ATTACHMENT 1

VIRGINIA ELECTRIC AND POWER COMPANY
PROPOSED TECHNICAL SPECIFICATION REVISION

F∆h INCREASE/STATISTICAL DNBR METHODOLOGY - AFFECTED TECHNICAL SPECIFICATION PAGES

conservative, than the loci of points of thermal power, coolant system average temperature, and coolant system pressure for which either the calculated DNBR is equal to the design DNBR limit or the average enthalpy at the exit of the core is equal to the saturation value. At low pressures or high temperatures the average enthalpy at the exit of the core reaches saturation before the calculated DNBR reaches the design DNBR limit and, thus, this arbitrary limit is conservative with respect to maintaining clad integrity. The plant conditions required to violate these limits are precluded by the protection system and the self-actuated safety valves on the steam generator. Upper limits of 70% power for loop stop valves open and 75% with loop stop valves closed are shown to completely bound the area where clad integrity is assured. These latter limits are arbitrary but cannot be reached due to the Permissive 8 protection system setpoint which will trip the reactor on high nuclear flux when only two reactor coolant pumps are in service.

Operation with natural circulation or with only one loop in service is not allowed since the plant is not designed for continuous operation with less than two loops in service.

TS Figure 2.1-1 is based on an $F\Delta H(N)$ of 1.62, a 1.55 cosine axial flux shape, and a deterministic DNB analysis procedure including margin to accommodate rod bowing[1]. TS Figure 2.1-1 is also bounding for a statistical treatment of key DNBR analysis parameter uncertainties including an enthalpy rise hot channel factor which follows the following functional form: $F\Delta H(N) = 1.56\,[1 + 0.3(1-P)]$ where P is the fraction of rated power. TS Figures 2.1-2 and 2.1-3 are based on an $F\Delta H(N)$ of 1.55, a deterministic treatment of key DNB analysis parameter uncertainties, and include a 0.2 rather than 0.3 part power multiplier for the enthalpy rise hot channel factor.

These hot channel factors are higher than those calculated at full power over the range between that of all control rod assemblies fully withdrawn to maximum allowable control rod assembly insertion. The control rod assembly insertion limits are covered by Specification 3.12. Adverse power distribution factors could occur at lower power levels because additional control rod assemblies are in the core; however, the control rod assembly insertion limits dictated by TS Figure 3.12-1A (Unit 1) and 3.12-1B (Unit 2) ensure that the DNBR is always greater at partial power than at full power.

The Reactor Control and Protection System is designed to prevent any anticipated combination of transient conditions for Reactor Coolant System temperature, pressure and thermal power level that would result in a DNBR less than the design DNBR limit(3) based on steady state nominal operating power levels less than or equal to 100%, steady state nominal operating Reactor Coolant System average temperatures less than or equal to 574.4°F and a steady state nominal operating pressure of 2235 psig. For deterministic DNBR analysis, allowances are made in initial conditions assumed for transient analyses for steady state errors of +2% in power, +4°F in Reactor Coolant System average temperature and ±30 psi in pressure. The combined steady state errors result in the DNB ratio at the start of a transient being 10 percent less than the value at nominal full power operating conditions. The steady state nominal operating parameters and allowances for steady state errors given above are also applicable for two loop operation except that the steady state nominal operating power level is less than or equal to 60%.

For statistical DNBR analyses, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95% probability that the minimum DNBR for the limiting rod is greater than or equal to the statistical DNBR limit. The uncertainties in the plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit, establishes a statistical DNBR limit which must be met in plant safety analyses using values of input parameters without uncertainties. The statistical DNBR limit also ensures that at least 99.9% of the core avoids the onset of DNB when the limiting rod is at the DNBR limit.

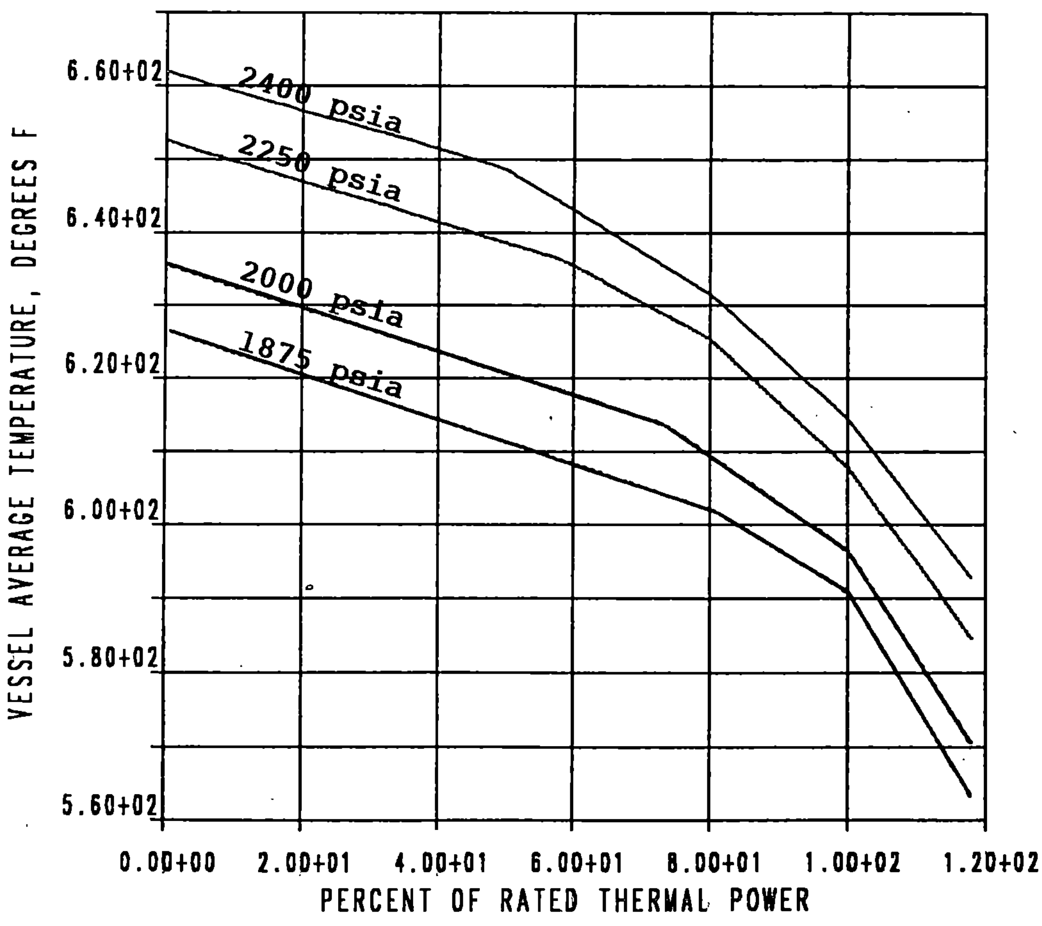
The fuel overpower design limit is 118% of rated power. The overpower limit criterion is that core power be prevented from reaching a value at which fuel pellet melting would occur. The value of 118% power allows substantial margin to this limiting criterion. Additional peaking factors to account for local peaking due to fuel rod axial gaps and reduction in fuel pellet stack length have been included in the calculation of this limit.

References

1) FSAR Section 3.4
2) FSAR Section 3.3
3) FSAR Section 14.2

TS FIGURE 2.1-1

REACTOR CORE THERMAL AND HYDRAULIC SAFETY LIMITS - THREE LOOP OPERATION, 100% FLOW



B. Power Distribution Limits

1. At all times except during low power physics tests, the hot channel factors defined in the basis meet the following limits:

$F_Q(Z) \leq 2.32/P \times K(Z)$ for $P > 0.5$

$F_Q(Z) \leq 4.64 \times K(Z)$ for $P \leq 0.5$

$F^N_{\Delta H} \leq 1.56 [1 + 0.3 (1\text{-}P)]$ for three loop operation

$\leq 1.55 [1 + 0.2 (1\text{-}P)]$ for two loop operation

where P is the fraction of rated power at which the core is operating, K(Z) is the function given in TS Figure 3.12-8, and Z is the core height location of F_Q.

