit.

3. Margins to Safety Criteria

During normal operation (rated power) at the maximum system pressure, the peak centerline temperature of the hot pellet is 4106°F. The safety criteria of fuel melting (4800°F) is reached at an overpower level of approximately 117 percent for MH-1A Core 3. For steady-state operation, fuel centerline temperature is relatively unaffected by other operating parameters.

The minimum DNB ratio during steady-state operation at design conditions is 1.58. The analysis shows that actual DNB due to high heat flux is not as limiting on reactor parameters as fuel temperature or hot channel quality.

	NORMAL OPERATING LIMITS	DESIGN LIMITS
PRESSURE, PSIA	1352	1352
INLET TEMP, °F	483	498
FLOW, GPM	10,200	9750

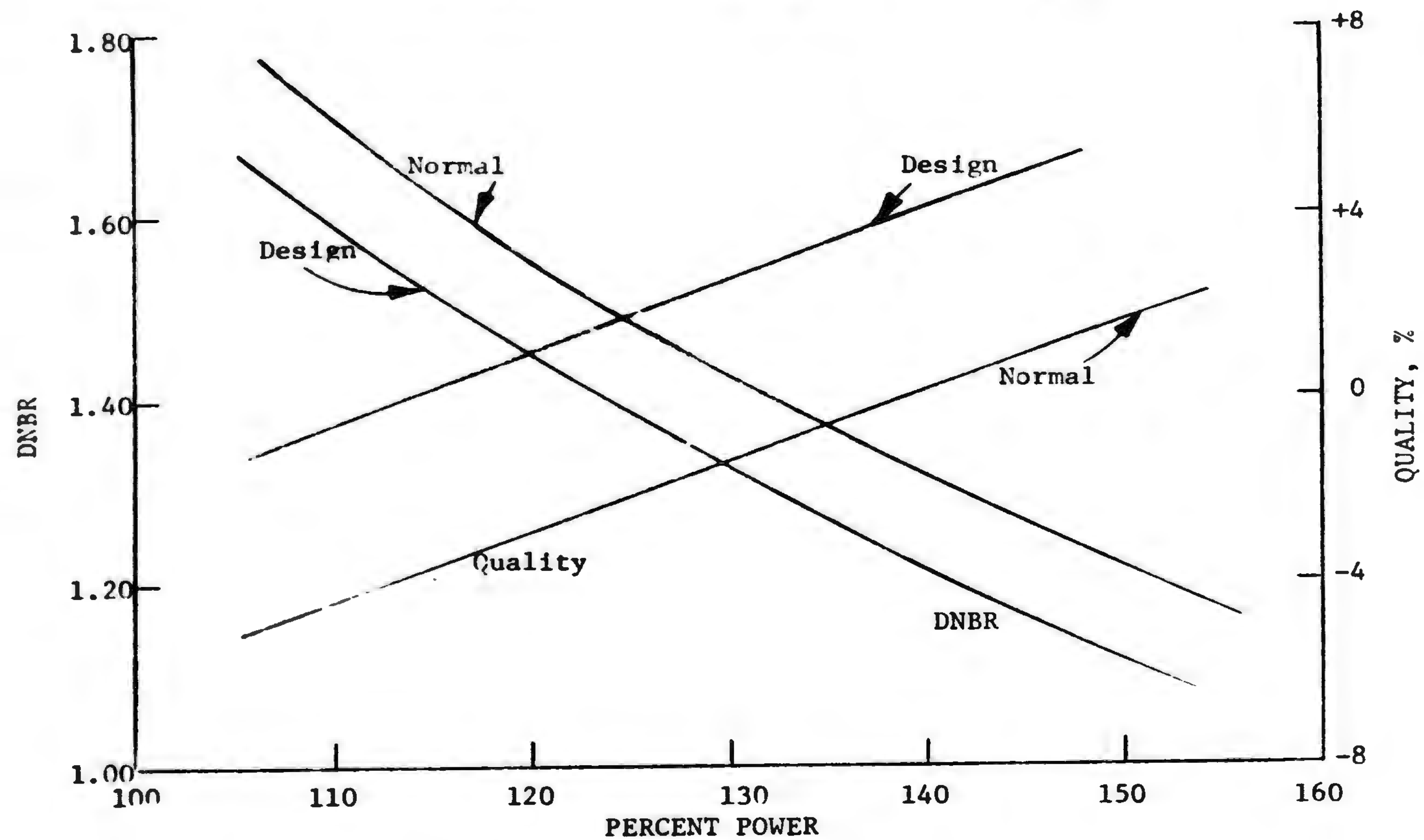


FIGURE IV-6 QUALITY AND DNBR VS POWER LEVEL

	NORMAL OPERATING LIMITS	DESIGN LIMITS
POWER, %	107	111
PRESSURE, PSIA	1352	1357
FLOW, GPM	10,200	9750

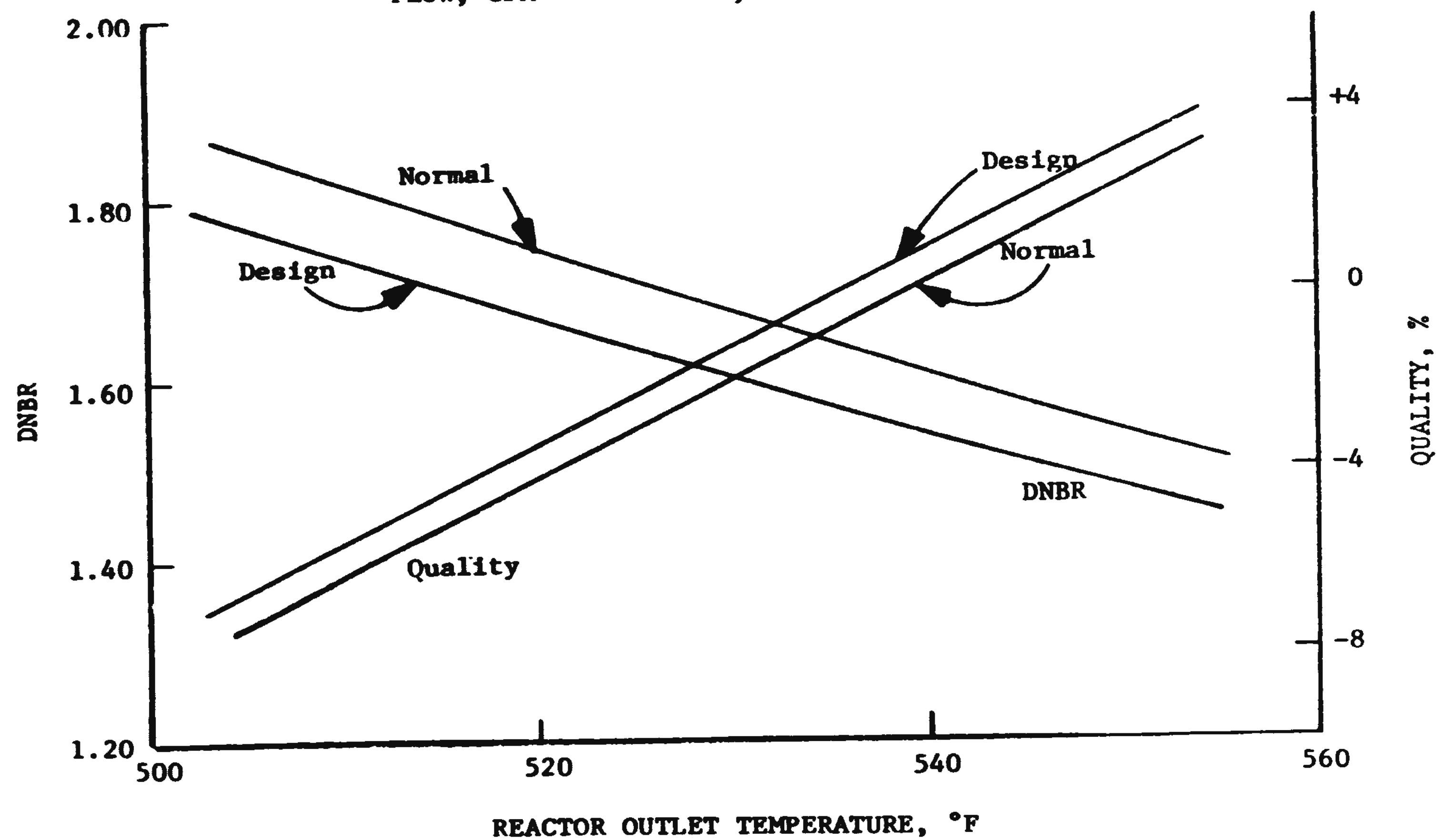


FIGURE IV-7 QUALITY AND DNBR VS REACTOR OUTLET TEMPERATURE

	NORMAL OPFRATING LIMITS	DESIGN LIMITS
POWER, %	107	111
PRESSURE, PSIA	1352	1357
INLET TEMP, °F	483	498

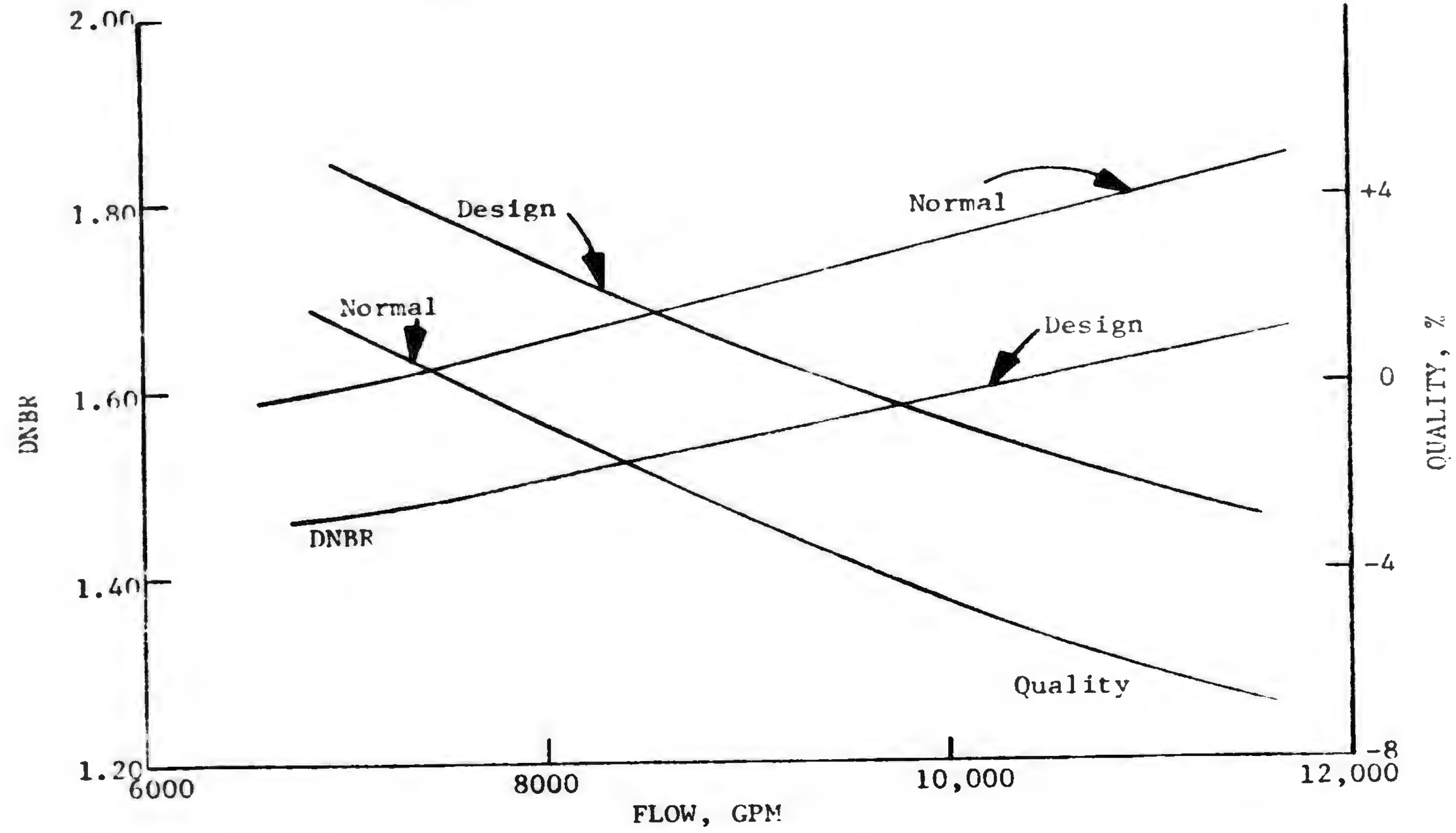


FIGURE IV-8 QUALITY AND DNBR VS FLOW RATE

	NORMAL OPERATING LIMIT	DESIGN LIMIT
POWER, %	107	111
INLET TEMP, °F	483	498
FLOW, GPM	10,200	9750

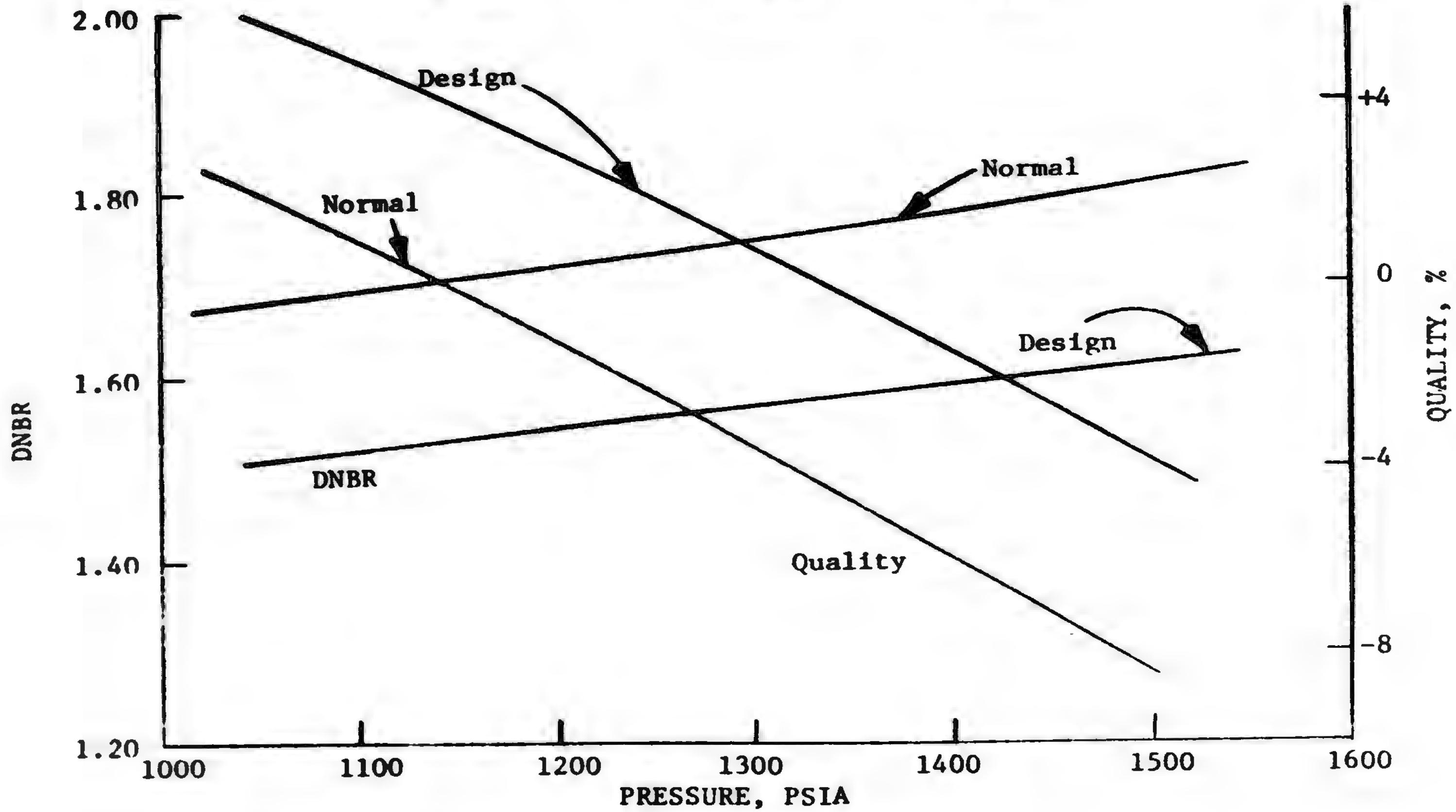


FIGURE IV-9 QUALITY AND DNBR VS SYSTEM PRESSURE

TABLE IV-6

MH-1A CORE 3 AVERAGE CHANNEL AND HOT CHANNEL CHARACTERISTICS

REACTOR CONDITION	NORMAL OPERATING BAND		DESIGN BAND
Power, % 45 MW_t	107		111
Inlet temperature, °F	483		498
Pressure, psia	1352		1352
Flow, gpm	10200		9750
CHANNEL DESCRIPTION	CORE AVERAGE *	HOT CHANNEL	HOT CHANNEL
Flow area, in^2	.2258	.1763	.1763
Hydraulic diameter, in	.567	.442	.442
Channel heat flux, 10^6 Btu/hr-ft^2	0.120	0.230	0.238
Power input, Btu/hr	49,400	94,700	98,200
Mass velocity, 10^6 lb/hr-ft^2	.720	.690	.657
Flow (channel average), lb/hr	1,130	845	804
Enthalpy rise, Btu/lb	43.7	96.3	105.8
Inlet enthalpy, Btu/lb	467.9	467.9	484.5
Outlet enthalpy, Btu/lb	511.6	564.25	590.3
Saturation enthalpy (1352 psia)	593	593	593
Peak local heat flux, 10^6 Btu/hr-ft^2	--	0.514	0.534
Min. DNB ratio	--	1.77	1.58

* Nominal channel dimensions with nuclear and engineering hot channel factors equal to unity, no mixing.

The hot channel exit enthalpy is less than the saturation enthalpy for steady-state operation at design conditions. If the reactor power is increased, the criteria of 2 percent quality is reached at 127 percent power. The DNB ratio of 1.30 is not reached until a power level of 132 percent, which exceeds the limiting power level for both fuel temperature and exit quality. Increasing the reactor temperature with other parameters held at their operating band limits results in a hot channel exit quality of 2 percent at reactor outlet temperature of 545°F. Similar limits on coolant flow and system pressure with respect to hot channel quality are 8350 gpm and 1240 psia.

Thus the most limiting criteria on reactor power is fuel temperature, and the most limiting criteria on the other operating parameters temperature, pressure, and flow is hot channel quality.

4. Safety Limits

Safety limits are developed by varyirg one or more system parameters beyond the operating band limits until the most limiting safety criterion is reached. The safety limits for MH-1A Core 3 as a function of the thermal power level and reactor outlet temperature for various system pressures are shown in Figures IV-10 and IV-11. The design flow of 9750 gpm, and 89 percent of design flow or 8650 gpm, were assumed in determining these curves. The safety limit is considered to be exceeded if the combination of reactor power and coolant outlet temperature is above the appropriate pressure line. At normal power levels these curves represent the locus of points at which the hot channel exit quality is equal to 2 percent. The limits on DNB ratio of 1.30 are not reached until higher power levels in excess of the fuel melting limit on power level of 117 percent.

Table IV-7 summarizes the nominal and hot channel or hot spot characteristics for Core 3 for the operating parameters at the (1) normal operating limits, (2) the alarm setpoints plus instrument error and (3) the scram setpoints plus instrument error on reactor power, temperature and pressure. Data from this table indicates that none of the core safety criteria are violated during steady-state operation up to the reactor scram setpoints.

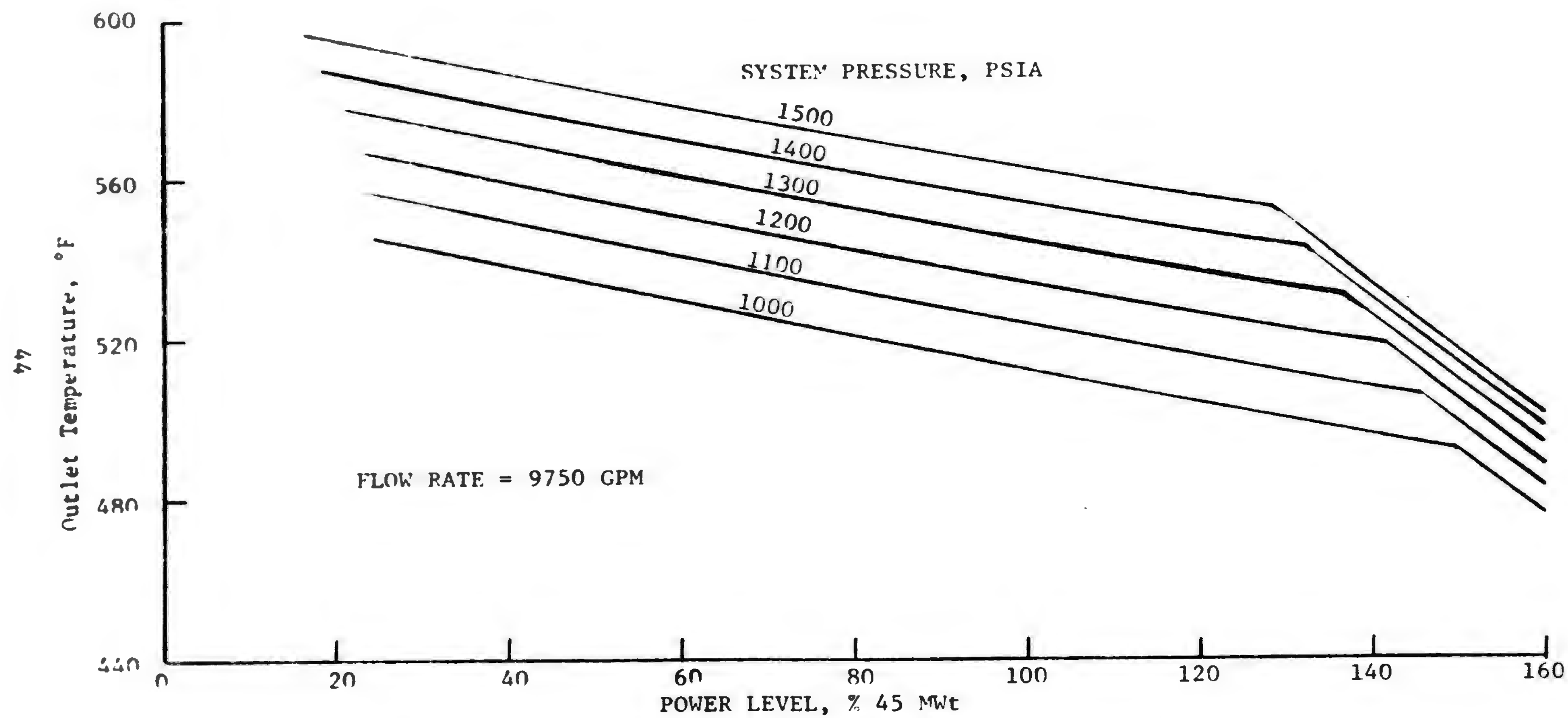


FIGURE IV-10, MH-1A CORE 3 SAFETY LIMITS ON DNBR AND HOT CHANNEL EXIT QUALITY FOR DESIGN FLOW

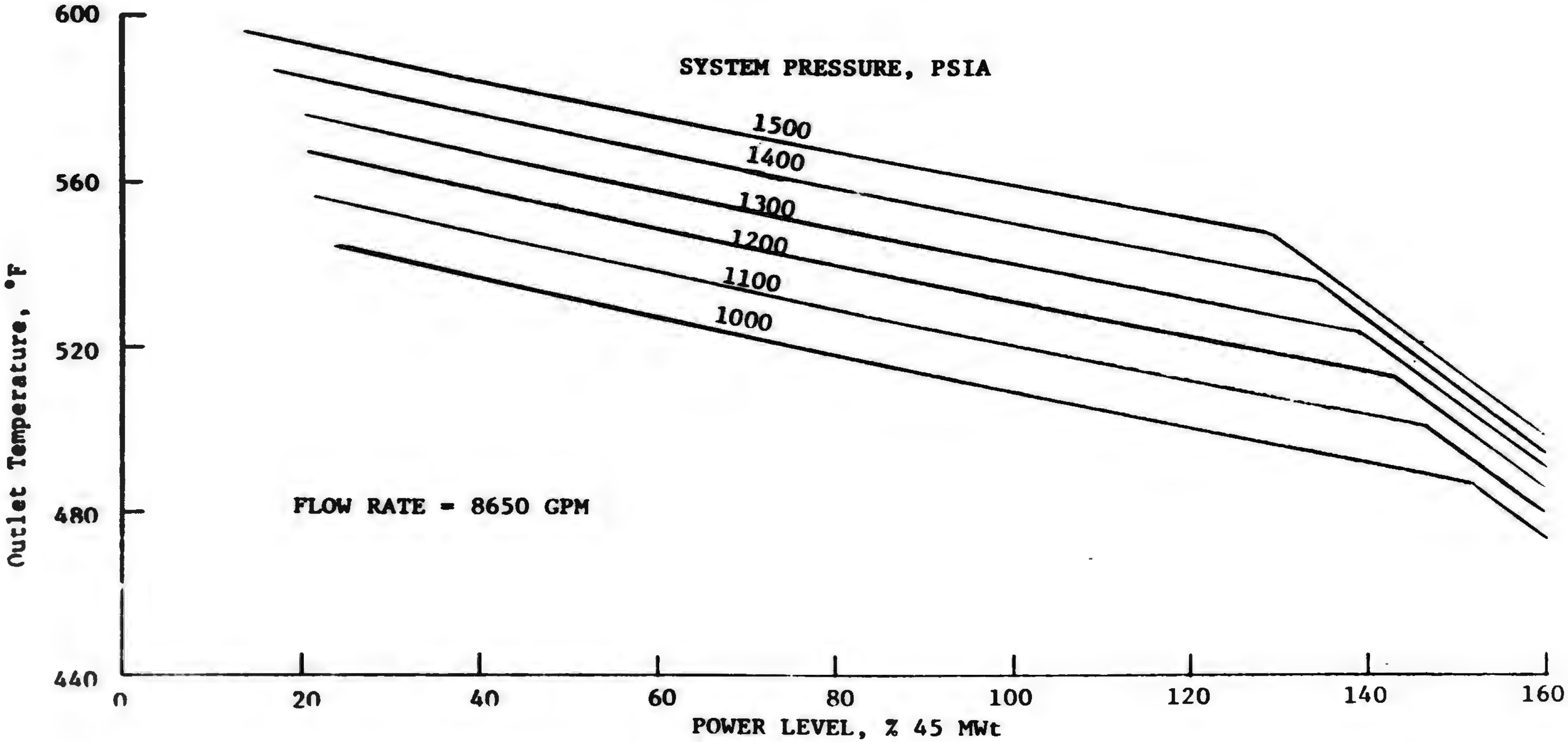


FIGURE IV-11, MH-1A CORE 3 SAFETY LIMITS ON DNBR AND HOT CHANNEL EXIT QUALITY FOR 89% DESIGN FLOW

TABLE IV-7

MH-1A CORE 3 SUMMARY OF HOT CHANNEL AND HOT SPOT CONDITIONS

	Hot Channel Factors		Channel Enthalpy Rise Btu/lb	Exit Quality(1) %	DNB Ratio	Max Fuel ℄ Temp °F
	ΔH	Heat Flux				
Normal Operating Limits						
1. Core average	1.00	1.00	43.7	-13.8	8.18	1172
2. Nominal channel(2)	1.867	4.142	81.6	-7.4	1.92	4270
3. Hot channel	1.923	4.292	96.3	-4.8	1.77	4403
Design Conditions						
4. Nominal channel(2)	1.867	4.142	90.0	-3.1	1.72	4438
5. Hot channel	1.923	4.292	105.8	-0.4	1.58	4575
Reactor Scram Conditions						
6. Nominal channel(2)	1.867	4.142	93.7	-1.5	1.63	4600
7. Hot channel	1.923	4.292	108.4	+1.7	1.51	4734

(1) Negative value of quality denotes degree of subcooling.
(2) Hot channel power distribution without engineering factors or adverse dimensions, no inter-channel mixing.