Prior to exceeding 75% power following each core loading and during each effective full power month of operation thereafter, power distribution maps using the movable detector system shall be made to confirm that the hot channel factor limits of this specification are satisfied. For the purpose of this confirmation:

a. The measurement of total peaking factor F_Q^{Meas} shall be increased by eight percent to account for manufacturing tolerances, measurement error and the effects of rod bow. The measurement of enthalpy rise hot channel factor $F_{\Delta H}$ shall be compared directly to the limit specified in Specification 3.12.B.1. If any measured hot channel factor exceeds its limit specified under Specification 3.12.B.1, the reactor power and high neutron flux trip setpoint shall be reduced until the limits under Specification 3.12.B.1 are met. If the hot channel factors cannot be brought to within the limits of $F_Q(Z) \leq 2.32/P \times K(Z)$ and $F^N_{\Delta H} \leq 1.56$ within 24 hours, the Overpower ΔT and Overtemperature ΔT trip setpoints shall be similarly reduced.

c. In hot, intermediate and cold shutdown conditions, the step demand counters shall be operable and capable of determining the group demand positions to within ±2 steps. The rod position indicators shall be available to verify rod movement upon demand.

2. If a rod position indicator channel is out of service, then:

a. For operation above 50% of rated power, the position of the RCC shall be checked indirectly using the movable incore detectors at least once power 8 hours and immediately after any motion of the non-indicating rod exceeding 24 steps, or

b. Reduce power to less than 50% of rated power within 8 hours. During operations below 50% of rated power, no special monitoring is required.

3. If more than one rod position (RPI) indicator channel per group or two RPI channels per bank are inoperable during control bank motion to achieve critically or power operations, then the requirements of Specification 3.0.1 will be followed.

F. <u>DNB PARAMETERS</u>

1. The following DNB related parameters shall be maintained within their limits during power operation:

Reactor Coolant System $T_{avg} \leq 578.4°F$
Pressurizer Pressure ≥ 2205 psig
Reactor Coolant System Total Flow Rate ≥ 273,000 gpm

a. The Reactor Coolant System T_{avg} and Pressurizer Pressure shall be verified to be within their limits at least once every 12 hours.

b. The Reactor Coolant System Total Flow Rate shall be determined to be within its limit by measurement at least once per refueling cycle.

2. When any of the parameters in Specification 3.12.F.1 has been determined to exceed its limit, either restore the parameter to within its limit within 2 hours or reduce reactor power to less than 5% of rated thermal power within the next 4 hours.

3. The limit for Pressurizer Pressure in Specification 3.12.F.1 is not applicable during either a thermal power ramp increase in excess of 5% of rated thermal power per minute or a thermal power step increase in excess of 10% of rated thermal power.

Basis

The reactivity control concept assumed for operation is that reactivity changes accompanying changes in reactor power are compensated by control rod assembly motion. Reactivity changes associated with xenon, samarium, fuel depletion, and large changes in reactor coolant temperature (operating temperature to cold shutdown) are compensated for by changes in the soluble boron concentration. During power operation, the shutdown groups are fully withdrawn and control of power is by the control groups. A reactor trip occurring during power operation will place the reactor into the hot shutdown condition. The control rod assembly insertion limits provide for achieving hot shutdown by reactor trip at any time, assuming the highest worth control rod assembly remains fully withdrawn, with sufficient margins to meet the assumptions used in the accident analysis. In addition, they provide a limit

In addition to the above, the peak linear power density and the nuclear enthalpy rise hot channel factor must not exceed their limiting values which result from the large break loss of coolant accident analysis based on the ECCS acceptance criteria limit of 2200°F on peak clad temperature. This is required to meet the initial conditions assumed for the loss of coolant accident. To aid in specifying the limits of power distribution, the following hot channel factors are defined.

$F_Q(Z)$, Height Dependent Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerance on fuel pellets and rods.

F_Q^E, Engineering Heat Flux Hot Channel Factor, is defined as the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of the fuel rod, and eccentricity of the gap between pellet and clad. Combined statistically the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux for non-statistical applications.

$F_{\Delta H}^N$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power for both LOCA and non-LOCA considerations.

It should be noted that the enthalpy rise factors are based on integrals and are used as such in the DNB and LOCA calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in radial (x-y) power shapes throughout the core. Thus, the radial power shape at the point of maximum heat flux is not necessarily directly related to the enthalpy rise factors. The results of the loss of coolant accident analyses are conservative with respect to the ECCS acceptance criteria as specified in 10 CFR 50.46 using the upper bound $F_Q(Z)$ times the hot channel factor normalized operating envelope given by TS Figure 3.12-8.

When an F_Q measurement is taken, measurement error, manufacturing tolerances, and the effects of rod bow must be allowed for. Five percent is the appropriate allowances for measurement error for a full core map (greater than or equal to 38 thimbles, including a minimum of 2 thimbles per core quandrant, monitored) taken with the movable incore detector flux mapping system, three percent is the appropriate allowance for manufacturing tolerances, and five percent is appropriate allowance for rod bow. These uncertainties are statistically combined and result in a net increase of 1.08 that is applied to the measured value of F_Q.

In the specified limit of $F^N_{\Delta H}$, there is a four percent error allowance, which means that normal operation of the core is expected to result in $F^N_{\Delta H} \leq 1.56\ [1 + 0.3\ (1\text{-}P)]/1.04$. The 4% allowance is based on the considerations that (a) normal perturbations in the radial power shape (e.g., rod misalignment) affect $F^N_{\Delta H}$, in most cases without necessarily affecting F_Q, (b) the operator has a direct influence on F_Q through movement of rods and can limit it to the desired value; he has no direct control over $F^N_{\Delta H}$, and (c) an error in the predictions for radial power shape, which may be detected during startup physics tests and which may influent F_Q, can be compensated for by tighter axial control. An appropriate allowance for the measurement uncertainty for $F_{\Delta H}^{N}$ obtained from a full core map (≥ 38 thimbles, including a minimum of 2 detectors per core quadrant, monitored) taken with the movable incore detector flux mapping system has been incorporated in the statistical DNBR limit. Measurement of the hot channel factors are required as part of startup physics tests, during each effective full power power month of operation, and whenever abnormal power distribution conditions require a reduction of core power to a level based on measured hot channel factors. The incore map taken following core loading provides confirmation of the basic nuclear design bases including proper fuel loading patterns. The periodic incore mapping provides additional assurance that the nuclear design bases remain inviolate and identify operational anomalies which would, otherwise, affect these bases.

For normal operation, it has been determined that, provided certain conditions are observed, the enthalpy rise hot channel factor $F_{\Delta H}^{N}$ limit will be met. These conditions are as follows:

1. Control rods in a single bank move together with no individual rod insertion differing by more than 15 inches from the bank demand position. An indicated misalignment limit of 13 steps precludes a rod misalignment no greater than 15 inches with consideration of maximum instrumentation error.

2. Control rod banks are sequenced with overlapping banks as shown in TS Figures 3.12-1A, 3.12-1B.

3. The full length control bank insertion limits are not violated.

4. Axial power distribution control procedures, which are given in terms of flux difference control and control bank insertion limits are observed. Flux differences refers to the difference

In some instances of rapid unit power reduction automatic rod motion will cause the flux difference to deviate from the target band when the reduced power level is reached. This does not necessarily affect the xenon distribution sufficiently to change the envelope of peaking factors which can be reached on a subsequent return to full power within the target band; however, to simplify the specification, a limitation of one hour in any period of 24 hours is placed on operation outside the band. This ensures that the resulting xenon distributions are not significantly different from those resulting from operation within the target band. The instantaneous consequences of being outside the band, provided rod insertion limits are observed, is not worse than a 10 percent increment in peaking factor for the allowable flux difference at 90% power, in the range ± 13.8 percent (± 10.8 percent indicated) where for every 2 percent below rated power, the permissible flux difference boundary is extended by 1 percent.

As discussed above, the essence of the procedure is to maintain the xenon distribution in the core as close to the equilibrium full power condition as possible. This is accomplished, by using the boron system to position the full length control rods to produce the required indicated flux difference.

A 2% quadrant tilt allows that a 5% tilt might actually be present in the core because of insensitivity of the excore detectors for disturbances near the core center such as misaligned inner control rod and an error allowance. No increase in F_Q occurs with tilts up to 5% because misaligned control rods producing such tilts do not extend to the unrodded plane, where the maximum F_Q occurs.

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the UFSAR assumptions and have been analytically demonstrated to be adequate to maintain a minimum DNBR which is greater than the design limit throughout each analyzed transient. Measurement uncertainties are accounted for in the DNB design margin. Therefore, measurement values are compared directly to the surveillance limits without applying instrument uncertainty.

The 12 hour periodic surveillance of temperature and pressure through instrument readout is sufficient to ensure that these parameters are restored to within their limits following load changes and other expected transient operation. The measurement of the RCS total flow rate once per refueling cycle is adequate to detect flow degradation.

TS FIGURE 3.12-8

HOT CHANNEL FACTOR NORMALIZED OPERATING ENVELOPE

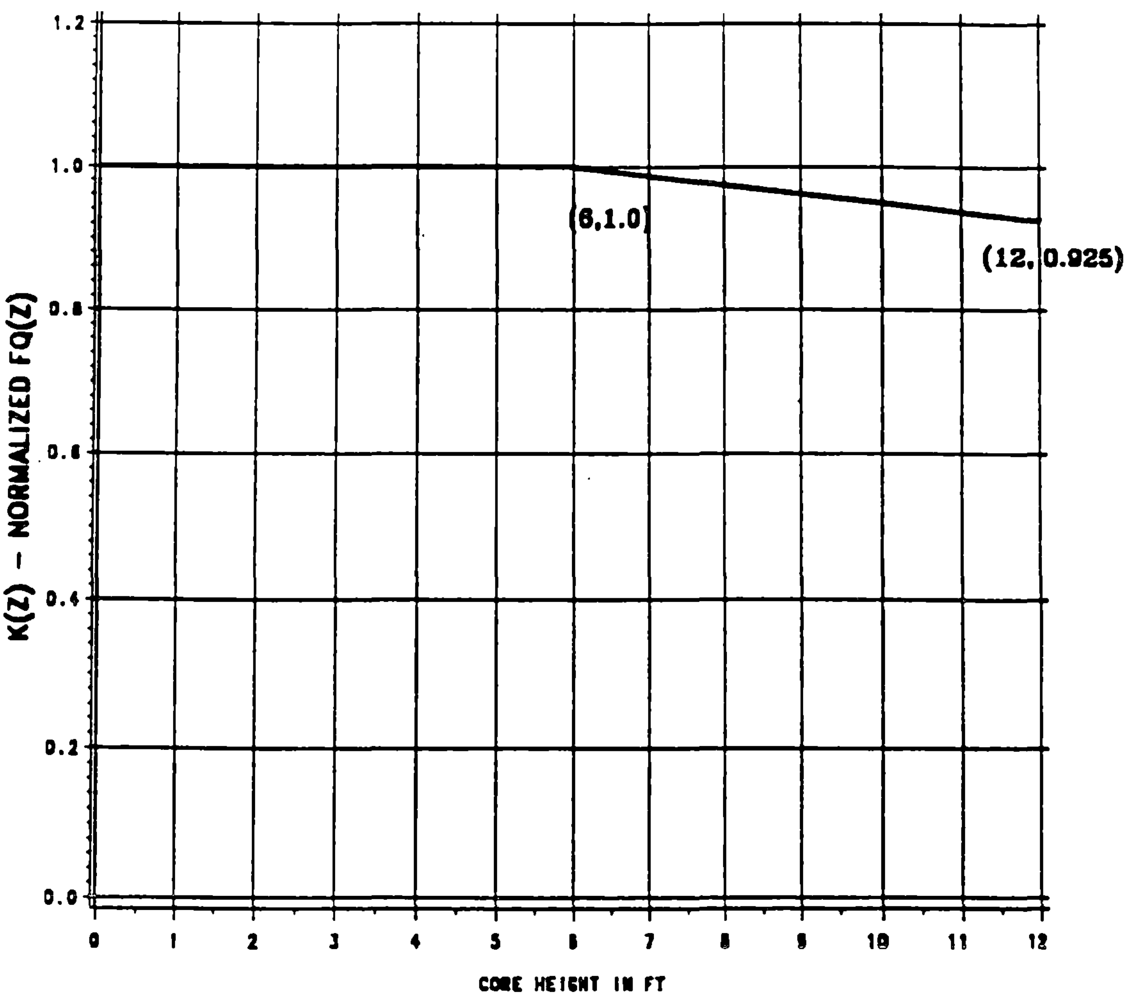


TABLE 4.1-2A (CONTINUED)

MINIMUM FREQUENCY FOR EQUIPMENT TESTS

	DESCRIPTION	TEST	FREQUENCY	FSAR SECTION REFERENCE
18.	Primary Coolant System	Functional	1. Periodic leakage testing [a] on each valve listed in Specification 3.1.C.7a shall be accomplished prior to entering power operation condition after every time the plant is placed in the cold shutdown condition for refueling, after each time the plant is placed in cold shutdown condition for 72 hours if testing has not been accomplished in the preceeding 9 months, and prior to returning the valve to service after maintenance, repair or replacement work is performed.	
19.	Containment Purge MOV Leakage	Functional	Semi-Annual (Unit at power or shutdown) if purge valves are operated during interval [c]	
20.	Containment Hydrogen Analyzers	a. Channel Check	Once per 12 hours	
		b. Channel Functional Test	Once per 31 days	
		c. Channel Calibration using sample gas containing: 1. One volume percent (± 0.25%) hydrogen, balance nitrogen 2. Four volume percent (± 0.25%) hydrogen, balance nitrogen 3. Channel calibration test will include startup and operation of the Heat Tracing System	Once per 92 days on staggered basis	
21.	RCS Flow	Flow ≥ 273,000 gpm	Once per refueling cycle	14

(a) To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

(b) Minimum differential test pressure shall not be below 150 psid.

(c) Refer to Section 4.4 for acceptance criteria.

ATTACHMENT 2

VIRGINIA ELECTRIC AND POWER COMPANY
PROPOSED TECHNICAL SPECIFICATION REVISION

$F_{\Delta h}$ INCREASE/STATISTICAL DNBR METHODOLOGY - DISCUSSION AND SAFETY EVALUATION OF PROPOSED CHANGES

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1.0 INTRODUCTION

This Safety Evaluation summarizes the analyses and evaluations performed to justify implementation of the Virginia Electric and Power Company (Virginia Power) Statistical DNBR Evaluation Methodology (Stat DNB) (1) and an increased Technical Specification $F\Delta h$ limit at Surry Units 1 and 2. The increased $F\Delta h$ limit is needed at this time primarily to accommodate the increased radial power factors which will result from the installation of flux suppression inserts (FSI) in Surry Unit 1. The FSI's reduce the peripheral core power and, hence, the fast neutron flux near vessel locations which are experiencing neutron irradiation embrittlement. The implementation of Stat DNB provides increased thermal-hydraulic margin for Surry Improved Fuel (SIF) by treating key DNBR analysis uncertainties in a less restrictive, but appropriately conservative, statistical manner.

Surry cores are currently comprised of two fuel types: the Westinghouse Standard (STD) 15x15 product (also called LOPAR, for "Low Parasitic") and the newer Surry Improved Fuel (SIF) 15x15 product. The use of LOPAR fuel is currently being phased out in favor of SIF. Future cycles will include no fresh LOPAR fuel assemblies; by Cycle 12, greater than 90% of the reload cores in both units will be composed of SIF fuel. The two fuel types have grids with slightly different hydraulic characteristics, and they must be analyzed with different DNB correlations (W-3L for LOPAR, and WRB-1 for SIF) (16), (17).