V. TRANSIENT ANALYSIS

This section presents a detailed analysis of five postulated transients:

(A) The continuous withdrawal of one or more control rods when the reactor is at power (Power Range Rod Withdrawal Transient).

(B) The Rod Creep Transient which is a special case at the Power Range Rod Withdrawal Transient.

(C) The continuous withdrawal of one or more control rods from an initial flux level far below full power (Source Range Rod Withdrawal Transient).

(D) A complete double-ended rupture of the main steam line interior or exterior to containment (Main Steam Line Rupture Transient).

(E) A loss of power to, or a failure of, one or more primary coolant pumps (Loss of Flow Transient).

These transient analyses of the MH-1A are used, with the steady-state thermal hydraulic analyses, to determine and confirm the validity of the safety system setpoints in order to insure core safety at all times.

A. Power Range Rod Withdrawal Transient

1. Background

It is possible during operation in the power range for the control rods to be withdrawn from the core without an increase in turbine load. This causes an initial increase in power which is defined both by the negative temperature coefficient of the coolant and the Doppler coefficient of the fuel. The resultant increase in heat flux with no corresponding increase in heat extraction by the steam generator causes a net increase in primary coolant temperature. Unless terminated, this power mismatch and coolant temperature rise will eventually result in one or more of the safety criteria being violated. To prevent the possibility of such a violation, the reactor safety protection system is designed to terminate any such transient with an adequate margin of safety. Thus it is necessary to analyze the worst cases of this transient to prove this adequacy.

2. Analytical Methods

Employed in the analysis of the power range rod withdrawal transient were the following digital computer programs:

(a) DRAKE

(b) CHIC-KIN

(c) COBRA

DRAKE is a digital code for the simulation of overall plant dynamics employing IBM's Continuous System Modeling Program (CSMP) language (Ref #17). The program solves the reactor kinetics equation with six delayed neutron groups for a four-axial zoned reactor. Reactivity introduced by control rod motion as represented through the use of a nonlinear function generation capability peculiar to CSMP feedback reactivity due to fuel temperature and primary coolant temperature is included.

The model includes a single node steam generator. The core is represented by a single nominal channel with four axial segments. The hot and cold primary coolant legs are represented by pure transport delays, and the reactor vessel inlet and outlet plenums, as well as the steam generator inlet and outlet plenums, are represented by first order mixing delays. The modeling of the secondary side of the steam generator includes the turbine throttle, but does not include effects due to the turbine or condenser. The inlet feedwater enthalpy is assumed constant.

Allowable forcing functions for the DRAKE program are steam generator secondary temperature, turbine throttle position, and reactivity. The details of this program are described in Ref. 8.

The CHIC-KIN computer program is used for intermediate and fast transients and is especially useful when a detailed analysis of the core thermal response is required (Ref 18). The program solves the basic momentum, continuity, and energy equations where any one of the following parameters are known functions of time: Coolant inlet temperature, core inlet flow, core pressure drop, and reactivity or core power level.

The core model considered by the CHIC-KIN program is represented by a single fuel element coolant passage where the coolant channel is closed. A detailed spatial representation of the fuel element by axial and radial sections is possible.

Representation of the fluid dynamics by a momentum integral model which allows flow reversal and spatially distributed pressure drops is also included in the core model. The heat release from the fuel element is described by one dimensional heat conduction to the clad water interface where the effect of the fuel rod time delay is included in the energy balance.

The CHIC-KIN program was used to determine the hot channel/hot spot heat flux and fuel temperatures during power range rod withdrawal transients. Input was the power level trace obtained from DRAKE. The COBRA computer program was described in Section IV.

3. <u>Reactor Kinetics Input Parameters</u>

(a) <u>Reactivity Insertion Rate</u>

The forcing function for rod withdrawal transients is the reactivity insertion rate. Since the MH-1A reactor has twelve control rods, any one or more of which may be withdrawn from the core, there is a spectrum of reactivity insertion rates which must be considered in an analysis of a power range rod withdrawal transient. That is, both slow and fast insertion rates, up to the maximum possible rate, must be considered because of the several ways (heat flux, temperature, etc.) in which a core safety criterion may be violated. A nonlinear function generation capability of DRAKE provides net bank worth given by the twelve rod bank position (see Fig. V-1). Thus the simulation of one through twelve rods withdrawing at any rate up to and including 2 inches/min. is possible.

(b). <u>Delayed Neutron Fraction and Decay Constants</u>

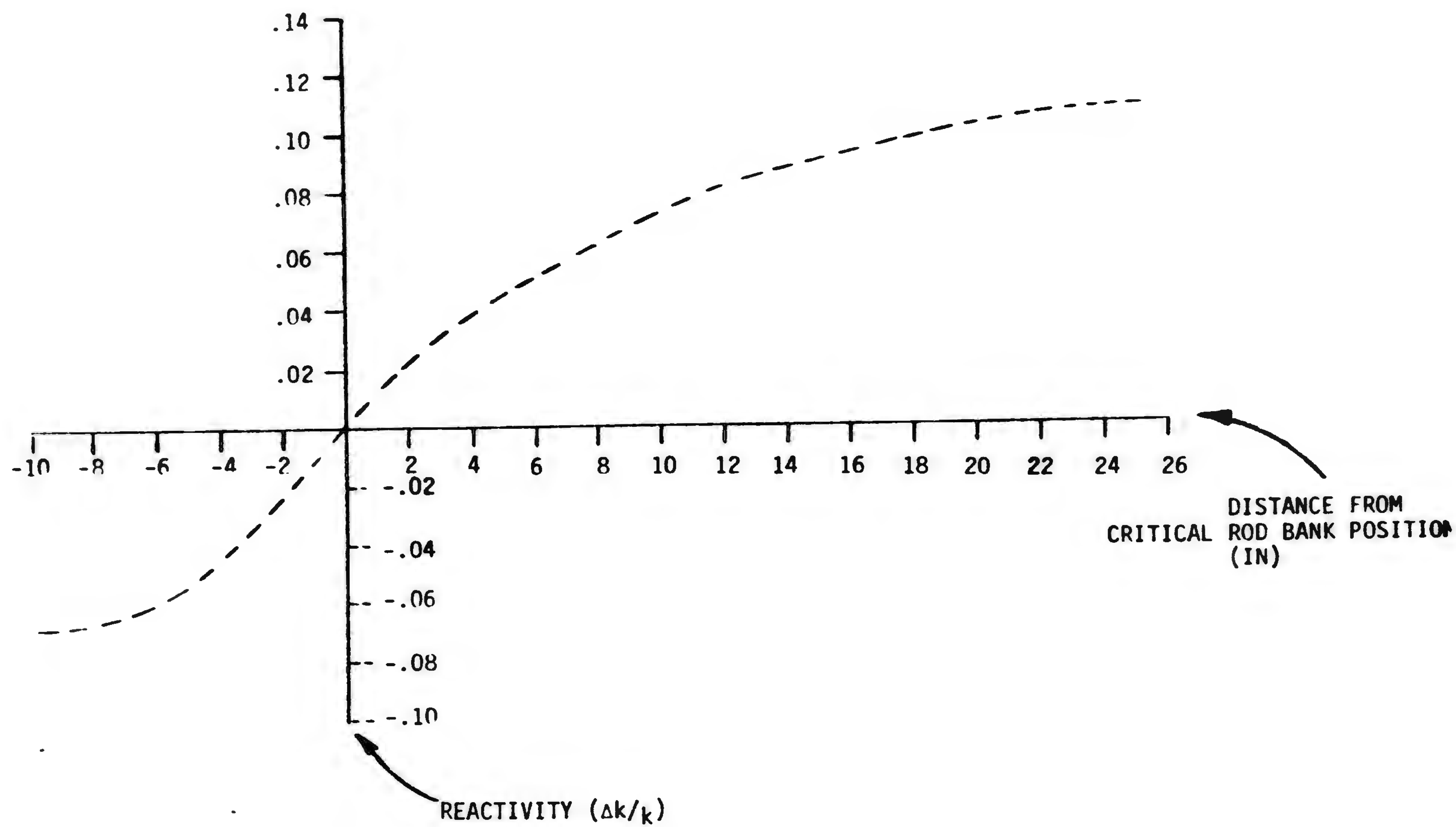
The effective delayed neutron fractions and their decay constants which were used in the reactor kinetics equations are given in Table V-1.

TABLE V-1

Delayed Neutron Fractions and Decay Constants

Group	$\beta_{i\,eff}$	λ_i
1	.000217	.01256
2	.001439	.03096
3	.001328	.11631
4	.002770	.31255
5	.000951	1.25256
6	.000310	3.4285
$\Sigma\beta_{eff}$ =	.007015	

FIG V-1 REACTIVITY ($\Delta k/k$) INSERTED VS. DISTANCE FROM CRITICAL ROD BANK POSITION



(c) Prompt Neutron Lifetime

The prompt neutron lifetime employed in this analysis was 1.301×10^{-5} sec (ref: Nuclear Analysis Section this report).

(d) Moderator Temperature Coefficient

The moderator temperature coefficient at the MH-1A varies with average coolant temperature. The relationship is:

Reactivity ($\Delta K/K$) = (.0056x average coolant temperature + .1756) 7×10^{-5} (ref: Nuclear Analysis Section this report.)

(e) Doppler Coefficient

The fuel temperature coefficient of reactivity (Doppler coefficient) employed in this analysis is:

$$-.95(1.212\times10^{-3}) \frac{\Delta K}{K} \;/\; (°R)^{1/2}$$

Since a Doppler coefficient is a nuclear resonance effect due primarily to the presence of U_{238}, the Core 2 value of

$$1.212 \times 10^{-3} \frac{\Delta K}{K} \;/\; (°R)^{1/2}$$

is used while the factor of .95 provides a conservative margin (Ref 8).

4. Thermal Hydraulic Input Parameters

The transients analyzed were assumed to be initiated from the worst-case operating band conditions of coolant temperature, coolant flow rate, and coolant pressure. The operating bands are described in Section II.

5. Scram Setpoints

The scram setpoints used were 110 percent power (115 percent with included error) and 530°F outlet temperature (534°F with included error).

6. Scram Response Times

The high-power scram response time employed in this analysis was 0.230 seconds. This represented the total time response to scram and included the detector, the power range and safety instrument channels, as well as the control rod clutch current decay time.

7. Hot Channel and Hot Spot Factors

The hot channel factor is applied to the initial reference level of heat generation (Btu/ft^3-hr) which, when input to CHIC-KIN, yields the hot channel enthalpy response.

(a) Hot Channel Factor:

$$1.78 \times 1.03 \times 1.05 \times .852 = 1.65$$

where: 1.78 = average radial peaking factor of the four adjacent pins comprising the hot channel

1.03 = engineering factor on enthalpy rise

1.05 = uncertainty in the radial peaking factor

.852 = a mixing factor describing the effect of flow redistribution and mixing on the hot channel enthalpy rise. Adverse channel dimensions and reduced inlet flow associated with the hot channel are directly input to CHIC-KIN.

Two hot spot factors exist; one is applied to heat flux to yield hot spot heat flux , the other to fuel temperatures to give hot spot fuel temperatures.

(b) Hot Spot Factor Applied to Heat Flux:

$$1.92 \times 1.03 \times 1.05 \times 1.05 = 2.20$$

where: 1.92 = highest radial peaking factor

1.03 = engineering factor on heat flux

1.05 = uncertainty in the radial peaking factor

1.05 = uncertainty in the axial peaking factor

(b) Hot Spot Factor Applied to Fuel Temperatures:

$$1.92 \times 1.03 \times 1.05 \times 1.05 \times .94 = 2.07$$

where all terms are identical to those of the hot spot factor applied to heat flux except:

.94 = a neutron flux suppression factor to correct for U_{235} enrichment and fuel pin diameter (Ref 10).

8. Power Range Rod Withdrawal Transient Initial Conditions

Previous analyses of an MH-1A power range rod withdrawal transient (Ref 8) confirm worst case initial conditions to be those of highest reactor power level and slowest rod withdrawal rate. These initial conditions consistently produced worst case results with regard to DNB ratio and hot spot fuel temperature. Using these results to narrow the choices, three cases were chosen for analysis. In addition, the special case rod creep transient is discussed.

Table V-2 is a list of the parameters and initial conditions for the cases which were used to analyze the power range rod withdrawal transient. Conservative or worst-case values of all parameters and initial conditions were employed. The primary coolant flow rate and pressure were 10,200 gpm and 1352 psia respectively, and correspond to the low end of the operating band of these parameters.

TABLE V-2

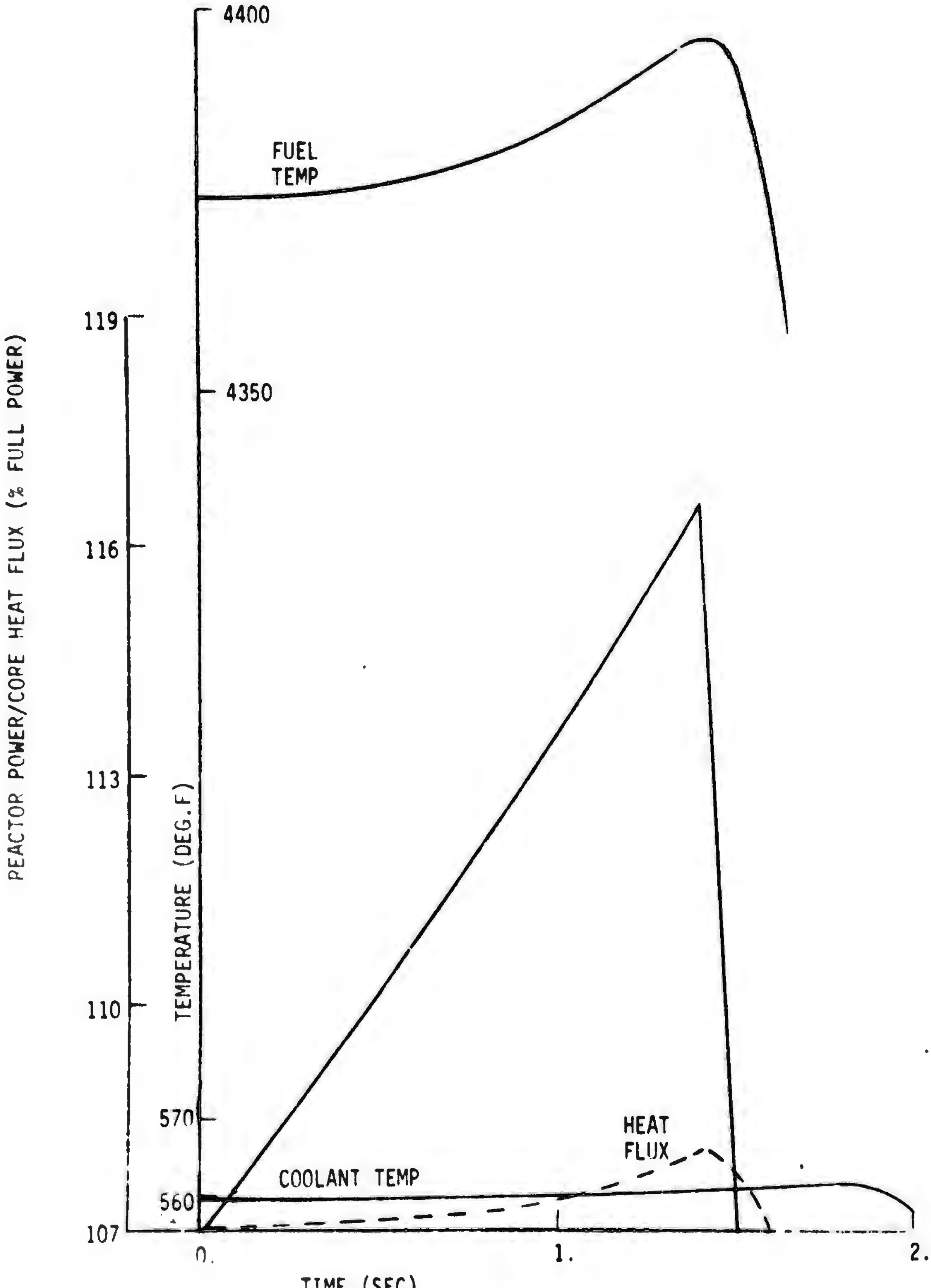
PRRWT INITIAL CONDITIONS

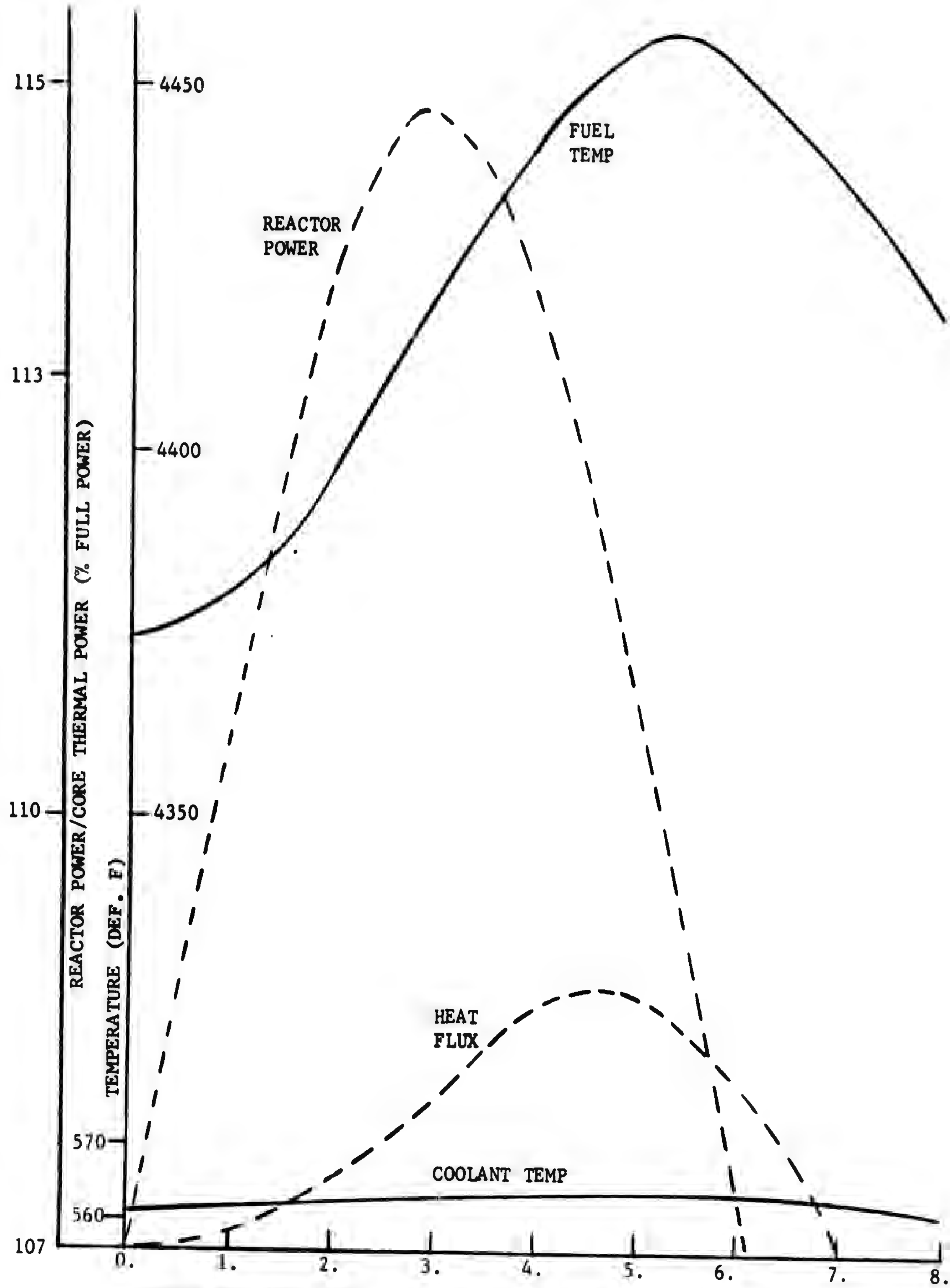
Case #	Rod Withdrawal Rate (in/min)	Core Inlet Temp. (°F)	Initial Power Level (%)
1	2	483	107
2	1	483	107
3	2/3	483	107

9. Results

The behavior of the MH-1A during typical worst-case power range rod withdrawal transients is shown in Figures V-2 and V-3. These figures show the instantaneous power, core heat flux, hot spot fuel temperature, and the hot channel exit coolant temperature, as functions of time during the transient. These figures represent the results from the DRAKE and CHIC-KIN programs for cases 1 and 2 respectively in Table V-2. It can be seen (Fig V-2) for the case of the transient initiated from a high power level and experiencing the maximum allowable rod withdrawal rate, the 110 percent high power scram terminates the transient, little power "overshoot" occurs, and coolant temperature rise is small. The maximum core thermal flux achieved is 108.1 percent full power and hot spot fuel temperatures reach a peak of 4397°F, far below the irradiated UO_2 melting point of 4800°F.

FIGURE V-2 PRRWT REACTOR POWER, CORE HEAT FLUX, HOT SPOT FUEL TEMPERATURE & HOT CHANNEL EXIT COOLANT TEMP VS. TIME FOR INITIAL POWER LEVEL 107% & 2"/MIN ROD WITHDRAWAL RATE





PRRWT REACTOR POWER, CORE THERMAL POWER, HOT SPOT FUEL TEMP & HOT CHANNEL EXIT COOLANT TEMP VS. TIME FOR INITIAL POWER LEVEL 107% & 1"/MIN ROD WITHDRAWAL RATE

FIGURE V-3

Figure V-4 shows core thermal power vs. reactor outlet temperature for this worst case power range rod withdrawal transient employing the maximum reactivity insertion rate. Since the framework for all such transients is the set of safety limits defined in the steady-state analysis (Section IV of this report), the limiting conditions here are 2 percent quality and 4800°F fuel temperature. The minimum DNB ratio safety limit of 1.3 would be violated subsequent to those of quality and fuel temperature, and thus is of no concern. Figure V-4 verifies all safety limits held safe.