A Statistical DNBR limit (SDL) and the parameters associated with a statistical treatment of uncertainties for SIF/WRB-1 have been developed in accordance with the method described in the Statistical DNBR Methodology Topical Report (VEP-NE-2-A) (1). With the exception of the Complete Loss of Flow Accident (LOFA), however, the accident analyses and evaluations supporting the increased $F_{\Delta h}$ limit will continue to utilize a conservative deterministic treatment of uncertainties.

The Small Break LOCA (SBLOCA) has been explicitly reanalyzed to assess the impact of an increased $F_{\Delta h}$ to 1.62. The reanalysis of the LOFA for SIF/WRB-1 utilizing a statistical treatment of uncertainties and a 1.62 $F_{\Delta h}$, and the SBLOCA reanalysis both conservatively assume a core power of 2546 MWth (approximately 104% of current nominal power).

The proposed Technical Specification surveillance limit $F_{\Delta h}$ is 1.56. For deterministic and non-DNB accident analysis, a maximum $F_{\Delta h}$ of 1.62 will be assumed, reflecting a 4% measurement uncertainty. Statistical DNB accident analysis will assume a maximum $F_{\Delta h}$ of 1.56, since the 4% measurement uncertainty is combined statistically in the DNBR limit. Revised core thermal limits have been developed to accommodate the increased $F_{\Delta h}$ limit. Because the existing protection system setpoints have been shown to provide bounding core thermal limit protection with an increased $F_{\Delta h}$ of 1.62, no revised protection system setpoints are being proposed at this time.

DNBR analysis of LOPAR and SIF fuel is currently performed utilizing a deterministic treatment of key DNBR analysis uncertainties. Because

LOPAR is being phased out of future reload core designs, an explicit Statistical DNBR implementation analysis for LOPAR is not warranted. Instead, Virginia Power will continue to employ a deterministic treatment of key DNBR parameter uncertainties for any LOPAR assemblies incorporated into future core designs.

For deterministic DNBR and non-DNB analysis (applicable to both SIF/WRB-1 and LOPAR/W-3), the following FΔh limits and treatment of FΔh uncertainties are proposed:

	Current	Proposed
Nuclear Design FΔh	1.435	1.50
Calculational Uncertainty	4%	4%
Measurement Uncertainty	4%	4%
Safety Analysis FΔh	1.55	1.62

For statistical DNBR analysis (applicable to SIF/WRB-1 only), the following FΔh limits and treatment of FΔh uncertainties are proposed:

	Current	Proposed
Nuclear Design FΔh	1.435	1.50
Calculational Uncertainty	4%	4%
Measurement Uncertainty	4%	Accommodated in the SDL
Safety Analysis FΔh	1.55	1.56

These numbers reflect a 4.5% increase in both the nuclear design and safety analysis FΔh limits. Appropriate Technical Specifications changes have been prepared to implement these changes. The information presented in this report will serve as the basis for the Technical Specifications change package.

2.0 PROPOSED TECHNICAL SPECIFICATIONS CHANGES

The Technical Specifications changes proposed in this report may be grouped into two categories. Those categories are: 1) the Statistical DNBR Methodology implementation, and 2) the $F_{\Delta h}$ increase. The proposed changes are listed and summarized below.

1. Statistical Methodology Implementation
   - TS 2.1 Basis and References, reference to fuel densification power spiking deleted (pages 2.1-4 and 2.1-6); margin is now provided by the Statistical Methodology
   - TS 2.1 Basis, treatment of allowance for initial conditions and description of design DNBR limit updated (page 2.1-5)
   - TS 3.12.F, a new requirement for DNB Parameter Surveillance is included (page 3.12-11)
   - TS 3.12 Basis, FQE penalty qualified as being applicable only in non-statistical analyses (page 3.12-14)
   - TS 3.12 Basis, a new section supplementing 3.12.F is added (page 3.12-19)
   - TS Table 4.1-2A, Minimum Frequency for Equipment Tests, a new requirement for periodic measurement of RCS flow.

2. $F_{\Delta h}$ Increase
   - TS 2.1 Bases updated to reflect new $F_{\Delta h}$ limit (page 2.1-4)
   - TS Figure 2.1-1 (Reactor Core Thermal and Hydraulic Safety Limits) revised
   - TS 3.12.B.1 updated to reflect new $F_{\Delta h}$ limit (page 3.12-3)
   - TS 3.12.B.1.a updated to reflect new $F_{\Delta h}$ limit (page 3.12-3)
   - TS 3.12 Basis updated to reflect new $F_{\Delta h}$ limit (page 3.12-15 and 3.12-16)
   - TS Figure 3.12-8 (K(z); Normalized FQ(z)) revised

More detailed discussions of these changes are included, where necessary, in the report sections which follow.

3.0 STATISTICAL DNBR METHODOLOGY IMPLEMENTATION

3.1 Background Information

In October, 1985 Virginia Power submitted to the NRC a topical report describing a proposed Methodology for the statistical treatment of key uncertainties in core thermal-hydraulic (DNBR) analysis. The Methodology provided DNBR margin through the use of statistical rather than deterministic uncertainty treatment. The margin could then be used to provide relief in areas where plant safety analysis is DNBR-limited. The Methodology was subsequently approved and the Staff's Safety Evaluation appears as a preface to the final version of the report (1).

A submittal package implementing the Methodology for North Anna was submitted to the NRC in June, 1987 (2). The package employed the margin to provide a relaxed Technical Specifications end-of-cycle Moderator Temperature Coefficient, as well as to employ several other minor features of benefit in reload safety verifications. The North Anna Implementation was approved by the NRC in June, 1989 (8) for both North Anna units following approval of the COBRA/WRB-1 topical report (3). The present report extends this NRC-approved methodology to Surry.

Conversion to the Statistical DNBR Methodology (1) provides a DNBR margin gain of approximately 10-15% for SIF/WRB-1. This margin can be used to offset a relaxation in key accident analysis parameters which are verified for each reload.

3.2 Review of the VEP-NE-2-A Methodology

The Statistical DNBR Evaluation Methodology is employed to determine a revised DNBR limit. This new limit combines the correlation uncertainty with the DNBR sensitivities of uncertainties to key DNBR analysis input parameters. Transient analysis with the revised Methodology does not require that the uncertainties be applied in the initial conditions; instead, nominal values may be used.

The Statistical DNBR Limit is developed by means of a Monte Carlo process. The variation of actual operating conditions about nominal statepoints due to parameter measurement and other key DNB uncertainties is modelled with a random number generator-based algorithm. This algorithm produces thousands of random statepoints at each nominal statepoint. The random statepoints are then supplied to Virginia Power's core thermal-hydraulics code, COBRA (4), which calculates the minimum DNBR. Each DNBR is randomized by a correlation uncertainty factor as described in Reference (1). The standard deviation of the resultant DNBR distribution is increased by a small sample correction factor to obtain its 95% upper confidence limit, thereafter being combined Root-Sum-Square with code and model uncertainties to obtain the total DNBR standard deviation. The Statistical DNBR Limit (SDL) is then

$$SDL = 1 + 1.645 * s(total) \qquad (3.2\text{-}1)$$

in which the 1.645 multiplier is the z-value for one-sided 95% probability of a normal distribution. This SDL thus provides 95/95 peak rod protection from DNB.

As an additional criterion, the SDL is tested to determine the full core DNB probability when the SDL is reached by the peak rod. This process is performed by summing the DNB probability of each rod in the core, using a bounding rod power census curve and the DNBR sensitivity to rod power. In order to ensure that at least 99.9% of the core avoids DNB at all times, the SDL is increased to reduce the full core DNB probability if it is necessary to do so.