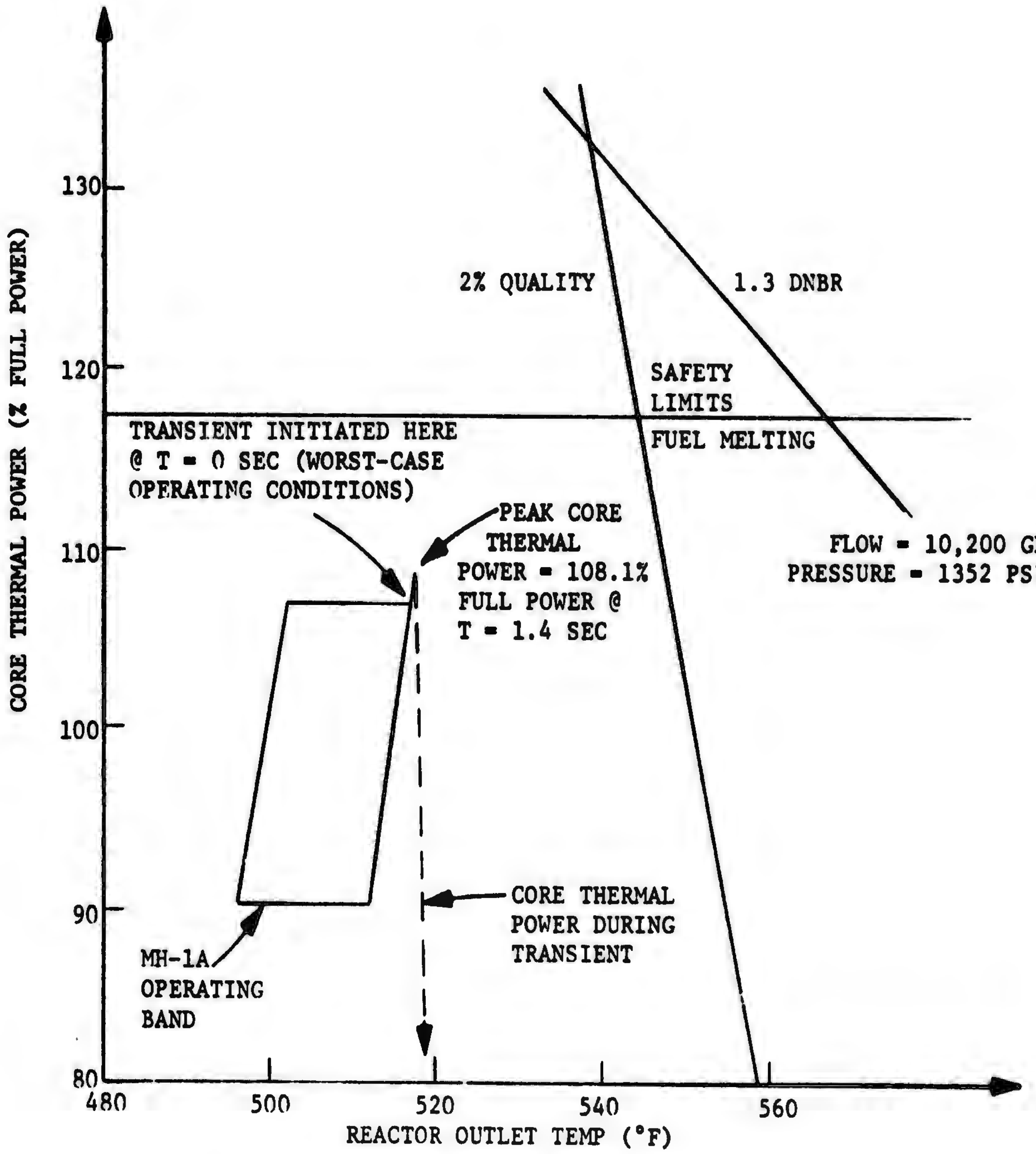
For slow reactivity insertion (low rod worth, or withdrawal of less than all control rods), the combined negative fuel and coolant temperature reactivity coefficients result in a slowing down of the power rise, a power peak, and subsequent decay (see Fig. V-3). Since the energy removed from the steam generator is constant (107 percent of full power), the power mismatch results in an increase in the temperature of the primary system. The transient will be terminated by a high coolant temperature scram (a setpoint of 530°F was employed). But since a maximum core thermal flux and hot spot fuel temperature of 108.8 percent and 4458°F respectively have been achieved, and the 2 percent quality safety limit is in no danger of violation (see Fig V-5), the transient now becomes, more properly, a less than worst case rod creep transient. That is, a quasi-steady-state behavior wherein it is assumed coolant temperature remains at its highest value during the transient, and reactor power remains at the scram setpoint f[illegible] a significant length of time. Rod creep transients are examined in detail in Para. B which follows.

Thus, the limiting combinations of 2 percent quality and maximum fuel temperature (4800°F) are verified held safe for the most severe power range rod withdrawal transients. These safety limits then assure safe operation of the MH-1A, during such a transient, with adequate margin.

B. Rod Creep Transient

1. Background

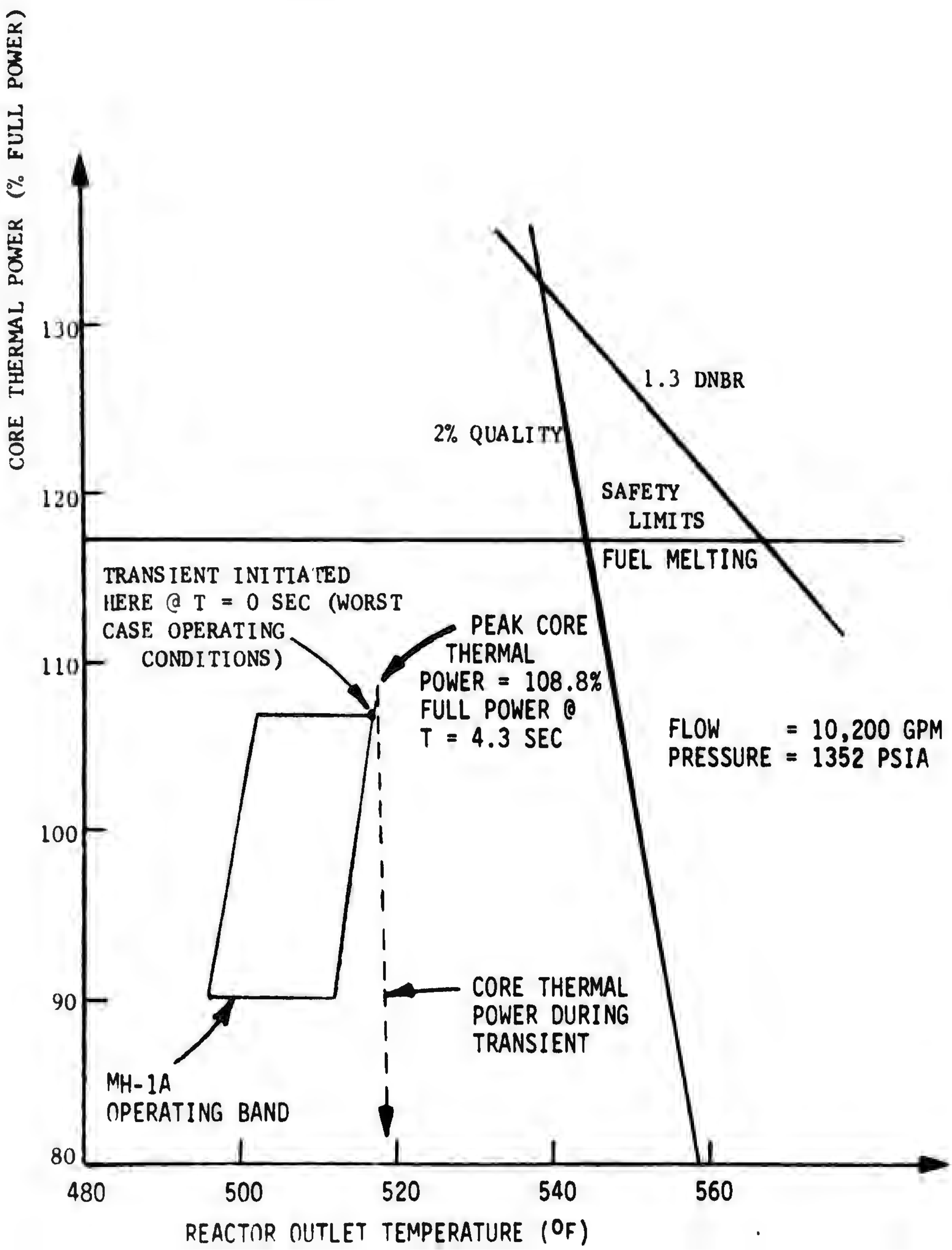
Of major interest in any safety analysis is the so called "rod creep" transient (RCT). A special case of the PRRWT, this is the only transient considered where two scram setpoints are reached. During this transient, it is assumed that the rods are uncontrollably withdrawn and the power increased to just below the level of the scram setpoint. Here the power is assumed to level out due to a balance between the reactivity subtracted by the negative temperature coefficient and that added by the control rod motion. Thus, the high temperature scram is the only remaining safety system protection which will terminate the RCT.



PRRWT CORE THERMAL POWER VS REACTOR OUTLET TEMPERATURE FOR INITIAL POWER LEVEL 107% & 2"/MIN ROD WITHDRAWAL RATE

FIGURE V-4

FIGURE V-5 PRRWT CORE THERMAL POWER VS. REACTOR OUTLET TEMPERATURE FOR INITIAL POWER LEVEL 107% & 1"/MIN ROD WITHDRAWAL RATE



2. Initial Conditions

The conditions existing immediately prior to scram initiation are:

(a). Power = Scram setpoint plus instrument error

(b). Vessel outlet temperature = Scram setpoint plus instrument error plus overshoot due to scram delay and lag in measurement.

(c). Primary system pressure = Worst case operating band (WCOB) plus increase due to heat-up transient.

For power this is just 110% + 5% = 115%

For temperature, the result is somewhat more complicated due to overshoot. The scram setpoint plus error is 530°F + 3.6°F. = 533.6°F

3. Results

In calculation of the overshoot, the important factors are power mismatch, temperature measurement lag, and scram delay. The power mismatch has been found to be no more than 95 percent (Ref 8). The rate of temperature increase for this mismatch is 1.58°F/sec. Figure V-6 (Ref 8) shows the lag between the actual vessel outlet temperature and the measured value as a function of power. The lag at 95 percent mismatch is 14.3°F. Thus, the vessel outlet temperature at time of scram is 530°F + 3.6°F + 14.3°F = 548°F.

Assuming an initial vessel outlet temperature of 514°F (at time 115 percent power is reached) then the time of the "heat up" portion of the transient is

$$\frac{(548°\mathrm{F} - 514°\mathrm{F})}{1.58°\mathrm{F/sec}} = 21.5 \text{ sec}$$

It is conservatively assumed that the primary system pressure is 1352 psia (WCOB) at the moment the power level reaches the scram setpoint. The surge into the pressurizer is a linear function of the heat-up power, and is:

$$S = \frac{1.4 \times 10^{-5}\ \mathrm{ft^3/Btu}}{.0202\ \mathrm{ft^3/lb}}\ (.95 \times 45635)\ \frac{\mathrm{Btu}}{\mathrm{sec}} = 28.1\ \mathrm{lb/sec} = 1.01 \times 10^5\ \mathrm{lb/hr}$$

(This information is fed into the computer code TOPS (Ref 15)).

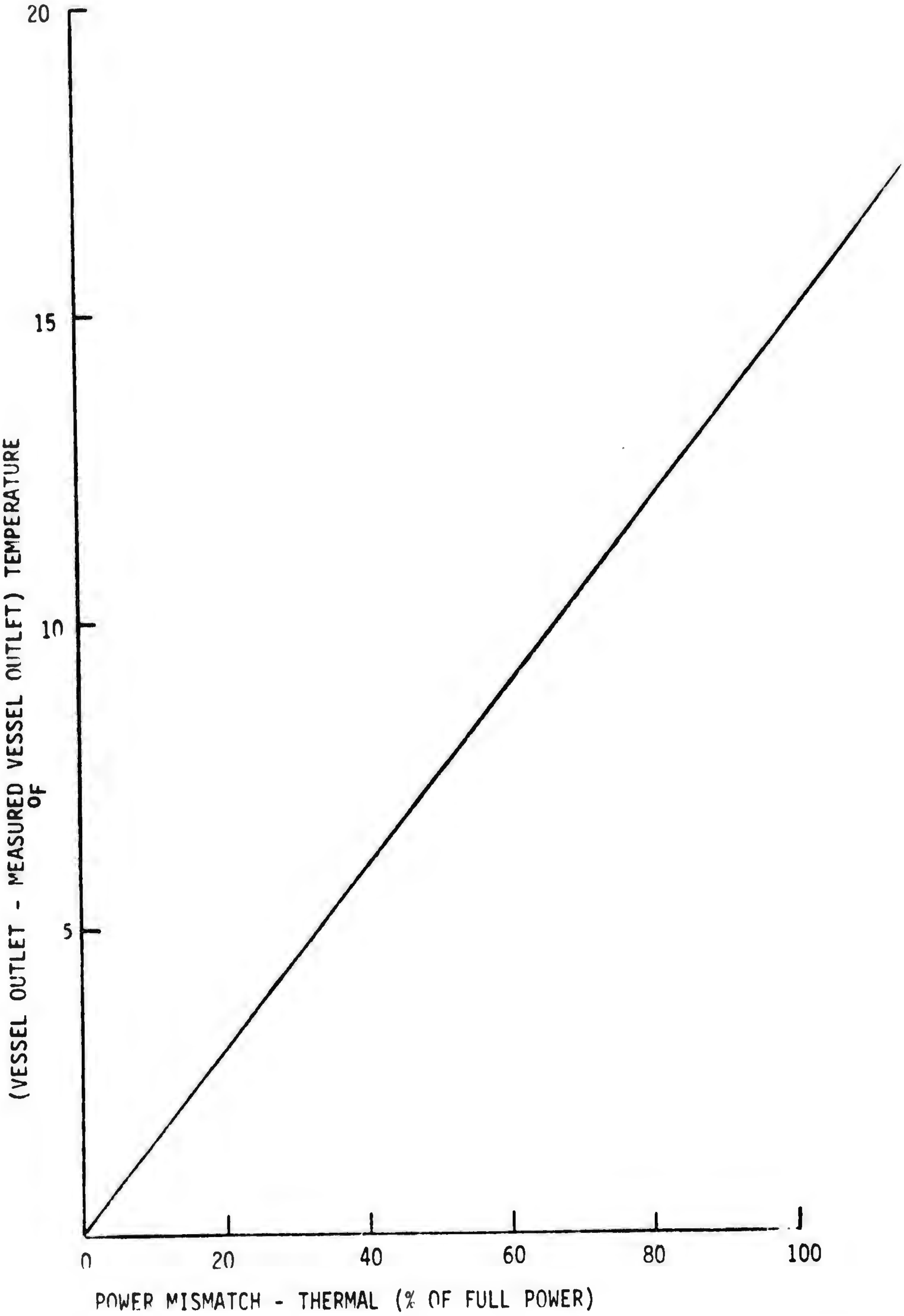


FIGURE V-6 TEMPERATURE OVERSHOOT VS. POWER MISMATCH DURING PRRWT

This code models the pressurizer of the primary loop. The physical dimensions, safety valve flows and setpoint, heater ratings and setpoints, spray flow rates and setpoints, and initial conditions (pressure, water level, etc), are input as parameters. The forcing function is the surge rate into (or out of) the pressurizer. The independent variables are pressure and temperature (both of course being intimately related).

The surge is calculated by the above equation (1). This equation has been programmed into DRAKE; and thus, the surge is available for transient modelled with this program.

The pressure transient output from TOPS (Ref 15) is dependent not only on surge rate and the associated natural thermodynamic compression-expansion actions, but also on heater, safety valve, and cooldown spray action, as well as heat transfer to the pressurizer wall. The resulting pressure from TOPS at the end of 20 seconds is 1431 psia. At this point the spray into the pressurizer slows the pressure surge to a large extent, and the pressure will remain at this value or slightly higher until the scram occurs.

The results of this transient are shown in Figure V-7. It will be noted that the RCT limits are within the safety limit envelope defined by the power to fuel melting and 2 percent quality at 1430 psia. Also, it can be seen that the primary system pressure is always above the pressure corresponding to 2 percent quality for the RCT.

C. Source Range Rod Withdrawal Transient

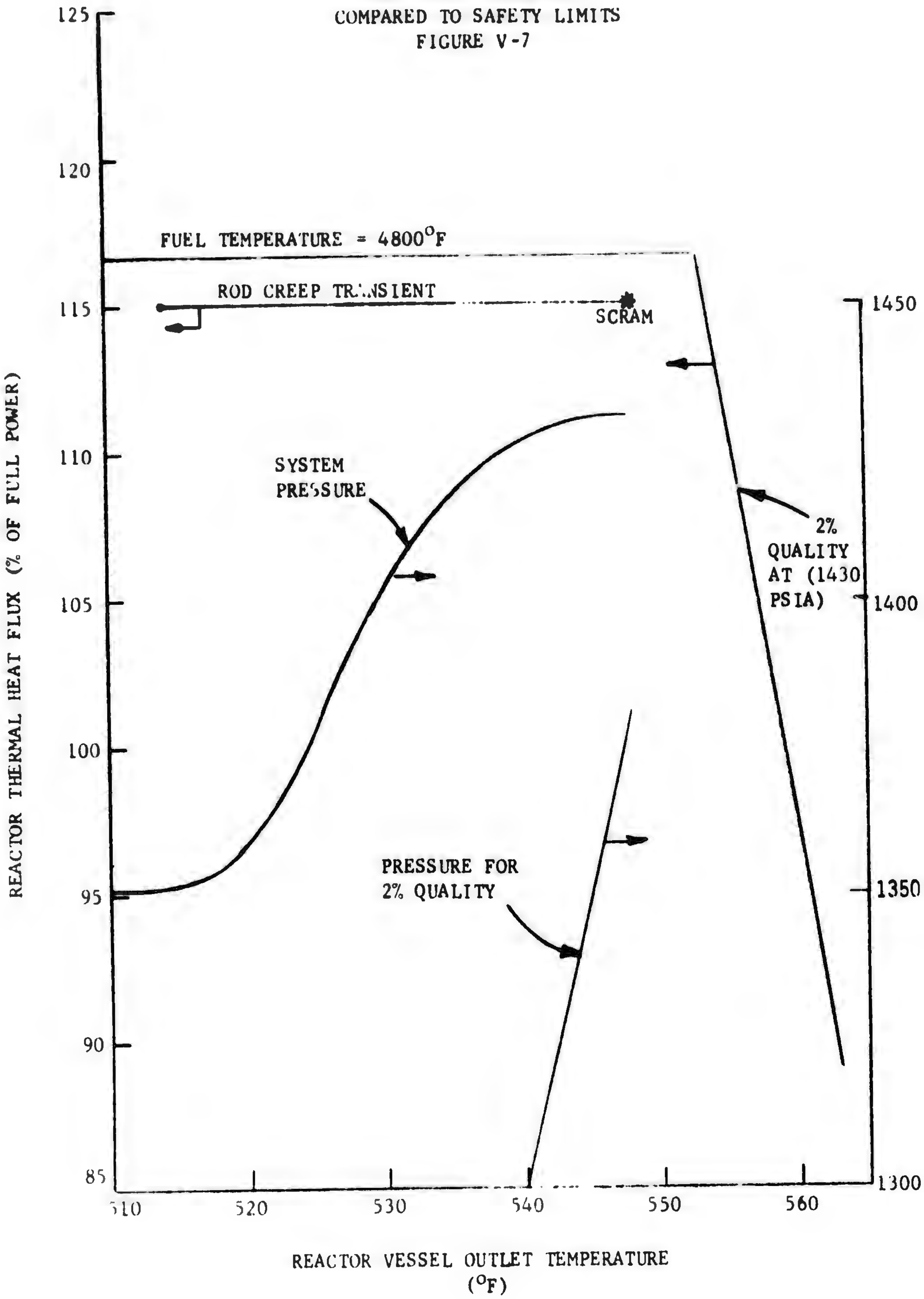
1. Background

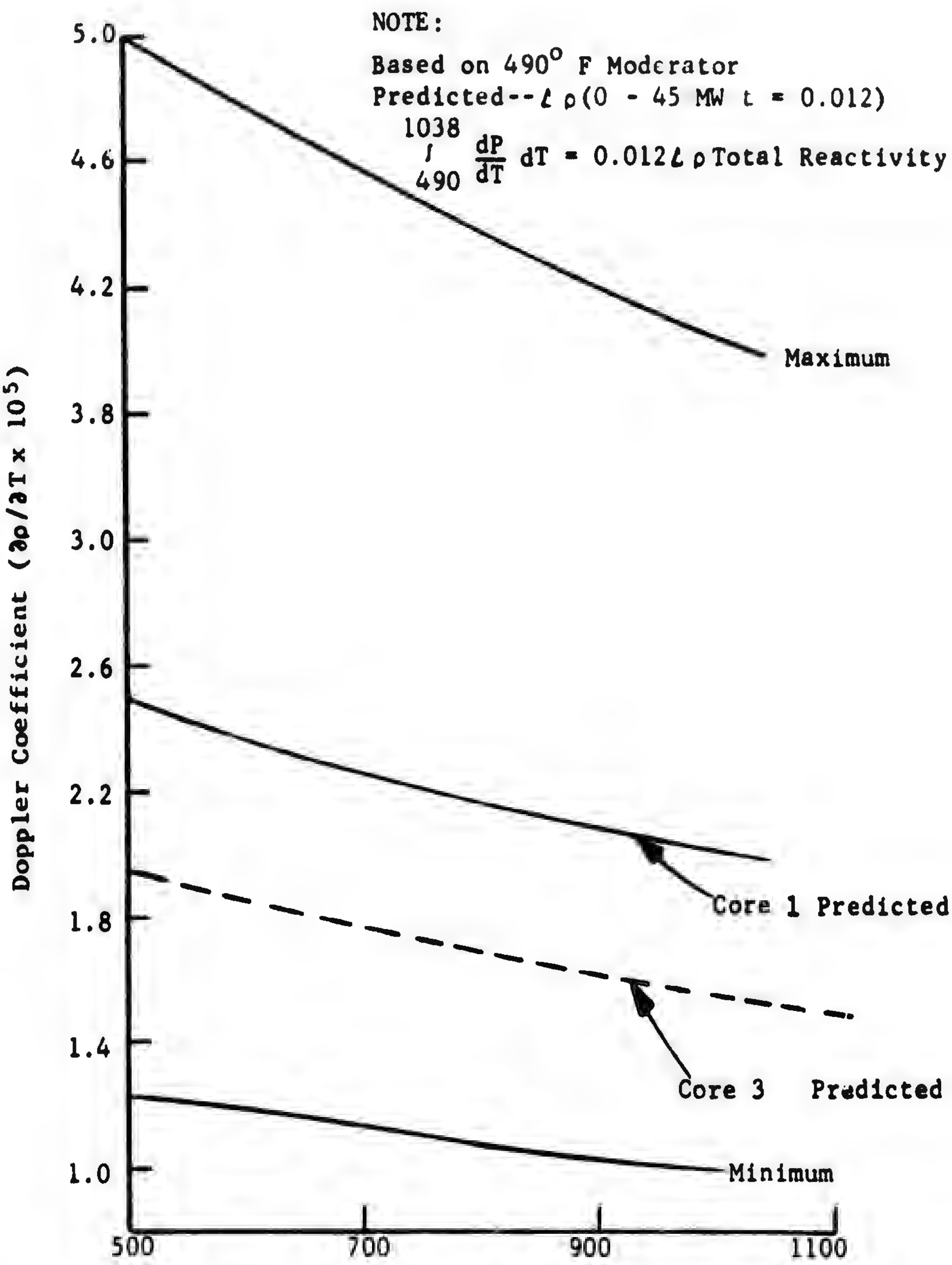
The original SAR for the MH-1A (Core 1) prepared by Martin (Ref 1) contains an extensive study of the SRRWT. The severity of the transient was found to increase with: (1) Increase in reactivity insertion rate, (2) increase in reactor subcriticality, and (3) decrease in the Doppler coefficient.