The Monte Carlo implementation analysis is described in the following section.

3.3 Implementation Analysis

3.3.1 Uncertainty Analysis

Consistent with the Reference (1) topical report, the uncertainties in vessel average temperature (Tavg), pressurizer pressure, core thermal power, vessel mass flow, core bypass flow, the nuclear enthalpy rise factor, and the engineering enthalpy rise uncertainty factor were selected for inclusion in the statistical DNBR methodology implementation analysis as statistically treated parameters. The magnitudes and functional forms of the statistically treated uncertainties were derived in a rigorous analysis of plant hardware and measurement/calibration procedures.

The uncertainties for Tavg, pressure, core thermal power and vessel flow were quantified in Reference (5). This analysis quantified all sensor, rack, and other components of a total uncertainty and combined them in a manner consistent with their relative dependence or independence. Total uncertainties were quantified at a 2σ level, corresponding to two-sided 95% probability, as

Vessel Tavg	± 3.0°F
Pressurizer Pressure	± 21.0 psia
Thermal Power	± 2.0%
Vessel Mass Flow	± 2.86%

A DNB Parameter Surveillance requirement is being proposed as a new Technical Specification (T.S. 3.12.F) for the implementation of the Statistical DNBR Evaluation Methodology at Surry. This new specification ensures that the RCS Tavg, pressurizer pressure, and RCS flow are maintained within the range of values assumed in safety analysis. In order that the station operations staff need not be concerned with applying uncertainties associated with the measured parameters defined by this new specification, the measurement uncertainties for Tavg and pressure presented above were statistically combined with an uncertainty representing the "control deadband" about the nominal parameter value. The uncertainties associated with the control deadband were assumed to be uniformly distributed, and were assigned values well in excess of the actual deadbands of the plant control systems.

No additional uncertainty need be combined with the flow measurement uncertainty to account for parameter surveillance concerns, since all statistical DNBR analyses assume the Technical Specification minimum measured flow, and the flow measurement uncertainty is factored into the statistical DNBR limit.

The total uncertainties for Tavg, pressure, core thermal power and vessel flow were increased beyond their calculated values to obtain the numbers actually used in the Monte Carlo calculations. This was done as a conservative measure, and to provide margin for future changes to the plant configuration.

The nuclear enthalpy-rise factor uncertainty was quantified in a Virginia Power analysis of 1236 measurement/prediction data points. An FΔh measurement uncertainty standard deviation of 1.9% bounded the actual values from several recent cycles for both units. The data normality was also verified. A standard deviation of 2.2% was actually employed in the Monte Carlo calculations.

Total core bypass flow consists of separate flow paths through the thimble tubes, direct leakage to the outlet nozzle, baffle joint leakage flow, upper head spray flow and core-baffle gap flow. These five components were each quantified based on the current Surry core configuration, their uncertainties conservatively modelled, and the flows and uncertainties totalled. The Monte Carlo analysis ultimately employed a bypass flow of 4.0% with an uncertainty of 1.5%, both of which are larger than the actual calculated values. The implementation analysis assumed that the probability was uniformly distributed. In addition, no credit was taken for independence of any of the bypass flow uncertainties.

The engineering enthalpy rise uncertainty factor consists primarily of the uncertainty in hot channel power and flow. These factors were quantified by means of a closed-channel calculation, in which bounding values of high hot channel power and low flow were employed. A uniformly distributed 2% uncertainty was found to conservatively bound the results.

3.3.2 Monte Carlo Calculations

The Monte Carlo analysis itself consisted of nine sets of 2000 calculations performed over the full range of normal operation and anticipated transient conditions. These conditions spanned the pressure range between the high and low trip setpoints, inlet temperatures between normal operation and a maximum heatup, powers up to the 118% overpower limit and a bounding low flow event. The DNBR standard deviation at each Nominal Statepoint was augmented by the correlation uncertainty, small sample correction factor, and the code uncertainty to obtain a total DNBR standard deviation. The peak rod limit was calculated by Equation (3.2-1) to be 1.25 for SIF/WRB-1. The Monte Carlo Statepoint analysis is summarized in Table 3.3-1.

The normality of the DNBR distribution at the limiting SIF/WRB-1 nominal statepoint (Statepoint E) was verified, assuring an appropriate estimate of the lower tail DNB probability.

3.3.3 Model Error Term

The more detailed 19-channel production model (4) was used in the development of the SDL for Surry. This is in contrast to the North Anna implementation analysis (2) which employed a 6-channel model to develop the SDL; 6-channel to 25-channel benchmark calculations were performed to develop a model error. At the expense of additional computer time, the 19-channel model was used to resolve local subchannel fluid conditions to an equivalent degree of detail as the models employed in the qualification of the WRB-1 correlation for use in the COBRA thermal-hydraulics code (3). As such, it is concluded that no correction for model error is necessary, and that the Model Error Term may be set to zero for the calculation of the total DNBR standard deviation.

Models with fewer than 19 channels may be employed for future sensitivity studies which require only assessments of relative DNBR. Such models have been properly benchmarked to their 19-channel counterparts to ensure their accuracy for this intended usage.

3.3.4 Full Core DNB Probability Summation

After the development of the peak rod 95/95 DNBR limits, the data statistics were used to determine the number of rods expected in DNB. The DNB probability summation for SIF/WRB-1 is summarized in Table 3.3-2. As may be seen, it was necessary to increase the the 95/95 peak pin limit to 1.27 to meet the full core 99.9% criterion. The full core DNB probability summation will be reevaluated on a reload basis to verify the applicability of the conservative fuel rod census ($F\Delta h$ vs. % of core) used in the implementation analysis.

3.3.5 Applicability of Methodology

It is necessary to demonstrate that this Methodology is valid at any statepoint in the intended range of application. To this end, Figure 3.3-1 is presented. This figure is a plot of the DNBR standard deviation (parameter uncertainties only) versus the pressure at which each was derived. The standard deviation is observed to be a function of pressure. Linear regression was employed to correlate s(P), which is a second order function for SIF/WRB-1 with high R^2. Afterward, the residuals were plotted as a function of temperature, power and flow. No trends were observed in the residual plots. Furthermore, the limiting statepoint (Statepoint E) is observed to be in the range of the expected maximum standard deviation. This substantiates the fact that the DNBR standard deviation has been conservatively maximized for any conceivable Condition I, Condition II or low flow DNB event.

The Statistical DNBR Evaluation Methodology may be applied to all Condition I and II DNB events, and to the Loss of Flow analysis. The accidents to which the Methodology is applicable are listed in Table 3.3-3. The range of application is consistent with the methodology as applied to North Anna (2). This Methodology will not be applied to accidents which begin from zero power where the $F\Delta h$ uncertainty is higher.

The Statistical DNB Methodology provides analytical margin by permitting transient analyses to be initiated from nominal operating conditions, and by allowing core thermal limits to be generated without the application of the $F\Delta h(e)$, $F\Delta h(N)$, and bypass flow uncertainties. These uncertainties are applied statistically into the DNBR limit to which the core thermal limits are calculated. With the exception of the Loss of Flow Accident reanalysis (Section 5.0), key analysis uncertainties continue to be applied to the initial conditions of accidents supporting a 1.62 $F\Delta h$. (See Section 4.0.) This treatment of uncertainties is more conservative than statistically folding uncertainties into the DNBR limit.

Table 3.3-1
SIF/WRB-1 Monte Carlo Analysis Summary

State-point	Power (%)	Inlet Temperature (°F)	Pressure (psia)	Flow (%)	DNBR Mean* (-)	Standard Deviation* (-)
A	118	582.8	2400	100	1.239	0.0797
B	104.3	604.7	2400	100	1.239	0.0789
C	118	574.7	2250	100	1.240	0.0826
D	107.7	590.8	2250	100	1.241	0.0814
E	118	560.5	2000	100	1.241	0.0841
F	114.0	566.8	2000	100	1.242	0.0836
G	118	553.6	1875	100	1.242	0.0812
H	100	541.3	2250	57.6	1.240	0.0790
I	118	566.1	2100	100	1.246	0.0816

* These are the DNBR mean and standard deviation reflecting the impact of parameter uncertainties only.