2. Results

The comparison of the Doppler coefficients for Core 1 and Core 3 is shown in Figure V-8. It may be seen that the minimum Doppler coefficient for Core 1 is approximately 65 percent of the Core 3 value. Using results corresponding to the minimum Core 1 Doppler coefficient would be conservative with respect to Core 3. The maximum reactivity insertion rate for Core 3 is about 20 percent below that used for Core 1 in this analysis

ROD CREEP TRANSIENT
COMPARED TO SAFETY LIMITS
FIGURE V-7





Range of UO_2 Doppler Coefficient with Average UO_2 Temperature

FIGURE V-8

Figures V-9 and V-10 show the effect of subcriticality and Doppler coefficient respectively as a function of the reactivity insertion rate on the severity of the transient. The worst case was found to be:

Hot - 490°F mean

Subcriticality - 20%

Reactivity insertion - 5.0×10^{-4} Δ P/sec

Doppler coefficient - 0.65 predicted value

For this case, the average fuel temperature was found to be less than 600°F, which is less than normal operating temperatures. For Core 3 the hot spot fuel temperature would be less than 960°F, thus, the SRRWT presents no danger to the core.

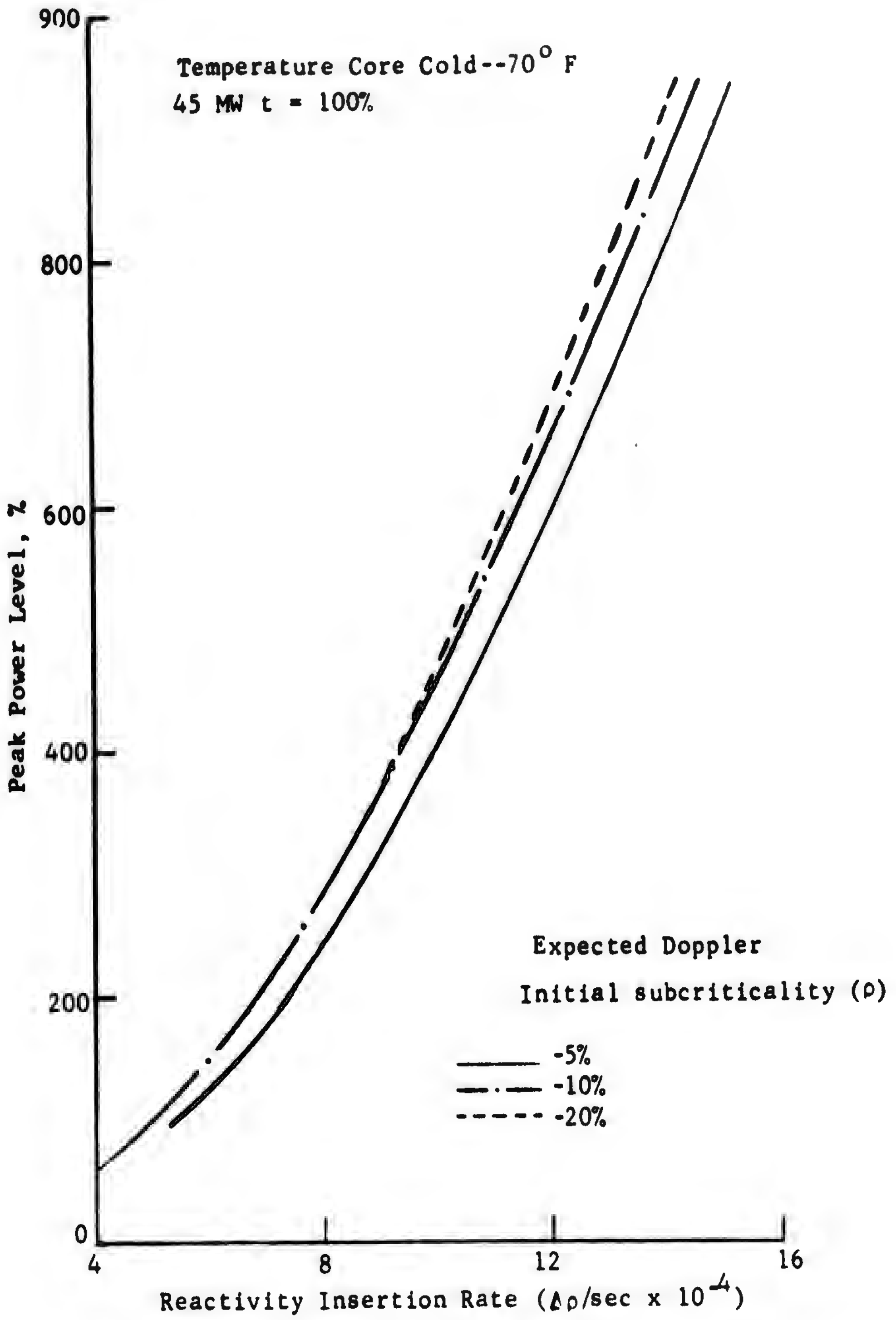
D. Main Steam Line Rupture Transient

1. Background

The integrity of the turbine cycle steam system affects the safety of the MH-1A reactor in that a rupture in this system can lead to an increase in the rate of heat transfer of the primary to the secondary system. A steam line rupture results in a sudden decrease in the secondary system temperature and pressure due to loss of water and steam through the break, with a resulting loss of energy from the secondary system. This transient is reflected in the primary system as a sudden increase in turbine load, causing a reduction in primary coolant temperature. The combination of the negative moderator temperature coefficient and the decrease in reactor inlet temperature results in a positive reactivity insertion rate, that is, a cold water transient. This transient results simultaneously in an increase in reactor power due to the positive reactivity insertion, and a decrease in primary system pressure due to the reduction in primary coolant temperature. The increase in reactor power forces an increase in fuel temperature which, via the negative fuel temperature coefficient (Doppler), results in a slowing down of reactor power. Unless terminated by a manual or automatic action, the increase in power level could result in one or more of the core safety criteria being violated. To prevent this possibility, the reactor safety system with its associated scram setpoints is designed to assure protection of the core with an adequate margin of safety.

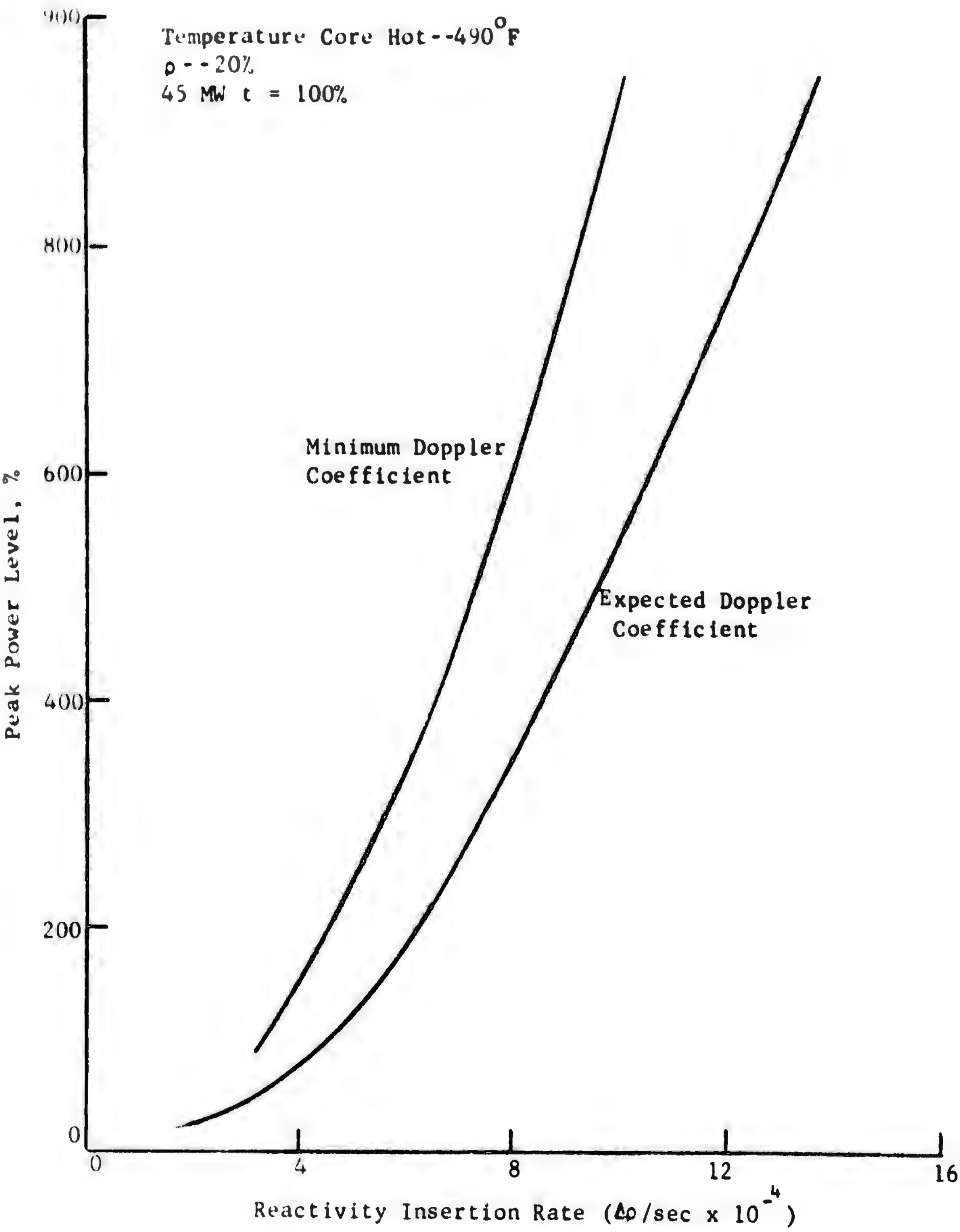
2. Analytical Methods

The methods used to analyze the results of the main steamline rupture transients were embodied in the following computer programs: COUGAR, DRAKE and CHIC-KIN.



Effect of Initial Subcriticality on Peak Power Level During Rod Withdrawal Accident

FIGURE V-9



Doppler Reactivity Effects on Power Levels During Rod Withdrawal Accident

FIGURE V-10

COUGAR is a digital computer program which describes the time dependant secondary system temperature and pressure by calculating mass and energy balances during the blowdown of the secondary side contents following the rupture of the main steam line. In previous analyses of the MH-1A main steam line rupture transient, COUGAR has shown the secondary side temperature to be of first order lag during blowdown with a time constant equal to 3 seconds (Ref 8).

The remaining programs (DRAKE and CHIC-KIN) are discussed in previous sections of this report.

The overall approach to analyze the main steam line rupture transient consisted of using the reactor plant transient code DRAKE to create the time dependant steam generator secondary temperature based upon COUGAR's 3-second time constant. Using this as secondary input, DRAKE then provided reactor power as a function of time. This power trace, in turn, was used in CHIC-KIN to determine the hot channel conditions concerning enthalpy rise, minimum DNB ratio, and maximum fuel temperature.

3. Input Parameters

(a) Nuclear Parameters

The nuclear kinetic parameters employed in the power range rod withdrawal transient described previously were used in the analysis of the main steam line rupture transient. Again, the reactivity insertion rate, upon plant scram, was provided by a nonlinear function generation capability of DRAKE (see Fig V-11).

(b) Thermal and Hydraulic Input Parameters

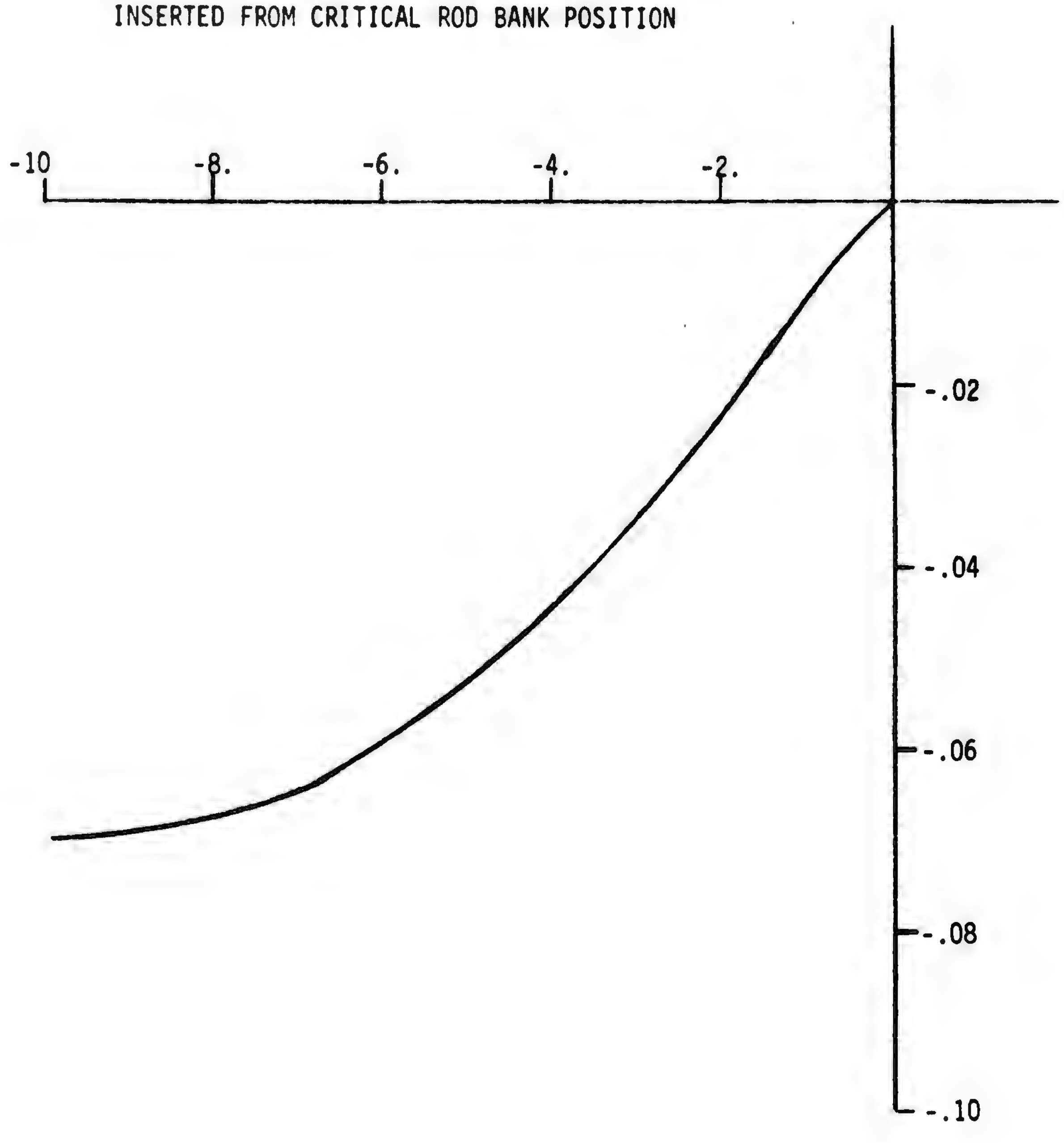
As in the case of the power range rod withdrawal transient, the main steam line rupture transients were analyzed from the worst case operating conditions for coolant temperature and coolant flow rate. Since the primary coolant pressure decreases during the transient, it may be reasoned that the pressurizer heaters may not be able to maintain the pressure within the operating band. Thus, the pressure of 1265 psia (low pressure scram point with associated error) was conservatively selected for the analysis.

The hot channel factor of 1.65, the hot spot factor of 2.20 applied to heat flux, and the hot spot factor of 2.07 applied to fuel temperatures remain unchanged from those developed in the analysis of the power range rod withdrawal transient.

FIGURE V-11

PFACTIVITY ($\Delta k/_k$) INSERPTED VS. DISTANCE

INSERTED FROM CRITICAL ROD BANK POSITION



(c) Initial Conditions

Previous analyses of an MH-1A main steam line rupture transient (Ref 8) confirm worst case initial operating conditions to be those of low system pressure, low primary coolant flow, and high reactor power level. These initial conditions consistently produced worst case results with regard to DNB ratio and hot spot fuel temperature. The initial conditions for the cases analyzed are given in Table V-3.

TABLE V-3

MSLRT INITIAL CONDITIONS

Case No.	Inlet Primary Coolant Temp. (°F)	Initial Reactor Power Level (%)	Primary Coolant Flow (gpm)	Primary System Pressure (psia)
1	483	107.	10,200	1265
2	522	.1	10,200	1265

Case No. 2 was used to again verify high initial reactor power level as worst case.

4. Results

The response of the MH-1A reactor to main steam line rupture transients initiated from power levels of 107 and 0.1 percent of full power are shown in Figures V-12 and V-13 respectively. It can be seen from Figure V-11 that the transient initiated from 107 percent power is a relatively slow transient. The positive reactivity inserted by the decreasing reactor inlet temperature causes the reactor to go on a positive period. The power level rises, causing an increase in heat flux and fuel temperature. The heat flux lags the power level because of the relatively long fuel time constant associated with the oxide fuel and helium gap of the MH-1A fuel pins. Negative Doppler reactivity feedback, due to the increase in fuel temperature, partially compensates for the positive reactivity due to the decreasing reactor inlet temperature.

The transient was terminated by a high power scram setpoint of 110 percent. The maximum power achieved is 118.5 percent, the maximum core thermal flux is 110.4 percent, and the maximum hot spot fuel temperature is 4396°F. The scram setpoint was reached at time equal to 2.43 seconds and the rods began to drop at time equal to 2.66 seconds after the initiation of the transient.

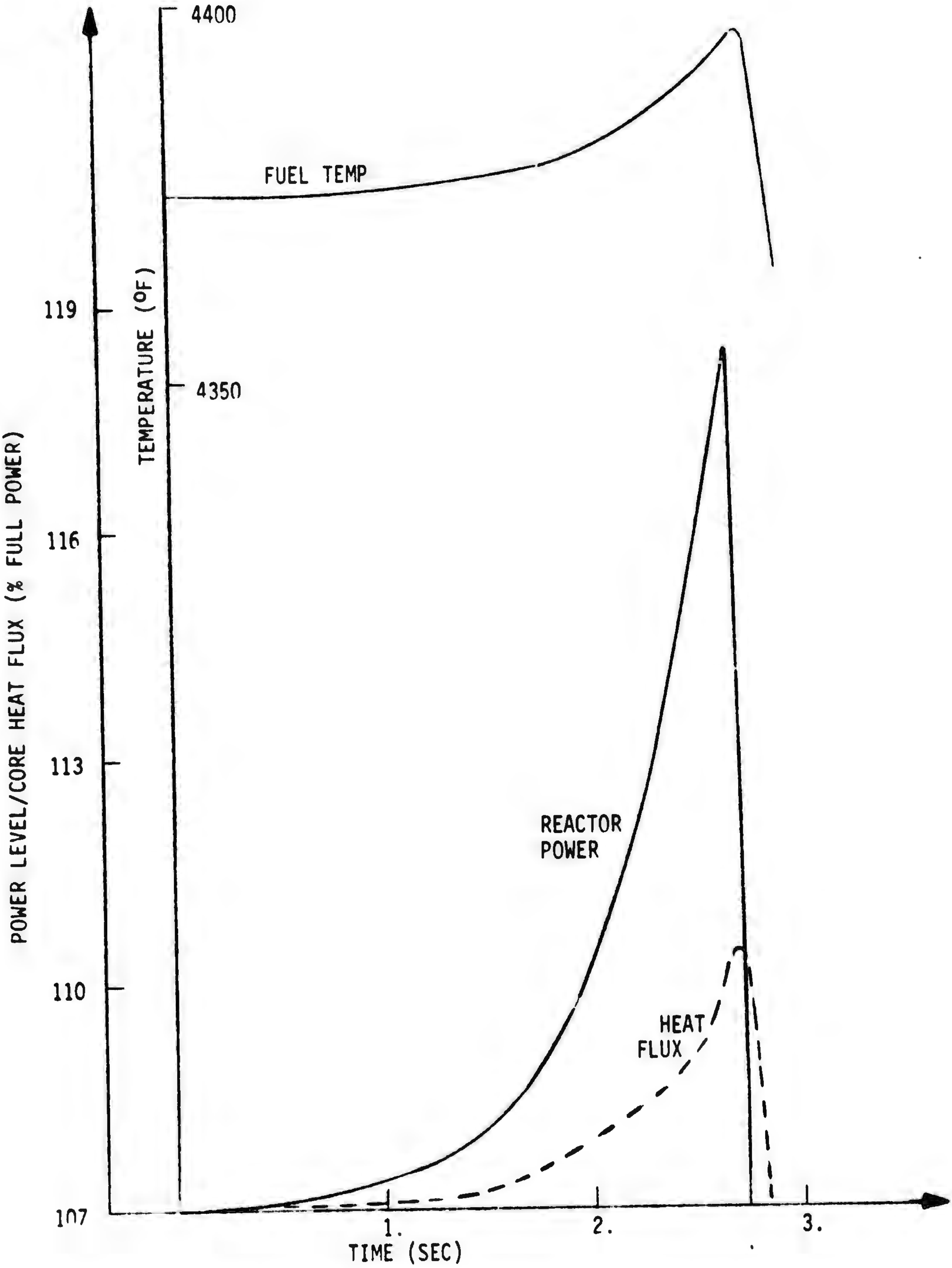


FIGURE V-12 MSLRT REACTOR POWER, HOT SPOT FUEL TEMPERATURE AND CORE THERMAL POWER VS. TIME FOR INITIAL POWER LEVEL 107%

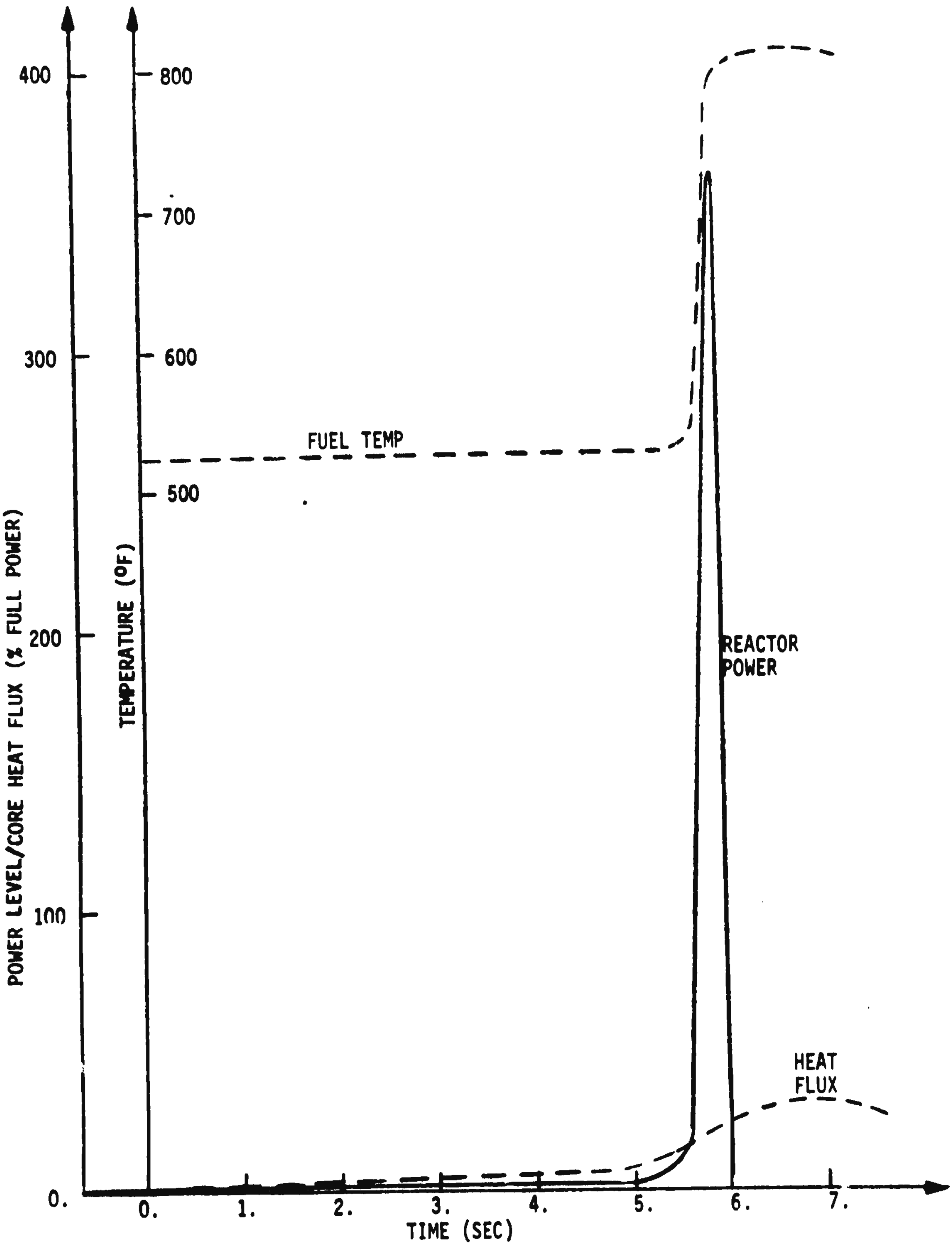


FIGURE V-13 MSLRT REACTOR POWER, HOT SPOT FUEL TEMPERATURE AND CORE THERMAL POWER VS. TIME FOR INITIAL POWER LEVEL .1%

Figure V-13 shows the response to the main steam line rupture transient for an initial power level of 0.1 percent. Due to the low initial power level, the initial rise in power does not result in an increase in fuel temperature. Thus, there is no significant reactivity feedback. The power level thus rises on a shorter period compared with the case of initial power level of 107 percent. After the power level has risen above approximately 10 percent, the fuel temperature increases and adds negative reactivity via the fuel Doppler effect.