Table 3.3-2

Core-wide DNB Probability Summation
(at a 1.27 SIF/WRB-1 SDL)

State-point	Rods in DNB (% of core)
A	< 0.07%
B	< 0.07%
C	< 0.08%
D	< 0.07%
E	< 0.08%
F	< 0.06%
G	< 0.07%
H	< 0.06%
I	< 0.08%

Table 3.3-3
Applicability of Statistical DNBR Evaluation Methodology
to Surry Accident Analyses

FSAR Section	Accident	VEP-NE-2-A Applicable?
14.2.1	Rod Withdrawal from Subcritical	No
14.2.2	Rod Withdrawal at Power	Yes
14.2.3	Malpositioned Part Length Control Rod	n/a
14.2.4	Control Rod Drop/Misalignment	Yes
14.2.5	CVCS Malfunction	No (non-DNB)
14.2.6	Startup of Inactive Loop**	Yes*
14.2.7	Excessive Heat Removal	Yes
14.2.8	Excessive Load Increase	Yes
14.2.9.1	Complete Loss of Flow	Yes
14.2.9.2	Locked Rotor	No
14.2.10	Loss of Load	Yes
14.2.11	Loss of Feedwater	No (non-DNB)
14.2.12	Loss of AC Power	No (non-DNB)
14.2.13	Turbine Generator Overspeed	No (non-DNB)
14.3.1	Steam Generator Tube Rupture	No (non-DNB)
14.3.2	Main Steamline Break	No
14.3.3	Control Rod Ejection	No (non-DNB)

*During power operation only

**N-1 operation is presently forbidden by the Technical Specifications

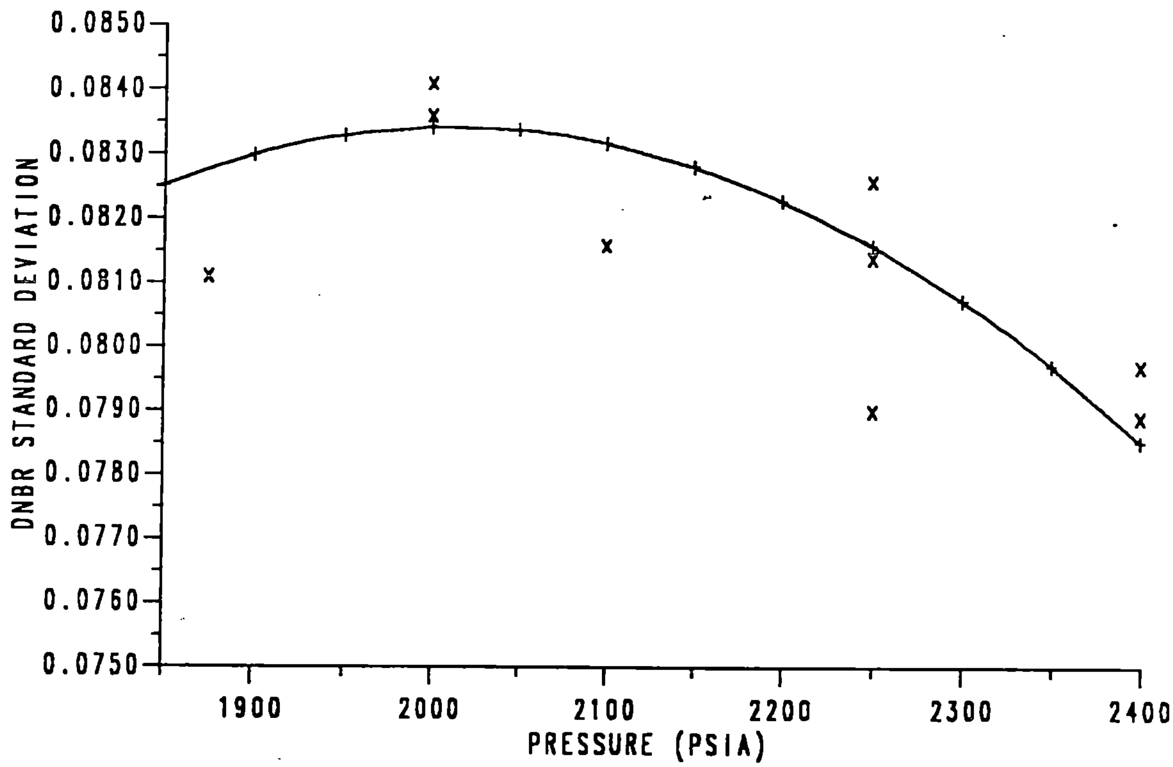


Figure 3.3-1 - SIF/WRB-1 DNBR Standard Deviation vs. Pressure

3.4 Revised SIF/WRB-1/Statistical Retained DNBR Margin

For the Statistical DNBR Methodology implementation, all retained margin was defined as a penalty upon the design DNBR limit; i.e., instead of working with the allowable Statistical DNBR Limit, a higher design limit is used. The difference is the retained DNBR margin (M):

$$\text{Design DNBR Limit} = \frac{SDL}{1 - M} \tag{3.4-1}$$

For SIF, the limit and margin are

Fuel/Correlation	SDL	Design Limit	Retained Margin
SIF/WRB-1	1.27	1.46	13.0%

This method of defining retained DNBR margin is preferable because all of the margin is found in a single, clearly defined location. North Anna's retained margin is similarly defined. The retained DNBR margin can be used to offset generic DNBR penalties which are difficult to model mechanistically in the DNBR analysis calculations. These include the rod bow penalty and the SIF/LOPAR transition core penalty.

The reload thermal-hydraulics evaluation prepared as part of the reload safety analysis process will present tables and descriptions of retained margin and applicable penalties. Retained margin will be tracked separately for LOPAR/Deterministic, SIF/Deterministic, and SIF/Statistical.

4.0 FΔh INCREASE

4.1 INTRODUCTION

The Surry Technical Specifications currently limit the full power radial peaking factor to 1.55. This limit is met by designing the Surry reload cores to no more than a 1.435 peak, which is 8% less than the 1.55 measured FΔh limit to account for uncertainties (4% calculational uncertainty, 4% measurement uncertainty). Virginia Power is proposing to increase the safety analysis FΔh from 1.55 to 1.62, thus increasing the core design FΔh from 1.435 to 1.50.

An increased FΔh limit is needed to accommodate increased radial power factors which will result from the installation of flux suppression inserts (FSI) in Surry Unit 1. The FSI's reduce the peripheral core power and, hence, the fast neutron flux near vessel locations which are experiencing neutron irradiation embrittlement. In order to achieve the target end-of-life fluence assumed in FSI implementation analyses, the FSI's are scheduled for installation in the S1C13 reload core.

An increase in the Technical Specification FΔh limit requires reanalyses in several areas. The radial factor is a key assumption in both the large break and small break Loss of Coolant Accident (LOCA) analyses, as well as in most non-LOCA analyses. Fortunately, margin exists in the Surry accident analyses in each of these areas which can absorb the impact of an FΔh increase. This margin is available through improved modelling and the absorption of excess conservatism in previous analyses.

The subsections which follow describe the LOCA and non-LOCA analyses and evaluations which have been performed to support increasing the safety analysis $F\Delta h$ to 1.62 for Surry. Although future reloads will contain no fresh LOPAR fuel, analyses and evaluations have been performed both for LOPAR fuel and for SIF. LOPAR is included so that previously burned assemblies may be incorporated into future reload core designs.