The scram setpoint is reached at time equal to 5.66 seconds and the control rods drop at time equal to 5.89 seconds. The maximum power achieved is 370 percent, the maximum core thermal flux is 33 percent, and the maximum hot spot fuel temperature is 820°F.

Figure V-14 shows core thermal power vs. reactor outlet temperature for the 107 percent initial power level main steam line rupture transient. Again, the framework for all such transients is the set of safety limits defined in the steady-state analysis, (Sec III) of this report, with the limiting conditions of 2 percent quality and 4800°F fuel temperature. The minimum DNB ratio of 1.3 would be violated subsequent to those of quality and fuel temperature, and is then of no concern. Figure V-14 verifies all safety limits held safe.

For the case of a main steam line rupture transient initiated from a power level of 0.1 percent, Figure V-15 reflects the fact that even wider margins exist to insure the integrity of the safety limits than with case #1.

Thus, the limiting combinations of 2 percent quality and maximum fuel temperature (4800°F) are verified held safe for the most severe main steam line rupture transients. These safety limits then assure safe operation of the MH-1A, during such a transient, with adequate margin.

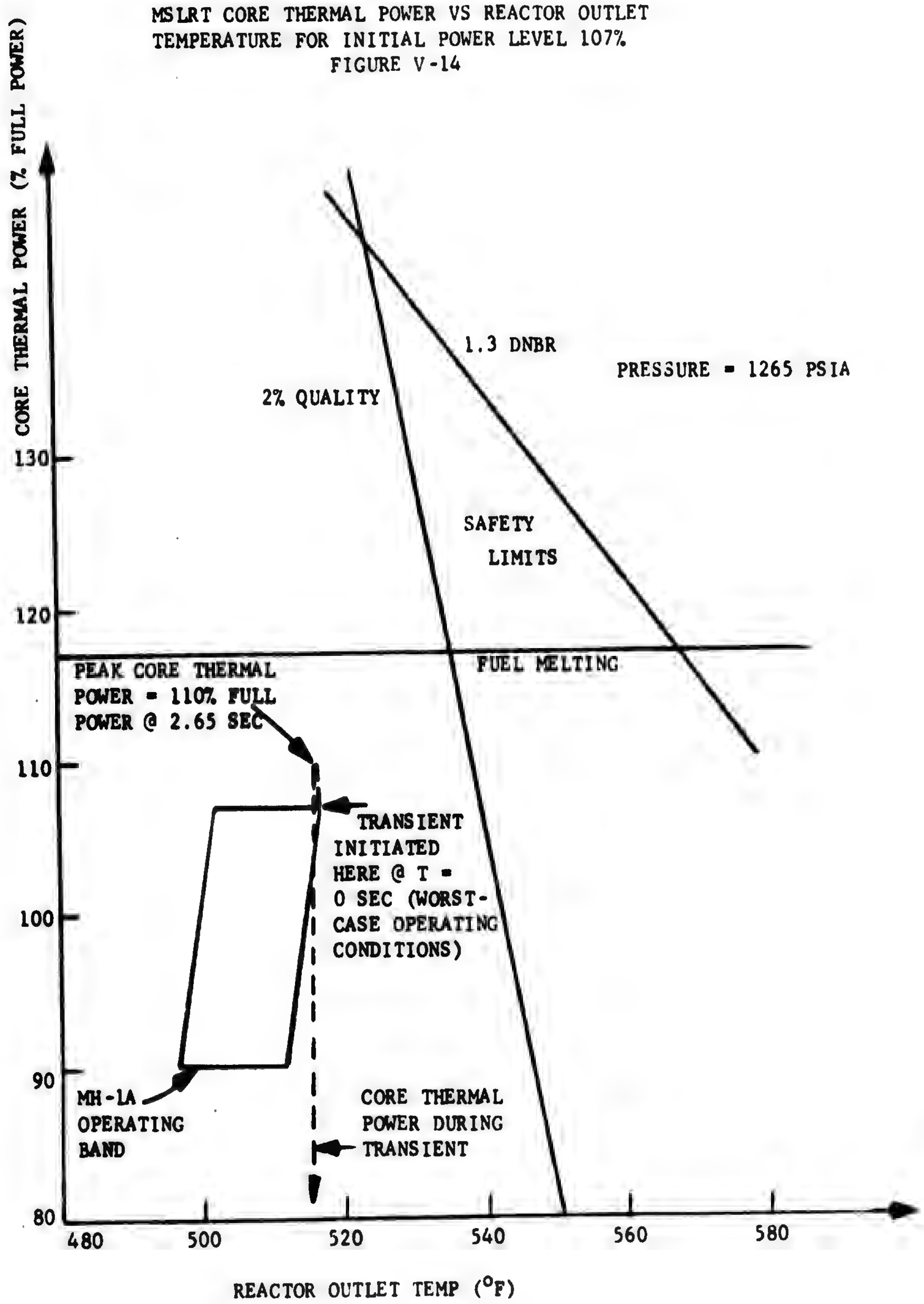
E. Loss of Flow Transient

1. Background

This transient (abbreviated LOFT) is initiated by the loss of power to one or both of the primary coolant pumps, or by a mechanical failure of either or both pumps. It has been shown that the loss of service of both pumps result is the most severe transient (Ref 8).

MSLRT CORE THERMAL POWER VS REACTOR OUTLET TEMPERATURE FOR INITIAL POWER LEVEL 107%

FIGURE V-14



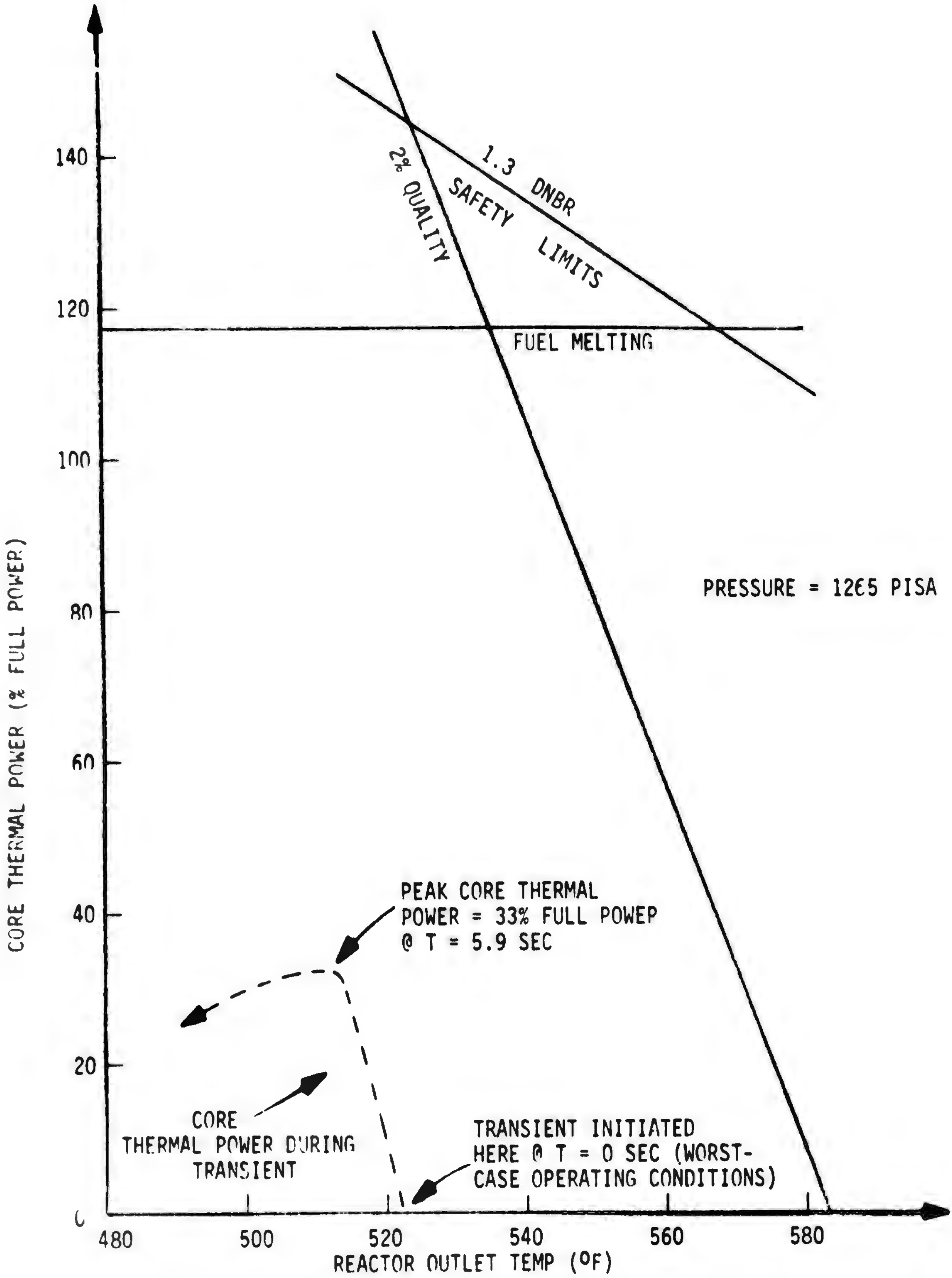


FIGURE V-15 MSLBT CORE THERMAL POWER VS. REACTOR OUTLET TEMPERATURE FOR INITIAL POWER LEVEL .1%

2. Analytical Method

To simulate the transient, the flow coastdown curve calculated in Ref 16 and shown in Figure V-16 was used as input to a variable-flow version of DRAKE. In this version all flow dependent parameters are placed in the "transient" section of the program in addition to including them in the "initial conditions" section. The DRAKE model then calculated the resulting power transient which was input to CHIC-KIN. The power and heat flux traces from DRAKE is shown in Figure V-17.

3. Initial Conditions

The parameters and initial conditions for this DRAKE run are shown in Table V-4.

TABLE V-4
DRAKE LOFT INPUT

Initial power (%)	107
Initial flow (gpm)	10,200
Low flow scram (gpm)	8,900
Scram delay (sec)	.5
Average primary temp (°F)	497

4. Results

Using the power trace from DRAKE, a primary system pressure of 1352 psia, and the Δ P vs time curve (Fig V-18) from Ref 16 - a hot channel analysis was made using the CHIC-KIN code. The results of this analysis are shown graphically in Fig V-19 and V-20. The hot channel inlet mass velocity as a function of time is shown in Fig. V-19. It will be noticed that it is characterized by oscillations of increasing period during the 9 seconds of the analysis. The oscillations are caused by boiling in the hot channel. A similar effect is noticed for the exit mass velocity as seen in Fig. V-20. At the end of the analysis (approximately 9 seconds) it can be seen that the nominal flow from Fig. V-11 is beginning to reach a limiting value due to natural convection while the heat flux is continually dying off as shown in Fig. V-18. Thus, the boiling will begin to subside and the flow oscillations will subside.

The hot channel exit quality is shown for the LOFT in Fig. V-21. As in the case of the mass velocity, there are oscillations. The peak quality reached is 18 percent as shown. For the same reasons as the flow oscillations, this quality will gradually decrease as the transient progresses.