4.2 NON-LOCA ACCIDENT ANALYSES

The non-LOCA evaluation of the proposed FΔh increase is performed by establishing protection setpoints which will ensure that the reactor core will not enter into Departure from Nucleate Boiling (DNB) with 95% probability, at a 95% confidence level, during normal operation and all anticipated transients as listed in Chapter 14 of the Surry UFSAR. The DNB criterion is met by verifying that the plant will not violate its DNB Ratio (DNBR) limit during these events for both fuel types.

Past thermal-hydraulic calculations for the Surry Power Station have included the LOPAR retained margin which is itemized in Table 4.2-1. This margin is a result of several pieces of the current LOPAR DNBR analysis methodology which are more conservative than is required by safety analysis. Included in these factors is a 7% "Fuel Densification Spike," which is applied as a 0.93 multiplier on all calculated DNBR's. This factor was originally used to account for heat flux spiking due to fuel densification, and later to offset large rod bow penalties. The name has been retained for historical purposes.

More recently, Westinghouse developed a revised rod bow methodology (9). Virginia Power took credit for this reduction in Reference (10). Later, the NRC approved a reduction in the maximum applicable burnup for evaluating the rod bow penalty, which reduced the penalty even further (11). These reductions in the rod bow penalty effectively free up the retained margin in the Fuel Densification Spike for other uses.

Virginia Power has chosen to use the 7% retained DNBR margin to absorb the impact of increasing the FΔh design limit for LOPAR fuel. Sensitivity studies incorporating a deterministic treatment of key analysis uncertainties have shown that the DNBR reduction due to the proposed FΔh increase will be approximately 7%. As a result, the FΔh increase, coupled with the elimination of the 7% Fuel Densification Spike, will have little net impact upon the consequences of current safety analyses for LOPAR fuel.

The SIF DNBR margin to accommodate the FΔh increase is derived from a different source, however. The Fuel Densification Spike was not included in the licensing basis of the new SIF product (16); as a result, it is not available to absorb the DNBR impact of the FΔh increase for SIF. However, as will be seen in Section 4.3, sufficient margin exists in the SIF analyses to absorb the DNBR penalty outright. The reason for the greater SIF margin is that different DNB correlations are used to analyze the LOPAR and SIF products, and the margin which is inherent in the use of the SIF correlation (i.e., the WRB-1 correlation (3), (12)) is sufficient to fully absorb the impact of the FΔh increase without any DNBR limit violations. Further details will be provided in the following sections.

TABLE 4.2-1

CURRENT RETAINED DNBR MARGIN FOR 15x15 LOPAR WESTINGHOUSE FUEL

(DETERMINISTIC ACCIDENT DNBR WITH W-3)

COMPONENT	MARGIN	REFERENCE
1.30 DNBR limit used vs. 1.24 allowable	4.8%	(18)
Hot Channel Pitch Reduction	3.3%	(18)
0.19 TDC used vs. 0.038 allowable	3.0%	(18)
Fuel Densification Spike (0.93 multiplier)	7.0%	(18)
Total Retained DNBR Margin	18.1%	

4.3 VERIFICATION OF THE REACTOR PROTECTION SETPOINTS

The first step in establishing the required DNBR protection with the increased radial factor was to determine whether the existing Technical Specification Core Thermal Limits (CTL's) were limiting for the 1.62 $F\Delta h$ for LOPAR. Using the COBRA Surry models (4), statepoint calculations were performed to determine if the 7% retained margin would offset the $F\Delta h$ increase. These calculations showed that the 7% retained margin usually, but not always, offsets the DNBR penalty associated with the $F\Delta h$ increase; i.e., the existing CTL's were not bounding, although the violations were no more than slight.

Because the existing CTL's were not bounding, it was necessary to develop new core thermal limits which provided the required DNBR protection for LOPAR. The LOPAR calculations were performed with the W-3/L-grid DNB correlation and a 1.30 DNBR limit.

The core thermal limits were defined by the most restrictive of the following:

1. Vessel exit boiling
2. A 1.30 DNBR for LOPAR, as calculated with the W-3 correlation, assuming 4.5% core bypass flow
3. A 15% hot channel outlet quality in LOPAR assemblies, assuming 6% core bypass flow

The LOPAR DNBR calculations were performed with 4.5% bypass flow in the COBRA models consistent with the previous SIF implementation package (16). A DNBR penalty of 3.0% was quantified as being necessary to offset the bypass flow increase to 6% associated with thimble plug removal. This was extracted from the available LOPAR retained margin (16).

The new core thermal limits are plotted in Figure 4.3-1 and are included in the proposed revised Technical Specifications. They do not differ by very much from the current Technical Specifications CTL's; however, they are in some cases slightly less conservative, so that it is necessary to note that the new limits are not entirely bounded by the current limits. The major changes are a result of the impact of the higher 1.62 $F\Delta h$ on the 15% hot channel exit quality limit.

Conveniently, however, the new core thermal limits are still bounded by the current $OT\Delta T$ trip function. At the statepoints where the LOPAR/1.62 $F\Delta h$ CTL's were more limiting than the current Technical Specification CTL's, the $OT\Delta T$ margin was sufficiently large to absorb the difference. The current $OT\Delta T$ trip function, including the $F(\Delta I)$ function, therefore remains unchanged for LOPAR fuel with a 1.62 $F\Delta h$.

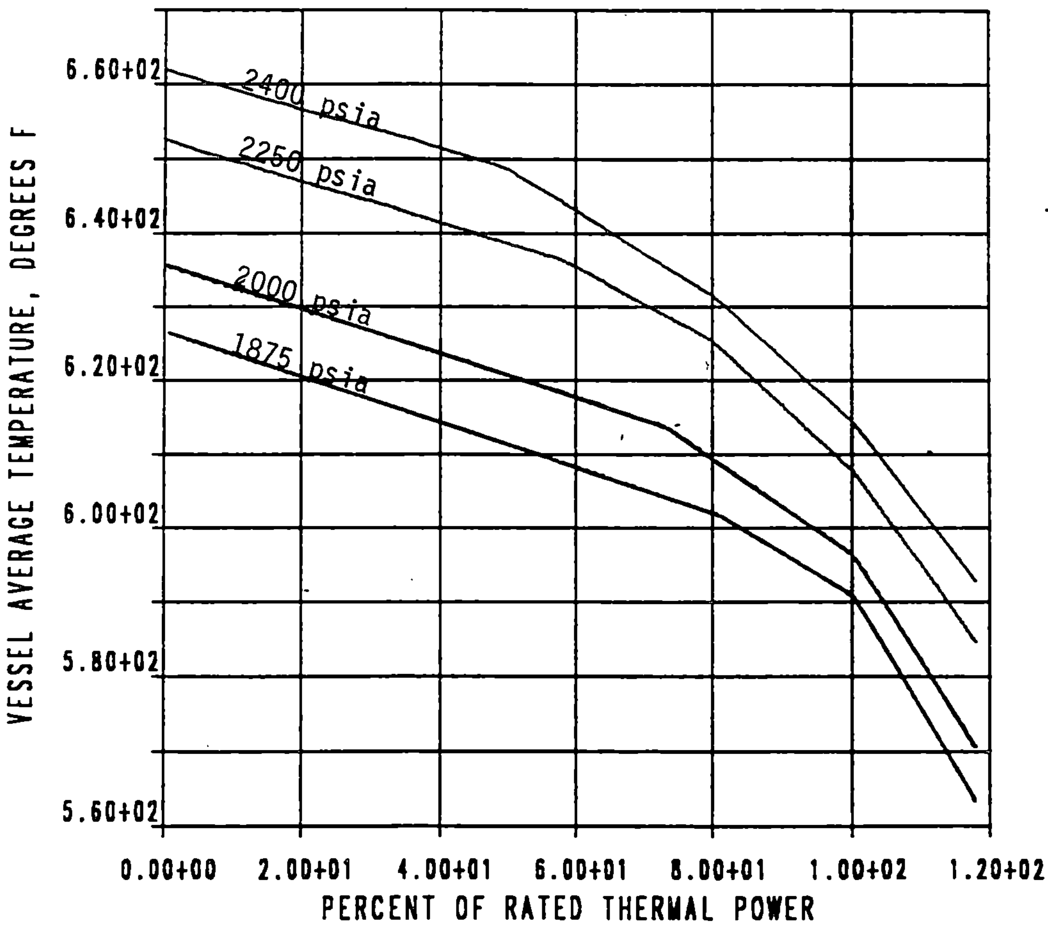
Similarly, the $OP\Delta T$ trip function was also shown to provide the required overpower protection, even with the increased radial power factor.