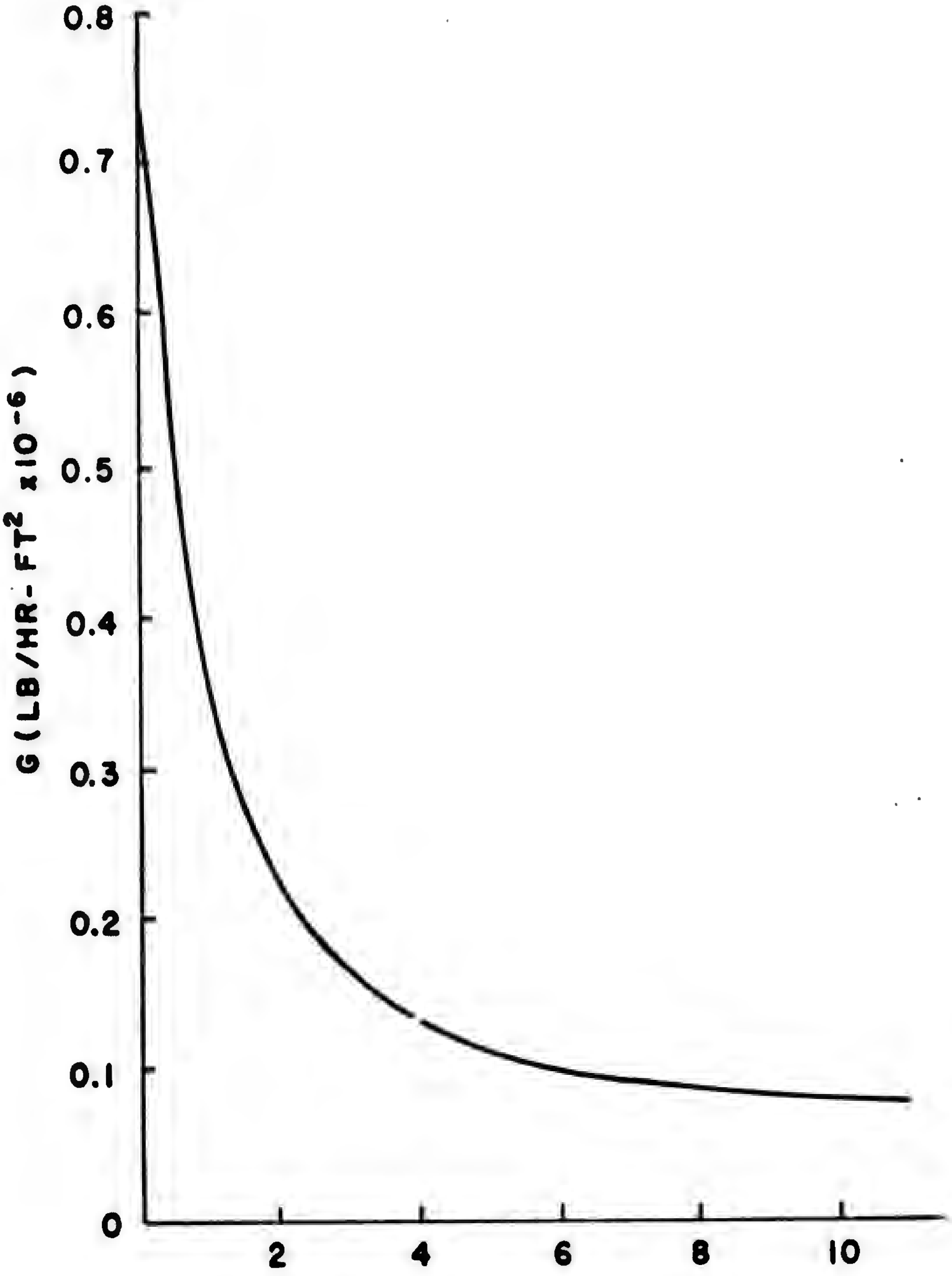


FIGURE V-16, LOFT NOMINAL CHANNEL INLET MASS VELOCITY VS. TIME

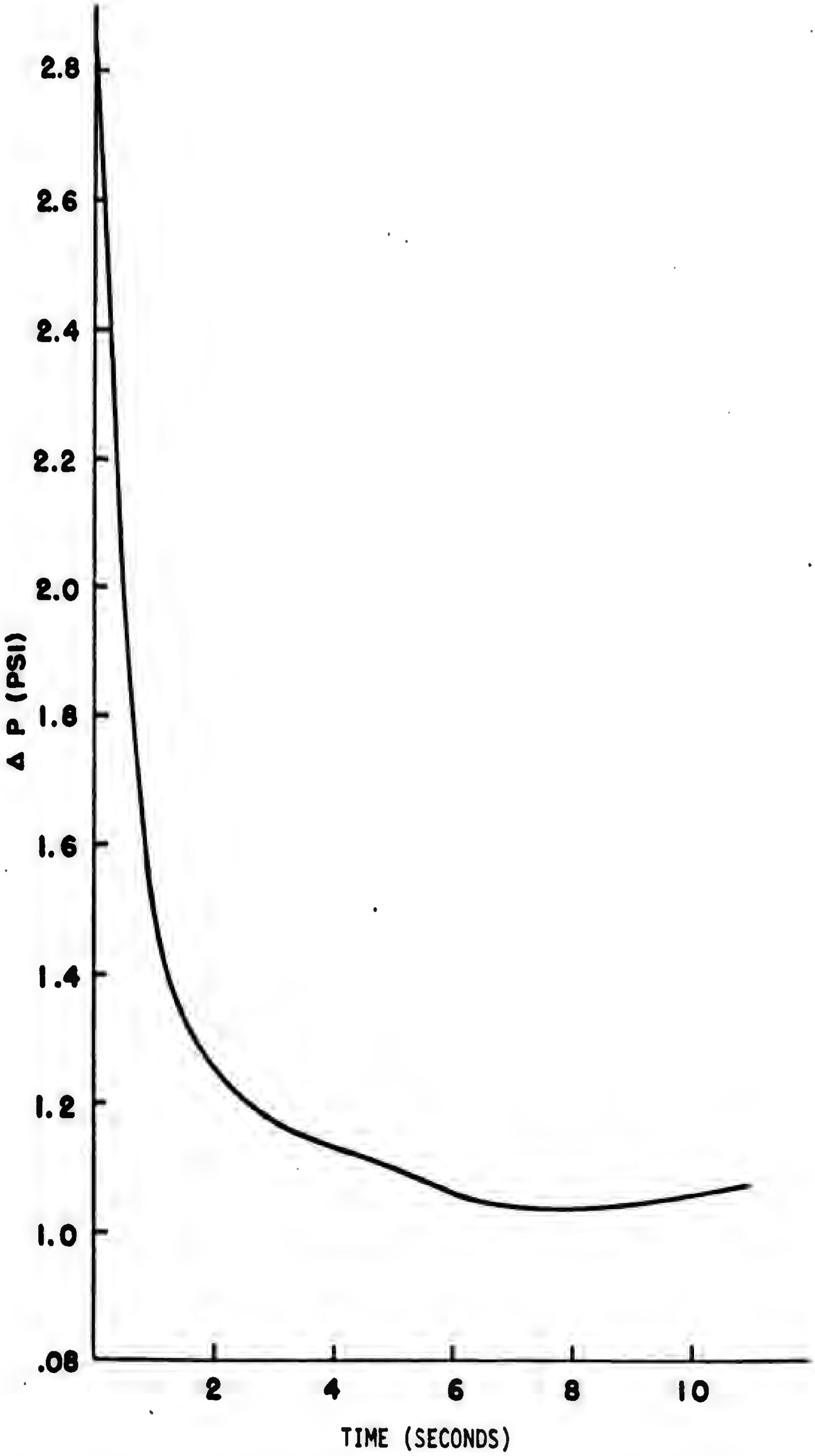


FIGURE V-17, LOFT PLENUM TO PLENUM PRESSURE DROP VS. TIME INCLUDING ELEVATION HEAD

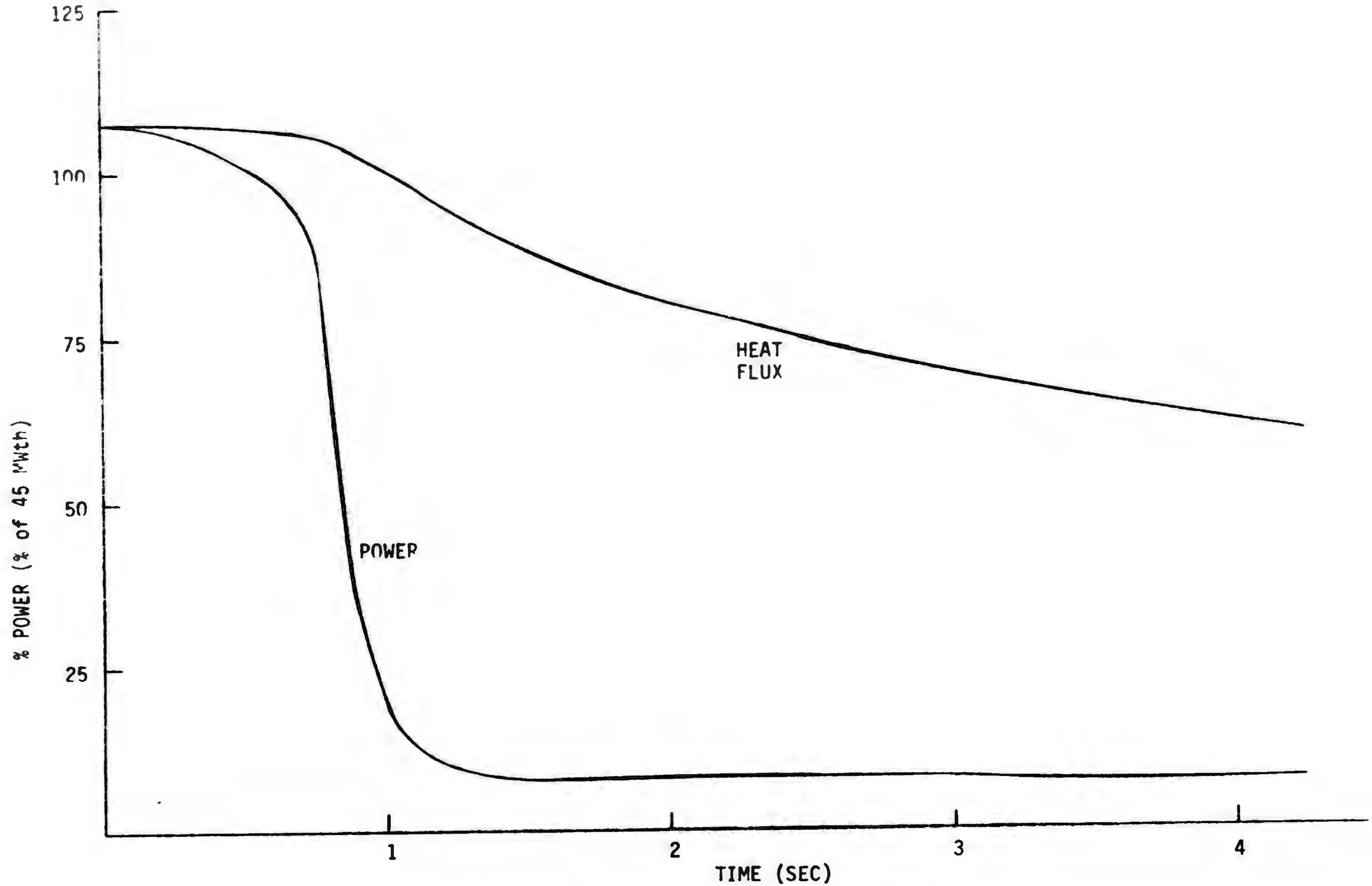


FIGURE V-18 POWER TRANSIENT RESULTING FROM LOFT

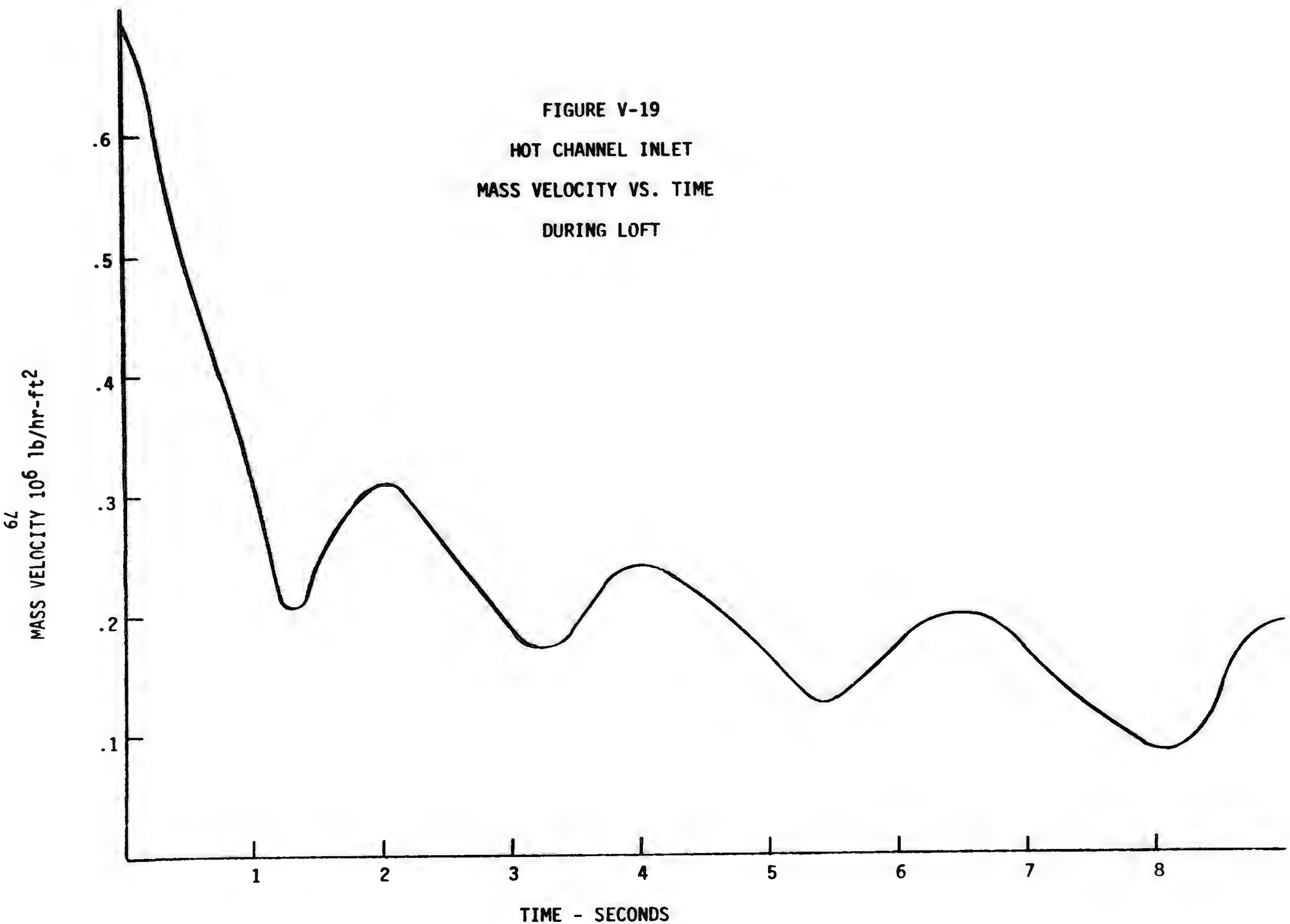


FIGURE V-19
HOT CHANNEL INLET
MASS VELOCITY VS. TIME
DURING LOFT

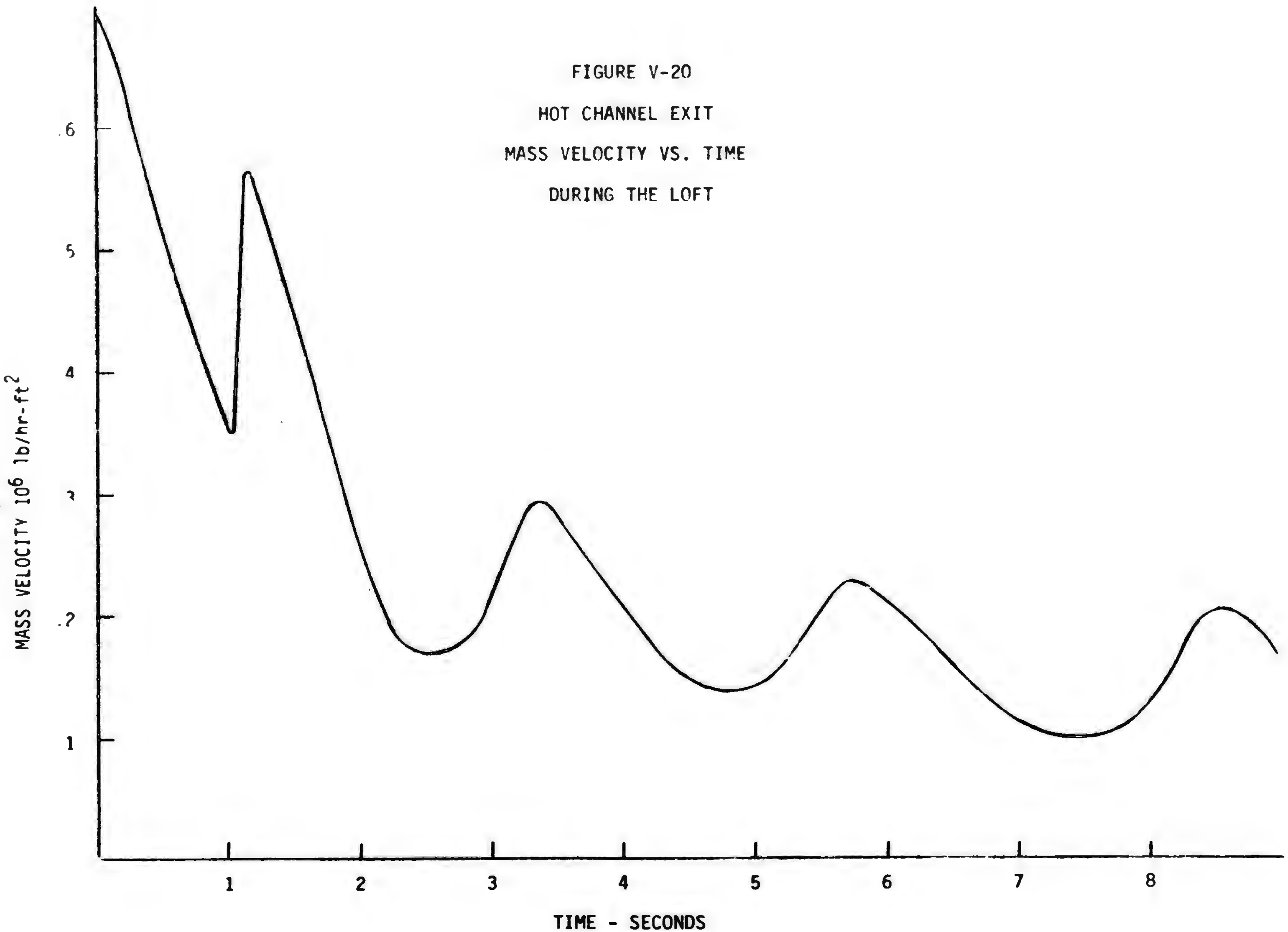


FIGURE V-20

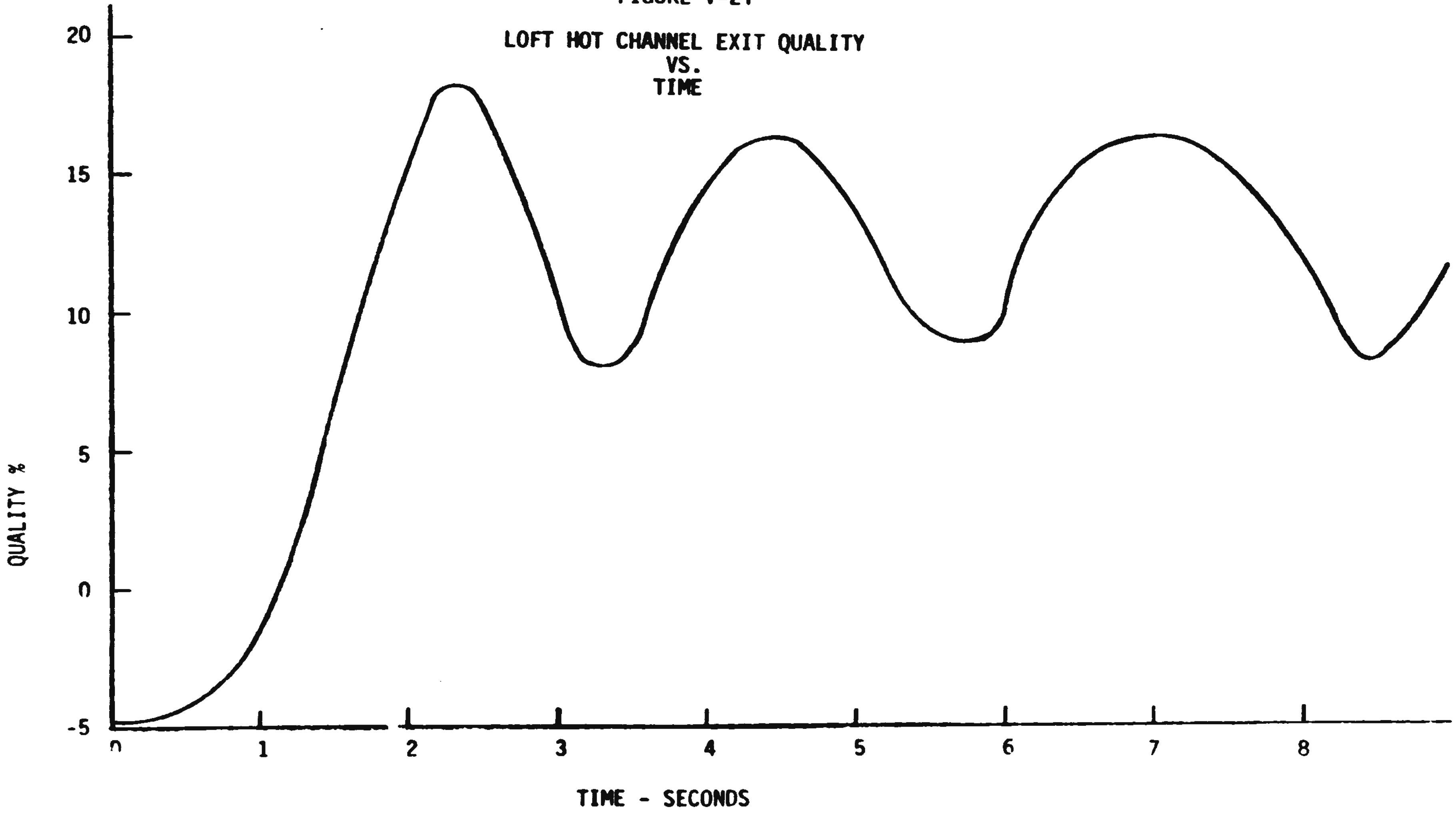
HOT CHANNEL EXIT

MASS VELOCITY VS. TIME

DURING THE LOFT

FIGURE V-21

LOFT HOT CHANNEL EXIT QUALITY
VS.
TIME



While the quality shown in Fig. V-21 is high, it should be remembered that this appears at the point where the heat flux is on the order of 20 percent of the average hot channel heat flux. In the vicinity of the maximum heat flux (about twice the average hot channel heat flux) the quality always stays negative. It is in this region that burnout is most likely to occur.

Figure V-22 shows the minimum DNBR during the initial phases of the transient. This curve is based on the maximum heat flux for the LOFT and the critical heat flux versus flow steady-state parametric for 107 percent power (see Fig V-23). As this critical heat flux is from a steady-state analysis, it is conservative to apply it here. It can be seen that the minimum DNB ratio is 1.5, above the 1.3 value where DNB may occur.

The safety criterion of 2 percent quality is obviously violated during the LOFT. However, in this case as the heat flux is continually dying, and there are no hydraulic forcing functions (i.e. pump input) to cause instability, the magnitude of quality is of no consequence to fuel integrity and therefore, plant safety.

It is a proposed Technical Specification requirement that the control rod motion must begin within 500 ms of pump failure. A study at the MH-1A (Ref 19 (DF)) has shown the actual value to be less than 400 ms. The LOFT study assumed a 500 ms scram delay with a 210 ms coastdown to scram setpoint for a total time of 710 ms before rod motion. Thus, if the scram setpoint were at the plant, there is still ample safety margin.

It is concluded that the MH-1A is adequately protected from the loss of flow transient by a scram setpoint of 8900 gpm. The scram in this analysis begins at 710 ms, leaving more than 300 ms margin from the actual value to account for error in the flow measurement and setpoint. For example, if the scram setpoint were in actuality at 8000 gpm rather than 8900 gpm, the additional delay would be about 100 ms - well within the greater than 300 ms margin.

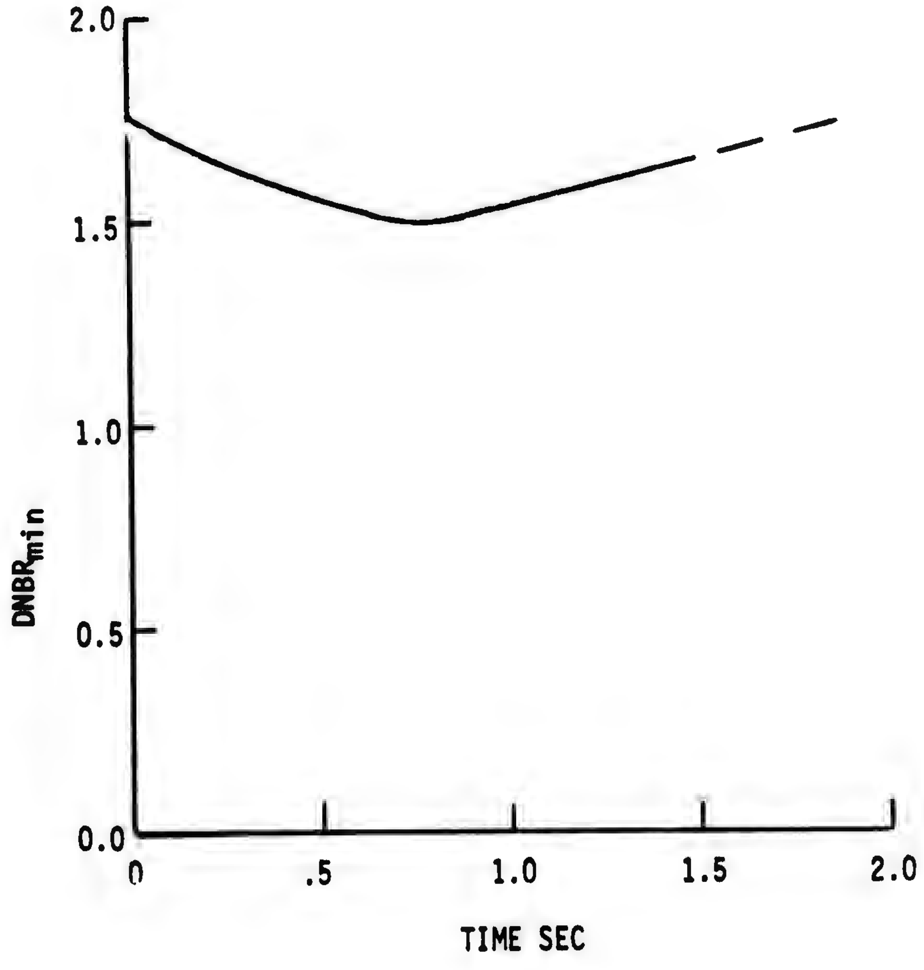
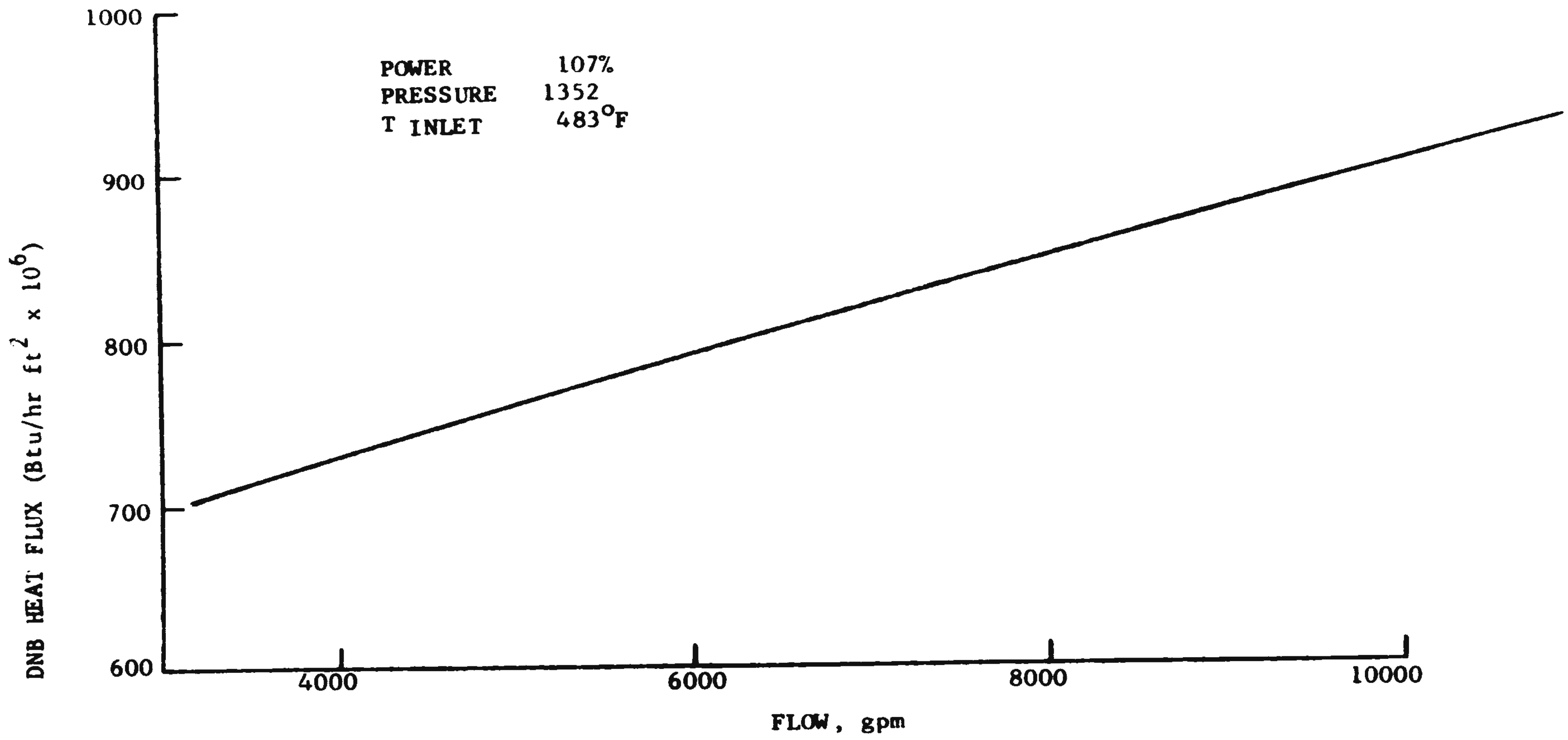


FIGURE V-22

DNBR$_{min}$ VS. TIME
FOR
MH-1A LOFT

DNB HEAT FLUX VS. COOLANT FLOW
AT STEADY STATE CONDITIONS (W-3 CORRELATION)
FIGURE V-23



VI. TRANSIENTS INITIATED FROM DESIGN BAND LIMITS AND SCOPING ANALYSES

This section presents the effects of transients initiated from the limits of the design band (alarm points) for each of three postulated transients: Power range rod withdrawal, main steam line rupture and loss of flow. Alarm points are those values of reactor power and the primary loop parameters which trigger an operator alarm and require corrective action. That is, the operating band must now be expanded as the design band to include these alarm points, and the results of transients initiated under these conditions examined.

Scoping provides information concerning minimum DNB ratios, maximum hot spot fuel temperatures, and qualitites for the above three transients when initiation occurs with all reactor parameters at their respective scram setpoints or worst case values. Table VI-1 shows nominal, alarm point, and scram settings for reactor power, reactor outlet temperature, primary coolant flow rate, and pressure.

TABLE VI-1

MH-1A PRIMARY LOOP PARAMETERS

Case	Reactor Power (% full power)	Outlet Temp (°F)	Flow (gpm)	Pressure (psia)
Nominal	100%	506	10800	1400
Alarm ± Error	111%	534	9750	1352
Scram Set Point ± Error	115%	534	8650	1265

A. Power Range Rod Withdrawal

This transient was analyzed in detail in Section V of this report. All reactor kinetics input parameters and the analytical methods remain unchanged.

1. The Design Band

(a) Initial Conditions

Power range rod withdrawal transients initiated from the limits of the design band (alarm points) incorporate a primary coolant flow rate of 9750 gpm, a reactor outlet temperature of 534°F, and a primary system pressure of 1352 psia. The fastest reactivity insertion rate caused by the maximum rod withdrawal rate of 2 inches/min was selected as the extreme case. Previous analysis shows slow reactivity insertion PRRWT's to be less than worst case rod creep transients. That is, quasi-steady-state behavior, wherein it is assumed coolant temperature remains at its highest value during the transient, and reactor power remains at the scram setpoint for a significant length of time. Rod creep transients are discussed in the following section.

Table VI-2 shows the three power levels chosen for analysis and their significance.

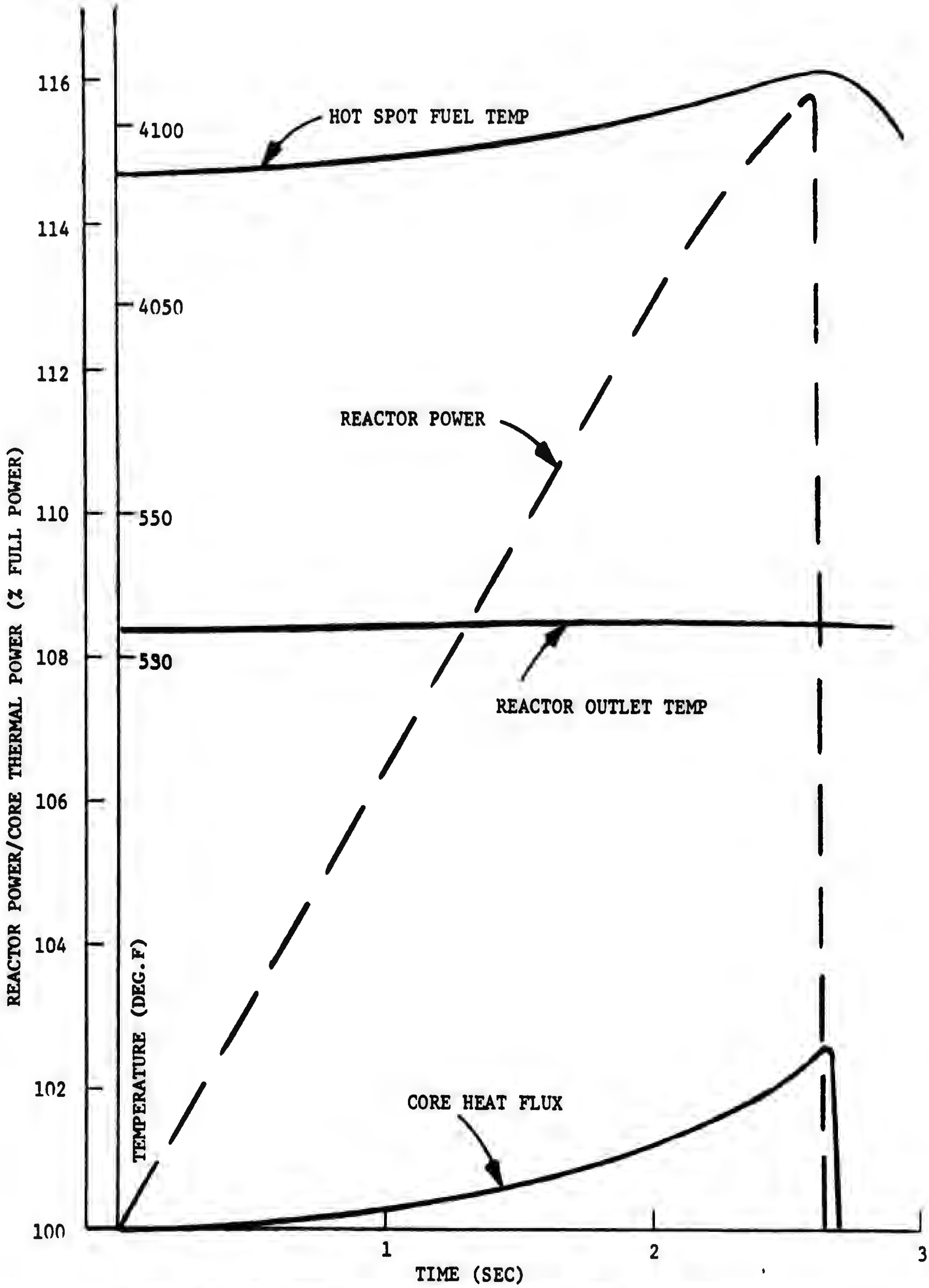
TABLE VI-2

DESIGN BAND RUN INITIAL POWER LEVELS

POWER LEVEL (% full power)	SIGNIFICANCE
100%	Nominal value
107%	Operating band limit
111%	Design band limit

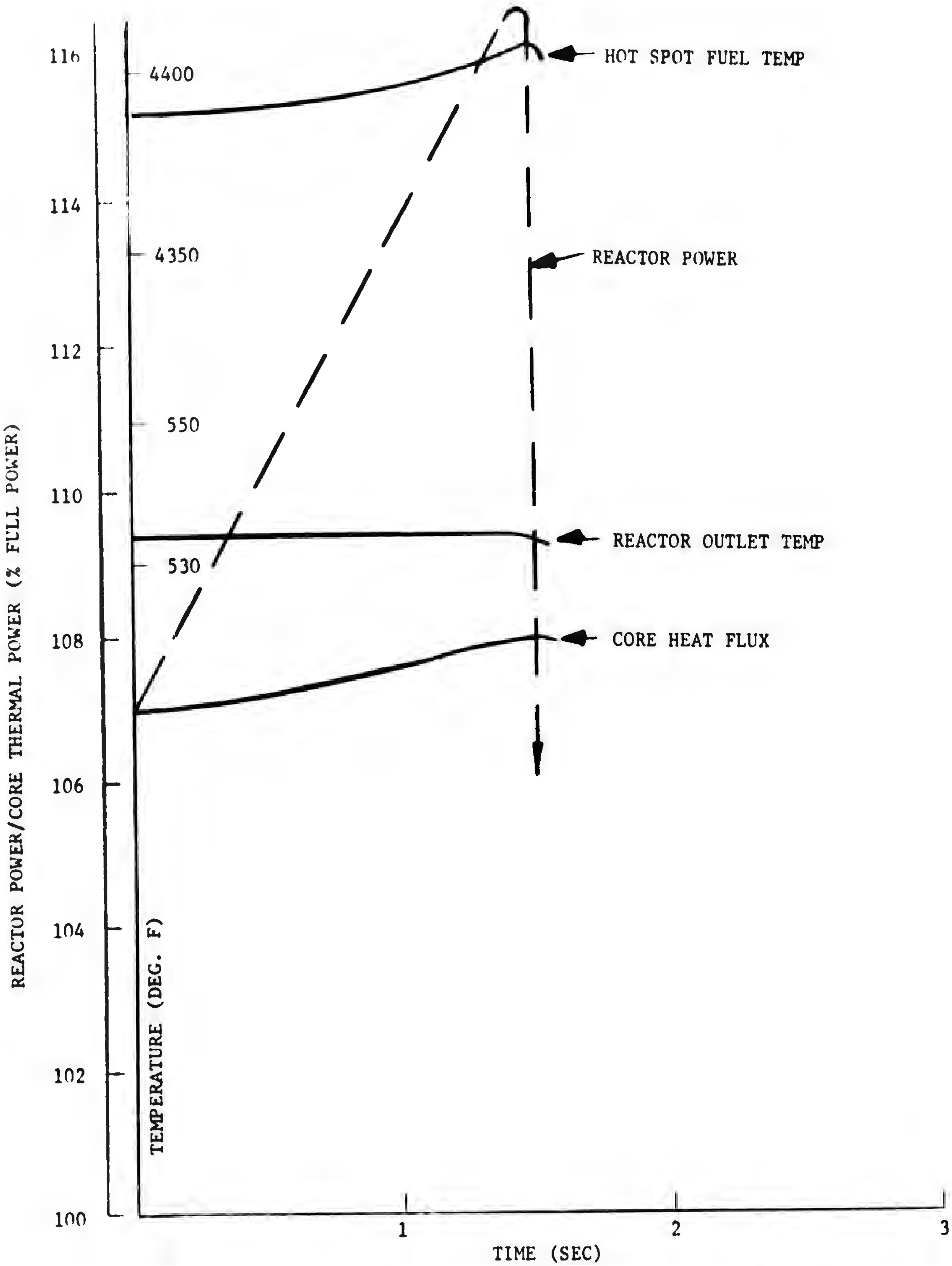
(b) Results

Figures VI-1, VI-2 and VI-3 show the transient behavior of the MH-1A for the three prescribed sets of initial conditions. Instantaneous neutron power, core heat flux, hot spot fuel temperature, and the reactor coolant outlet temperature are displayed as functions of time. Figure VI-3 show the maximum core heat flux achieved during this worst case to be 111.8 percent full power. The hot spot fuel temperature reaches a peak of 4564°F, far below the irradiated UO_2 melting point of 4800°F. Figure VI-4 shows core thermal power vs. reactor outlet temperature for this transient. Since the framework for all such transients is the set of safety limits defined in Section IV of this report, the limiting conditions here are 2 percent quality and 4800°F fuel temperature. The minimum DNB ratio safety limit of 1.3 would be violated subsequent to those of quality and fuel temperature, and thus is of no concern. Figure VI-4 verifies these limiting conditions are held safe for this most severe power range rod withdrawal transient.



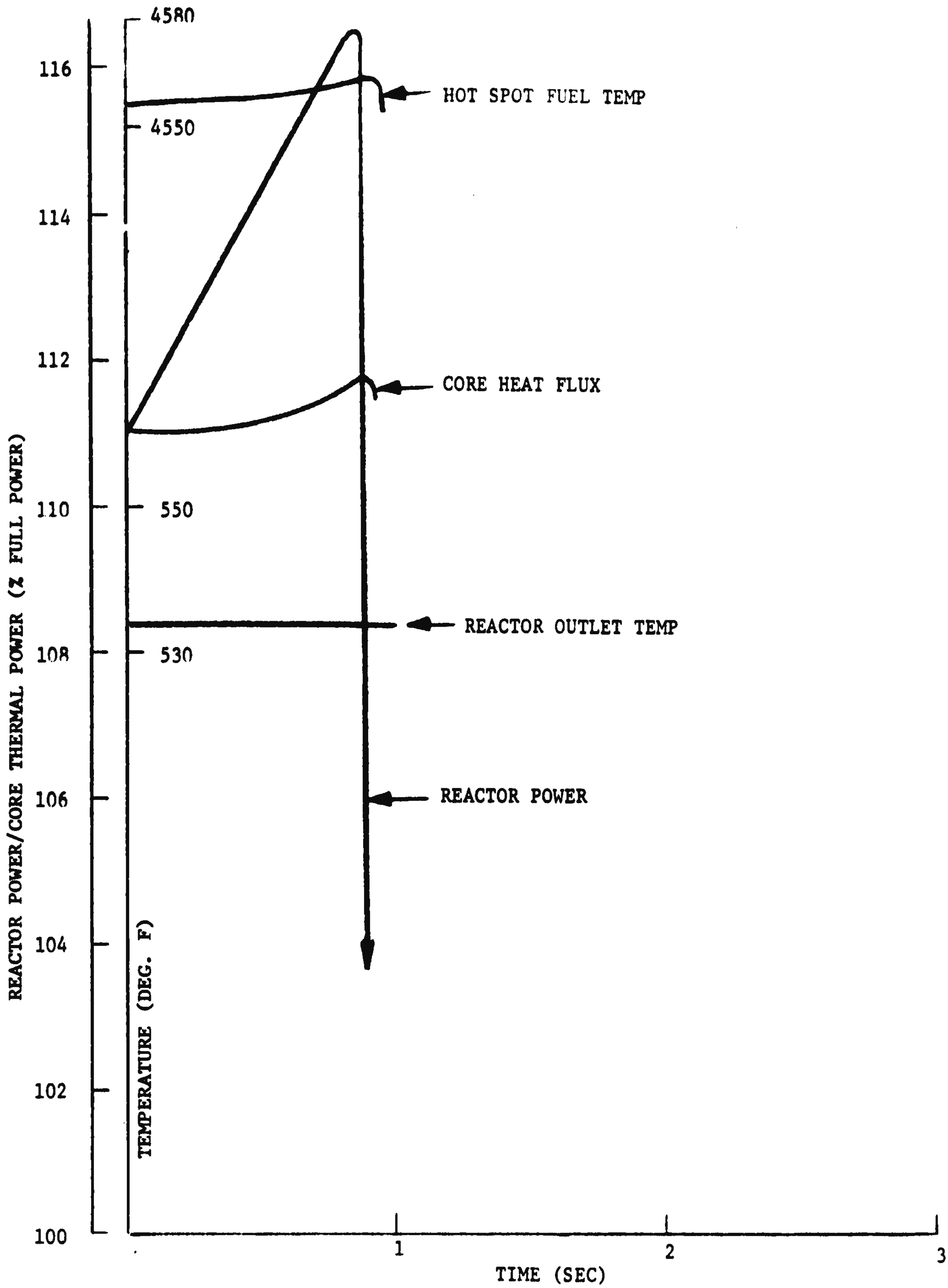
PRRWT REACTOR POWER, CORE THERMAL POWER, HOT SPOT FUEL TEMP & REACTOR OUTLET COOLANT TEMP VS TIME FOR INITIAL POWER LEVEL 100%, 2"/MIN ROD WITHDRAWAL RATE AND FLOW OF 9750 GPM

FIGUR VI-1



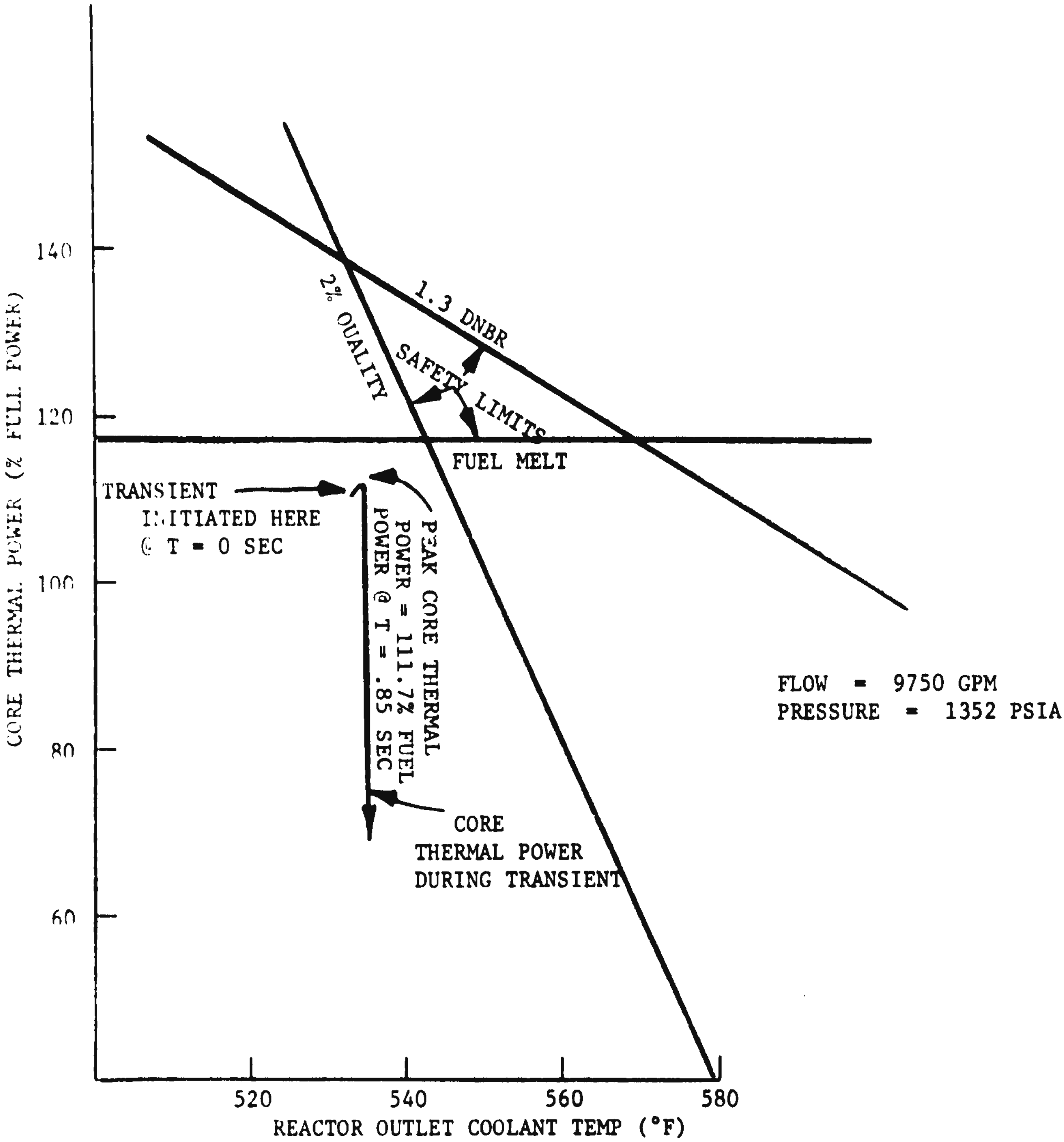
PRRWT REACTOR POWER, CORE THERMAL POWER, HOT SPOT FUEL TEMP & REACTOR OUTLET COOLANT TEMP VS TIME FOR INITIAL POWER LEVEL 107%, 2"/MIN ROD WITHDRAWAL RATE AND FLOW OF 9750 GPM

FIGURE VI-2



PRRWT REACTOR POWER, CORE THERMAL POWER, HOT SPOT FUEL TEMP & REACTOR OUTLET COOLANT TEMP VS TIME FOR INITIAL POWER LEVEL 111%, 2"/MIN ROD WITHDRAWAL AND FLOW OF 9750 GPM

FIGURE VI-3



PRRWT CORE THERMAL POWER VS REACTOR OUTLET TEMP FOR INITIAL POWER LEVEL 111% & 2"/MIN ROD WITHDRAWAL RATE

FIGURE VI-4

2. Scoping

(a) Initial conditions

Scoping a PRRWT requires all reactor and primary loop parameters to be set at their scram points, including associated errors. Table VI-3 contains these initial conditions.

TABLE VI-3

SCOPING RUN INITIAL CONDITIONS

Reactor power	115%
Reactor outlet temp	534°F
Primary coolant flow	8650 gpm
Primary system pressure	1265 psia

(b) Results

Scoping begins with reactor power at the high power scram point. During the .23 second lag time before negative reactivity insertion (see Section V) reactor power, core heat flux and reactor outlet coolant temperature will rise at rates approximating those achieved in the 111 percent initial power case (see Fig. VI-3). Table VI-4 indicates the conservative additions which must be made to these parameters at their 115 percent power steady-state value to simulate scoping at 115 percent initial power level.

TABLE VI-4

SCOPING AT 115% INITIAL POWER

Parameter	Steady-state value	Length of time parameter increases after scram initiated	Increases in parameter	Final value of parameter
Reactor power	115%	.23 sec	1.4%	116.4%
Core heat flux	115%	.28 sec	.5%	115.5%
Outlet temperature	534°F	.31 sec	1°F	535°F

This transient would then result in a minimum DNB ratio of 1.48, a maximum quality of +2.8 percent and a centerline hot spot fuel temperature of no more than 4740°F.

B. Rod Creep Transient

This transient was analyzed in detail in Section V of this report.

1. The Design Band

(a) Initial Conditions Prior to Initiation of Transient

It is assumed that the plant is operating with actual power, outlet temperature, system pressure, and flow of 115 percent, 534°F, 1352 psia and 9750 gpm, respectively.

(b) Results

The rod creep is initiated with no increase in power and a 14°F (Section V) increase in outlet temperature prior to scram. Concurrent with this increase in temperature is a 43 psi increase in system pressure.

The maximum quality and minimum DNB ratio achieved during this transient are 2 percent and 1.45, respectively.

As previously noted, the initial conditions on power and temperature are 115 percent and 534°F, respectively. It is further assumed that there is a 14°F overshoot in the outlet temperature. These are two consistant but conservative assumptions. If the plant is operating, steady-state, at the temperature scram point prior to transient initiation, the temperature overshoot will be negligible. If the transient is initiated at a lower temperature the overshoot is no more than 14°F, however, the pressure surge will be larger. The higher pressure at scram initiation leads to lower quality and higher DNB ratio.

2. Scoping

(a) Initial Conditions

This transient is analyzed by a steady-state analysis with the following input:

Power	115%
Flow	8650 gpm
Pressure	1265 psi
Outlet temp.	548°F

(b) Results

The maximum quality and minimum DNB ratio are +5.5 percent and 1.41, respectively.

C. Main Steam Line Rupture Transient

The MH-1A MSLRT is discussed in detail in Section V of this report. Reactor kinetics input parameters and analytical technique remain unchanged.

1. Design Band Limits

(a) Initial Conditions

A MSLRT initiated from the design band limits involves a primary coolant flow of 9750 gpm, a reactor outlet coolant temperature of 534°F, and a primary system pressure of 1265 psia. Since the primary coolant pressure decreases during the transients, it may be reasoned that the pressurizer heaters may not be able to maintain the pressure within the design band. Thus, the pressure of 1265 psia (low pressure scram point with associated error) was conservatively selected for the analysis. Table VI-5 shows the two power levels chosen as initial conditions.

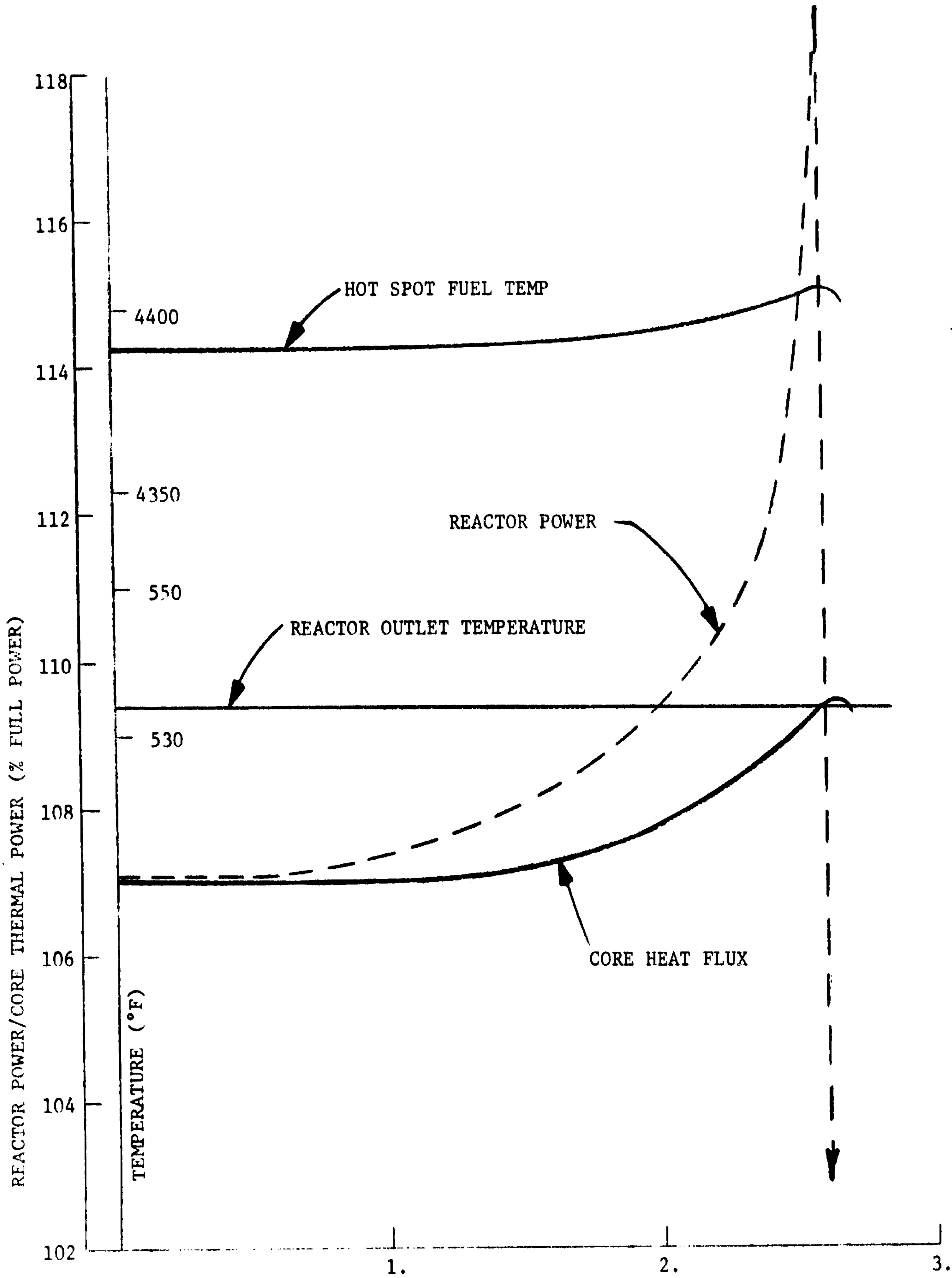
TABLE VI-5

INITIAL POWER LEVELS

Power Level (% full power)	Significance
107%	Operating band limit
111%	Alarm point

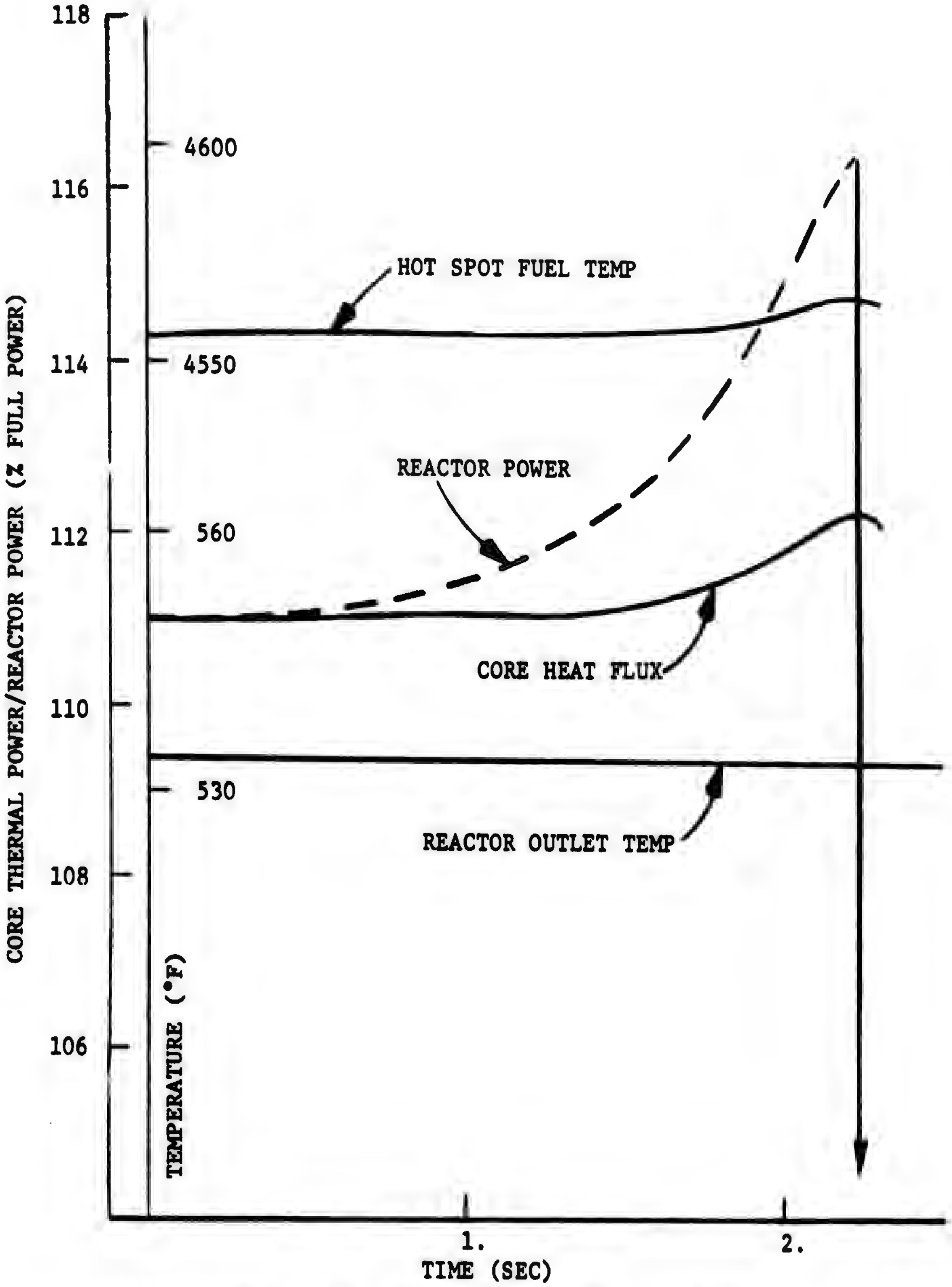
(b) Results

Figures VI-5 and VI-6 show the behavior of the MH-1A during MSLRT for each of the two sets of initial conditions. The figures show instantaneous reactor power, core heat flux, hot spot fuel temperature, and the reactor outlet coolant temperature as functions of time during the transient. Figure VI-6 shows the maximum core thermal flux achieved during this worst case to be 112.2 percent full power. The hot spot fuel temperature reaches a peak of 4564°F, far below the irradiated UO_2 melting point of 4800°F.



MSLRT REACTOR POWER, CORE THERMAL POWER, HOT SPOT FUEL TEMP AND REACTOR OUTLET COOLANT TEMP VS TIME FOR INITIAL POWER LEVEL 107%, INITIAL REACTOR OUTLET TEMP 534°F AND FLOW OF 9750 GPM

FIGURE VI-5



MSLRT REACTOR POWER, CORE THERMAL POWER HOT SPOT FUEL TEMP AND REACTOR OUTLET COOLANT TEMP VS TIME FOR INITIAL POWER LEVEL 111%, REACTOR OUTLET TEMP 534°F AND FLOW 9750 GPM

FIGURE VI-6

Figure VI-7 shows core thermal power vs. reactor outlet coolant temperature for this worst case transient. Again, the safety limits framework of 2 percent quality and 4800°F fuel temperature are verified held safe for this most severe MSLRT.

2. Scoping

Scoping a MSLRT requires all reactor and primary loop parameters to be set at their respective scram points, including associated errors.

Initial Conditions

MSLRT Scoping Run Initial Conditions

Reactor power	115%
Reactor outlet temp	534°F
Primary coolant flow	8650 gpm
Primary system pressure	1265 psia

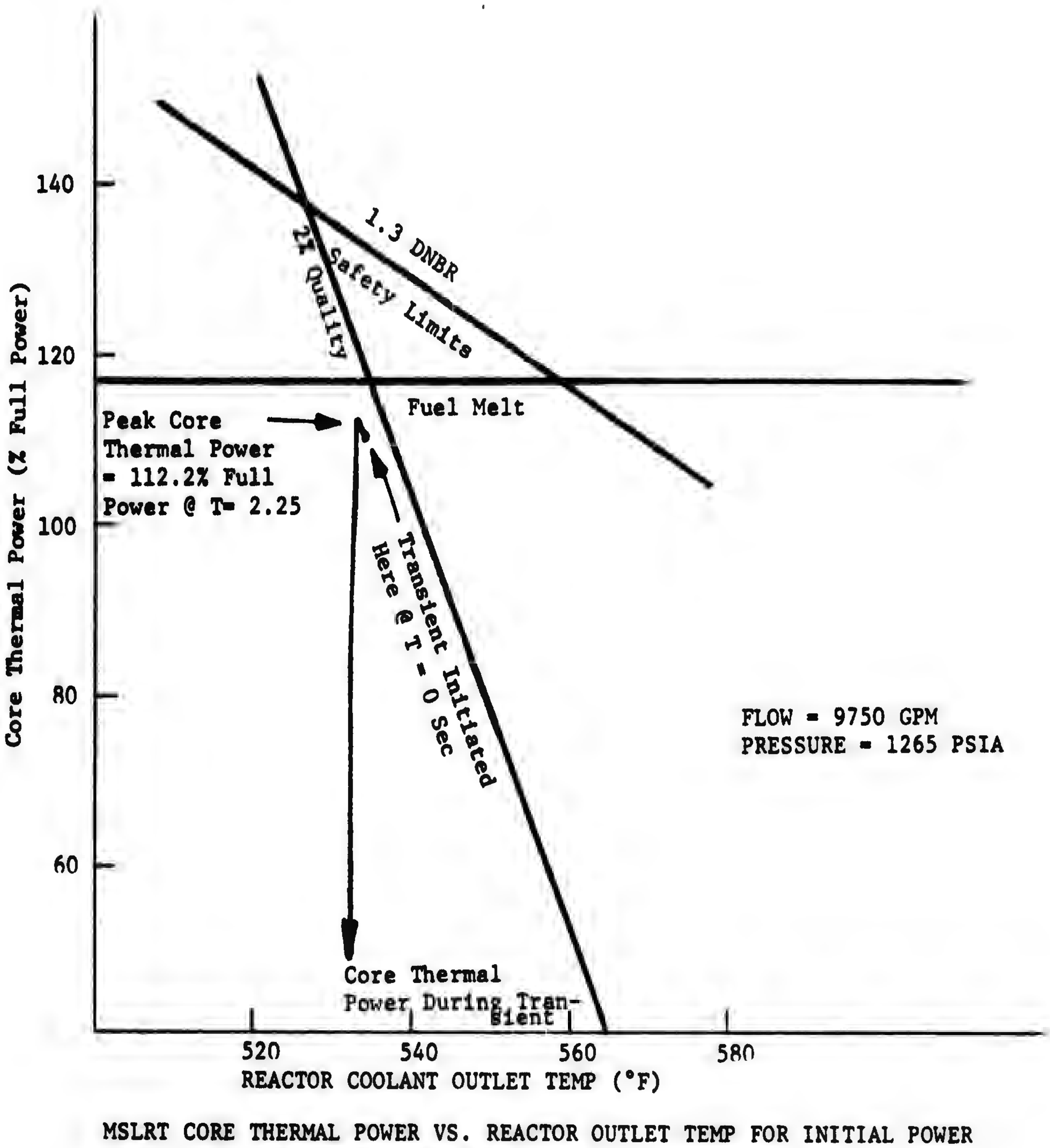
Initiation of a MSLRT scoping run occurs with reactor power at the high power scram point. During the .23 second lag time before negative reactivity insertion (see Sec. V) reactor power and core heat flux will rise as reactor outlet coolant temperature falls at rates approximating those achieved in the 111 percent case (Fig. VI-6). Table VI-6 below indicates the conservative corrections which must be applied to these parameters at their 115 percent power steady-state value to simulate scoping initiated at 115 percent power.

TABLE VI-6

SCOPING RUN RESULTANT PARAMETERS

Parameter	Steady-state value	Time parameter move adversely subsequent to scram initiation	Change in parameter	Peak adverse parameter value
Reactor power	115%	.23 sec	1.5%	116.5%
Core heat flux	115%	.28 sec	.45%	115.45%
Reactor outlet temp	534°F	.31 sec	1°F	533°F

This transient would then result in a minimum DNB ratio of 1.47, a maximum quality of +2.9 percent and a centerline hot spot fuel temperature of no more than 4742°F.



MSLRT CORE THERMAL POWER VS. REACTOR OUTLET TEMP FOR INITIAL POWER LEVEL 111% AND OUTLET TEMP 534°F

FIGURE VI-7

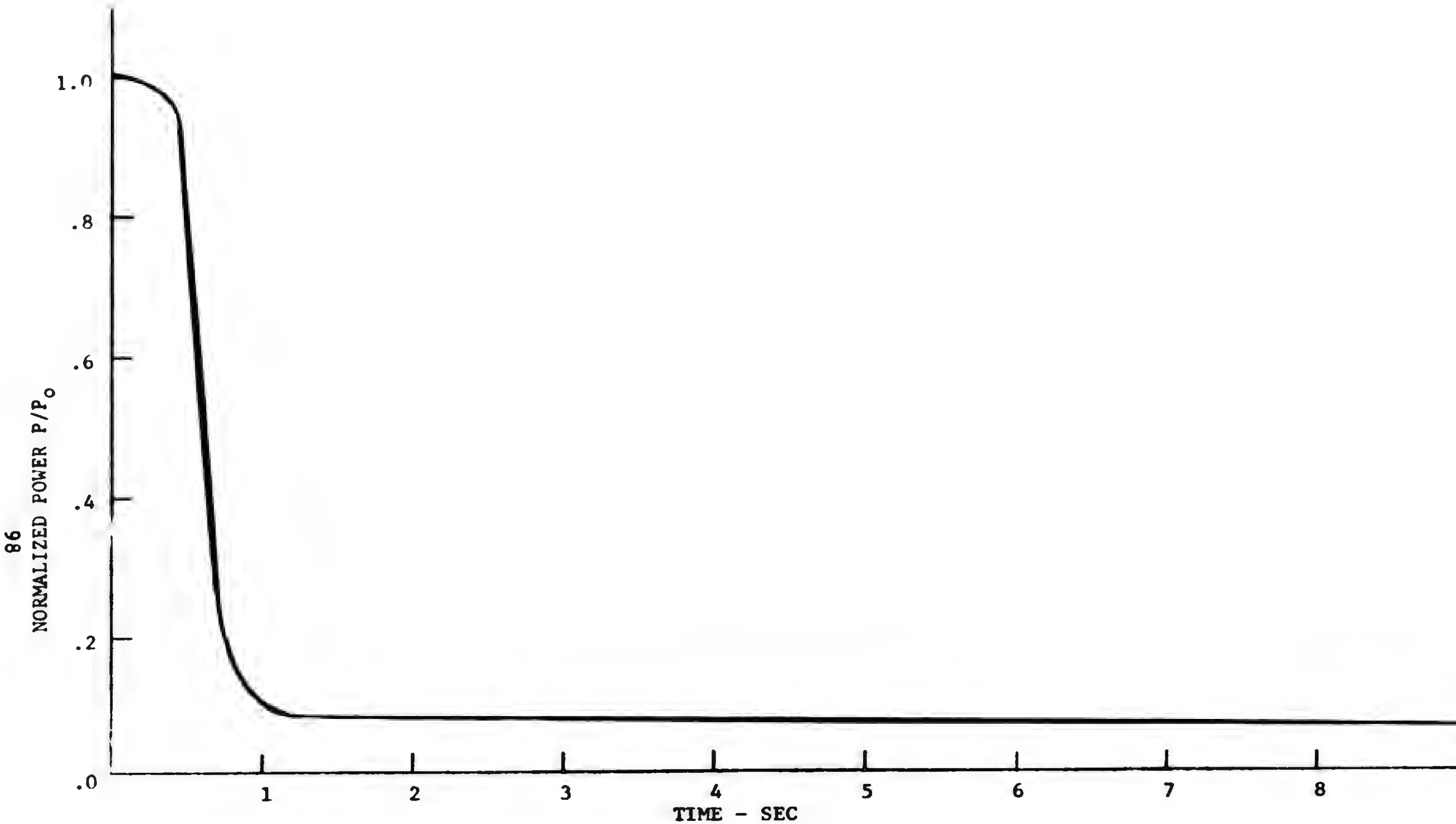
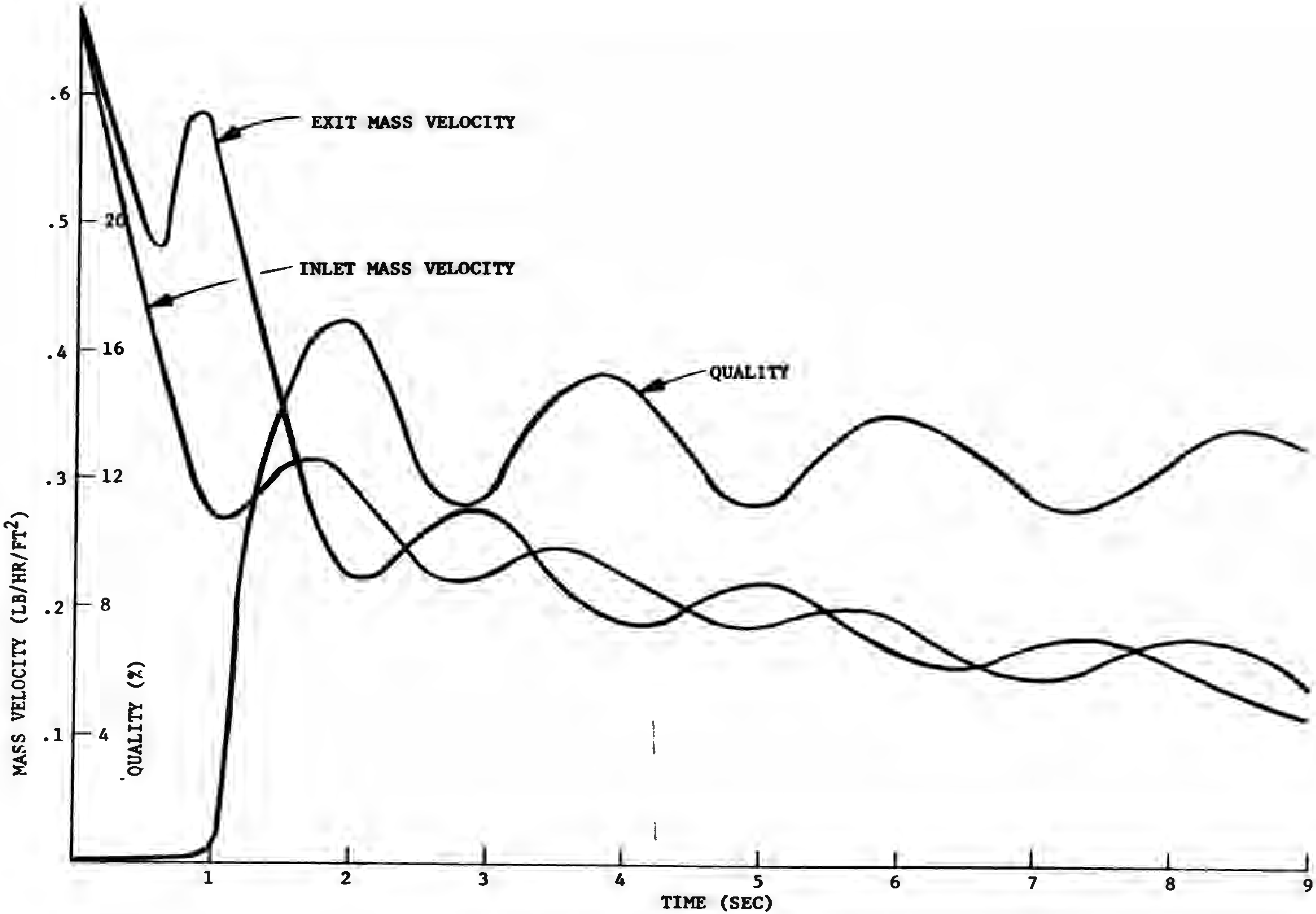


FIGURE VI-8 MH-1A CORE 3 LOFT (WCDB) P/P_o VS TIME



MH-1A LOFT FROM WCDB FIGURE VI-9

D. MH-1A Loss of Flow Analysis

1. Worst Case Design Band

(a) Initial Conditions

The LOFT was assumed initiated from the following conditions:

Power	111%
Flow	9750 gpm
Temp (outlet)	534°F
Pressure	1352 psia

A delay from LOFT initiation to rod motion of 500 ms was assumed (see Technical Specifications). Figure VI-8 shows the power vs. time (loss of service of both pumps was assumed).

(b) Results

Figure VI-9 illustrates the results obtained from the CHIC-KIN program for this transient. All oscillations, inlet mass velocity, exit mass velocity, and exit quality are decaying with time. The exit quality remains high at about 13 percent. However, due to the steadily decreasing heat flux, this presents no burnout problems.

2. Scoping

(a) Initial Conditions

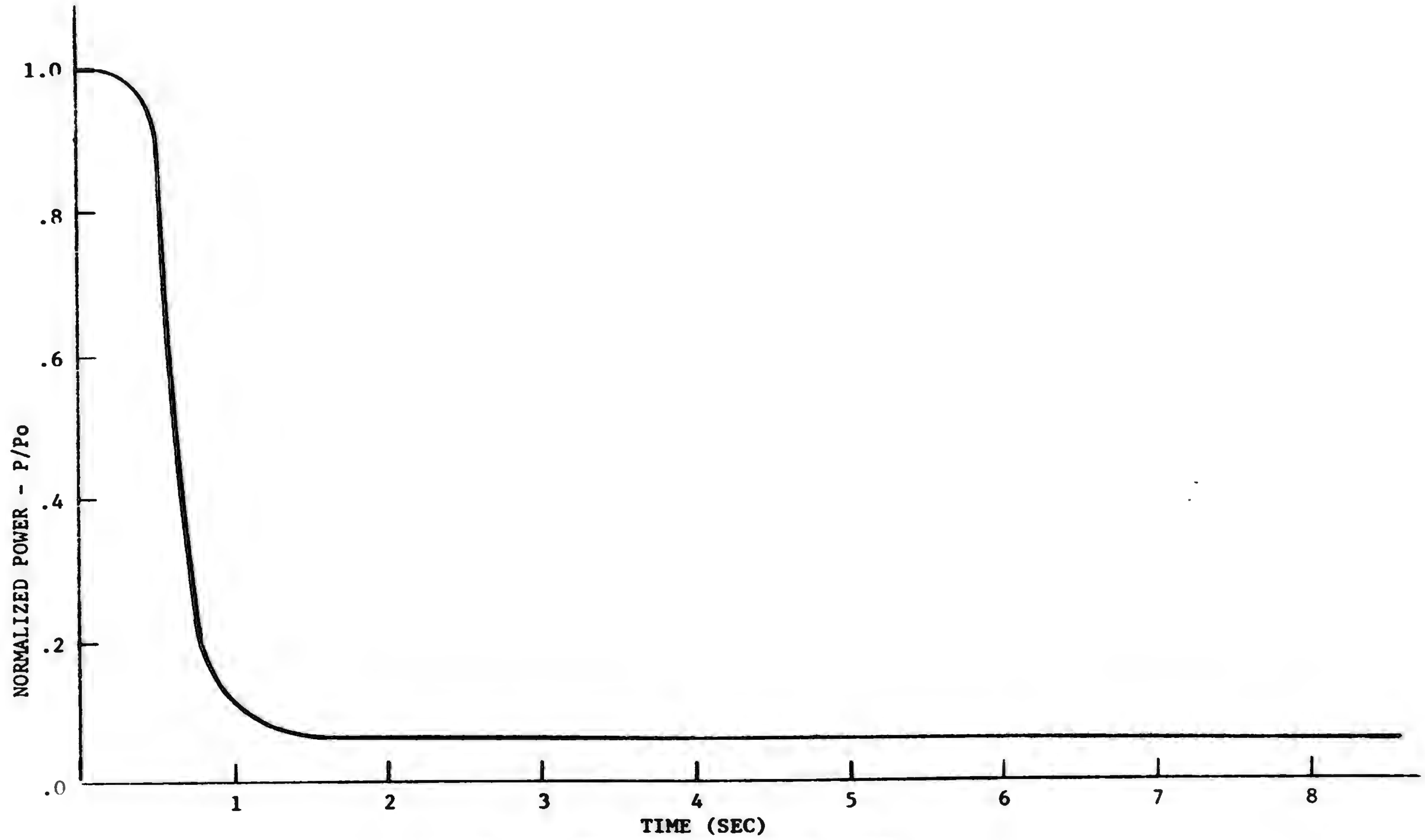
The initial conditions for the scoping transient were:

Power	115%
Flow	8650 gpm
Temperature outlet	534°F
Pressure	1265 psia.

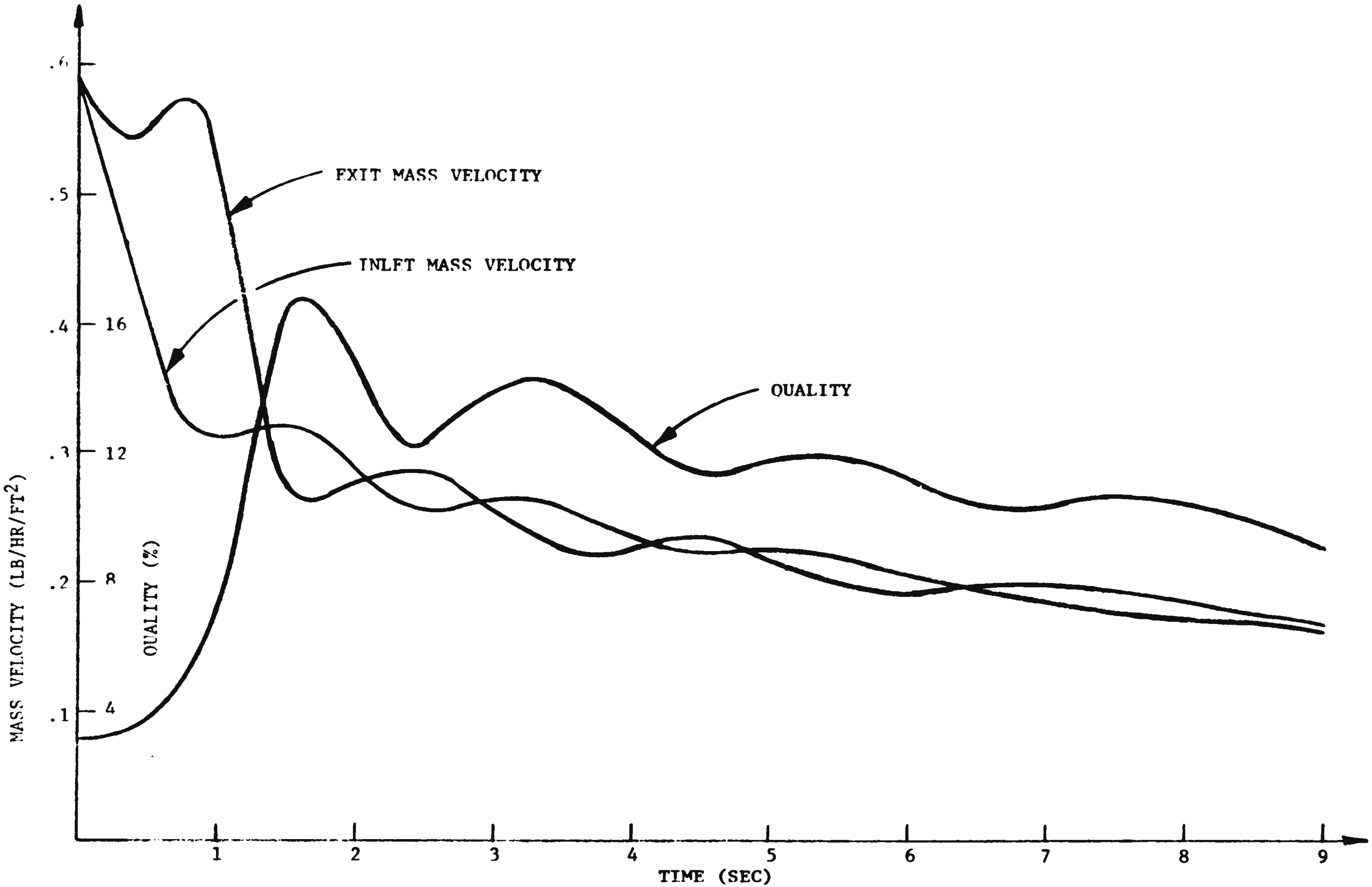
The same 500 ms delay was used. Figure VI-10 shows the power trace.

(b) Results

The combination of low flow and high power gave 4 percent more inlet subcooling than for the WCDB transient. This combination coupled with lower pressure yielded an initial exit quality of 3 percent. The subcooling and initial quality effected the transient to yield even more desirable results than were obtained with the WCDB - LOFT. This can be seen in Figure VI-11 where the oscillations are smother than in Figure VI-9 and the quality is decreasing faster.



MH-1A SCOPE OF LOFT P/Po VS TIME
FIGURE VI-10



MH-1A LOFT SCOPE FIGURE VI-11

REFERENCES

1. "Safety Analysis STURGIS (MH-1A) Floating Nuclear Power Plant," Nuclear Power Field Office, Revision A, 2 October 1967.

2. "General Design Criteria for Nuclear Power Plant Construction Permits," Proposed Amendment to 10CFR Part 50, Federal Register Vol 32, No. 132, July 11, 1967.

3. L. S. Tong, "Prediction of Departure from Nucleate Boiling for an Axially Non-uniform Heat Flux Distribution," J. Nuclear Energy, Vol. 21, (1967).

4. Lyons, M. F., et al., "UO_2 Pellet Thermal Conductivity from Irradiations with Central Melting", ANS Transactions, Vol 7 No. 1, June 1964.

5. Memorandum for Record, "CHEETAH - A Computer Code to Calculate Steady-State and Transient Temperature Profiles in a Cylindrical UO_2 Fuel Pellet", USAERG-ED, 17 April 1970

6. D. S. Rowe, "Cross Flow Mixing Between Flow Channels During Boiling, Part 1, COBRA - Computer Program for Coolant Boiling in Rod Arrays", BNWL-371 Pt 1, Battelle Northwest, March 1967.

7. "Specifications for MH-1A Floating Nuclear Reactor Replacement Cores", NYO-10738, New York Operations Office, January 1966.

8. Meyers, C. A., et al., "MH-1A Steady-State and Transient Thermal-Hydraulic Design Analysis," ED-6916, USAERG, 25 July 1969.

9. D. S. Rowe and E. W. Angle, "Crossflow Mixing Between Parallel Flow Channels During Boiling," Part II Measurement of Flow and Enthalpy in Two Parallel Channels," BNWL-371 PT2, Pacific Northwest Laboratory, December 1967.

10. Belle, "Uranium Dioxide: Properties and Nuclear Applications," July, 1961.

11. Gregory, M. J. et al., "The Theoretical Nuclear Analysis of the MH-1A": Cores II and III (ED 7102).

12. Thom, J. R. S., W. M. Walker, T. A. Fallon, and G. F. S. Reising, "Boiling in Subcooled Water During Flow Up Heated Tubes or Annuli," Proc. Instn. Mech. Engs. 1965-66, Vol. 180, Part 3C, (1966).

13. J. S. Christensen, J. J. Allio, A. Biancheria, "Melting Point of Irradiated Uranium Dioxide," WCAP-6065, Westinghouse Atomic Power Division, February 1965.

14. "High Performance UO_2 Program Molten Fuel Rod Operation to High Burnup," Lyons et al, GEAP-5100-2, November 1966.

15. Redfield, J. A. and S. C. Margolis, "TOPS - Fortran Program for the Transient Thermodynamics of Pressurizers", WAPD-TM-515, December 1965.

16. Schmidt, E. R. et al, "MH-1A Loss of Flow Transient Safety Analysis and Parametric Study", NUS-550, April 1969.

17. System/360 Continuous System Modelling Program (360A-CX-16X)-Users Manual, H20-0367-2, IBM.

18. Redfield, J. A., "CHIC-KIN - A Fortran Program for Intermediate and Fast Transients in a Water Moderated Reactor WAPD-TM-479, January 1965.

19. VandenBerg, A. J. "MH-1A Reactor Coolant Pump ΔPressure Setpoint", Engineering Support Study, E&I Br/ED/USAERG, 16 July 1970.