The next step was to verify the applicability of the current Technical Specification $OT\Delta T/OP\Delta T$ setpoints and $F(\Delta I)$ function for SIF. To accomplish this, new core thermal limits were developed which provided the required DNBR protection for SIF. The SIF calculations were performed utilizing a statistical treatment of key DNB parameter uncertainties, the

WRB-1 DNB correlation, and a 1.46 design DNBR limit. The core thermal limits were defined by the most restrictive of the following:

1. Vessel exit boiling
2. A 1.46 DNBR for SIF, as calculated with the WRB-1 correlation.
3. A 30% hot channel outlet quality in SIF assemblies

The OTΔT/OPΔT setpoints and F(ΔI) function developed for this set of CTL's have been shown to be bounded by the setpoints and F(ΔI) function currently in the Technical Specifications. Core thermal limit protection has, therefore, been shown to be provided by the existing protection setpoints and F(ΔI) function for both fuel types. As a result, no Technical Specifications change to either the OTΔT or OPΔT trip setpoints will be needed to support the increase in allowable FΔh to 1.62.



Revised Core Thermal Limits

Figure 4.3-1

4.4 SUMMARY OF CHAPTER 14 DNB EVENTS

The following summarizes the reanalyses or re-evaluations of all of the FSAR accidents to justify an increase in the FΔh safety analysis limit from 1.55 to 1.62. Unless otherwise noted, the following reanalyses or re-evaluations presume a deterministic treatment of key analysis uncertainties, and a core thermal power of 2441 MWth.

Rod Withdrawal from Subcritical (FSAR Section 14.2.1). The currently docketed analysis as described in the FSAR notes that the RWSC reaches a peak heat flux of 38% of nominal and an average fuel temperature of 763°F (the HFP value is 1475°F). This analysis was performed specifically for the LOPAR fuel type, at low flow, and included a 0.93 DNBR multiplier as a Fuel Densification penalty. The minimum DNBR was in excess of 2.2, well above W-3's 95/95 DNBR limit of 1.30. Because the major thermal-hydraulic distinction between SIF and LOPAR is the rod drop time, which does not impact the RWSC analysis results, the applicability of the LOPAR RWSC analysis was extended to SIF by Reference (17). The thermal-hydraulic effects of transition from LOPAR to SIF were fully accounted for by the application of a 3% DNBR penalty against retained DNBR margin. Since the 7% retained DNBR margin from the Fuel Densification Penalty approximately offsets the FΔh increase to 1.62, little if any change would be expected in the minimum DNBR. A RWSC with an FΔh increase of 4.5% would still be well above the W-3 95/95 DNBR limit of 1.30.

Rod Withdrawal at Power (FSAR Section 14.2.2). The RWAP is the most severe OTΔT DNB event, and is analyzed to verify the conservatism of the OTΔT setpoints. Since the current setpoints were shown to bound the Core Thermal Limits for both fuel types with a 1.62 FΔh, no RWAP reanalysis was required.

Malpositioned Part Length Control Rod (FSAR Section 14.2.3). No analysis was required, as part length rods have been removed from Surry.

Control Rod Drop/Misalignment (FSAR Section 14.2.4). Peak FΔh's for this event are calculated on a reload basis and compared to the limit value at which the DNBR limit is reached. This acceptance criterion will be verified every cycle, and there is presently more than enough FΔh margin in this event to accommodate the 4.5% FΔh increase.

CVCS Malfunction (FSAR Section 14.2.5). Boron dilution events at conditions other than at-power are evaluated on a reload basis to ensure that adequate time exists for operator response to correct an inadvertent boron dilution. This reload evaluation is independent of the design limit FΔh. The consequences of boron dilution event at power are identical to those of a slow RWAP, which was demonstrated above to require no reanalysis, so no CVCS malfunction reanalysis is required to support an FΔh increase. Meeting the operator response time criterion for the Boron Dilution at Power analysis is independent of the magnitude of FΔh.

Inactive Loop Startup (FSAR Section 14.2.6). N-1 loop operation is presently forbidden by the Surry Technical Specifications, so that no reanalysis of this event was necessary.

Excessive Heat Removal (FSAR Section 14.2.7). The currently docketed FSAR analysis shows the EHR to be an event of increasing or barely decreasing DNBR (3% in the latter case) which stabilizes without a reactor trip. Since the HFP DNBR with an increased FΔh will still be well above the applicable DNBR Limits, no reanalysis is necessary. Further, the OTΔT trip function remains available to provide any needed DNB protection for such full flow events.

Excessive Load Increase (FSAR Section 14.2.8). The currently docketed FSAR analysis shows a 10% DNBR decrease from the HFP value in the worst case, always remaining well above the limit value. In fact, the OTΔT trip was not even activated. Substantial margin to the DNBR limit remained. Similar margin will exist with the increased FΔh and the Statistical Methodology; further, the OTΔT trip function remains available to provide any needed DNB protection.

Loss of Flow (FSAR Section 14.2.9.1). The existing LOFA analysis applicable to both LOPAR and SIF included a 1.62 FΔh (17). However, because the LOFA is the most limiting DNB event to which the Statistical DNBR Evaluation Methodology may be applied, it has been reanalyzed for SIF utilizing a statistical treatment of uncertainties, a 1.56 FΔh, and a core power of 2546 MWth (approximately 104% of current nominal power). This reanalysis is described in Section 5.0.

Locked Rotor (FSAR Section 14.2.9.2). A revised detailed core thermal-hydraulics analysis was performed to verify that the FSAR locked rotor analysis assumption of no more than 5% failed fuel will continue to remain valid for a peak FΔh of 1.62 for both LOPAR and SIF. This result will be verified for each reload core. The overpressure criterion is not affected by the proposed FΔh increase and thus did not require reanalysis.

Loss of Load (FSAR Section 14.2.10). The FSAR discussion of the currently docketed analysis notes that the LOL is a non-decreasing DNBR event, so that an FΔh increase would not pose a threat to the DNBR limit.

Loss of Normal Feedwater (FSAR Section 14.2.11). The consequences of a LONF event are not dependent upon the magnitude of FΔH because the LONF is not a DNB-limited event. Therefore, the proposed FΔH increase has no impact upon the results of the licensing analysis.

Loss of AC Power (FSAR Section 14.2.12). The consequences of this event are not dependent upon the magnitude of FΔH because it is not a DNB-limited event. Therefore, the proposed FΔH increase has no impact upon the results of the licensing analysis.

Turbine Overspeed (FSAR Section 14.2.13). This analysis is not dependent upon FΔh, so that the proposed increase has no impact upon its consequences.

Steam Generator Tube Rupture (FSAR Section 14.3.1). This analysis is not dependent upon FΔh, so that the proposed increase has no impact upon its consequences.

Main Steamline Break (FSAR Section 14.3.2). MSLB statepoint radial factors are recalculated every cycle in order to assess reactivity and DNBR impact, so that the consequences of an FΔh increase will automatically be examined on a reload basis. An FΔh increase will not affect the transient statepoints.

Control Rod Ejection (FSAR Section 14.3.3). The Rod Ejection accident is dependent upon the 3-D factor FQ but not the radial factor FΔh, so that the proposed FΔh increase, with the FQ limit remaining unchanged, will have no impact upon the consequences of the Rod Ejection accident. FQ is verified as remaining within its limit on a reload basis.

Large Break LOCA (FSAR Section 14.5). The currently approved LBLOCA analysis (7) assumed a 1.62 FΔh (1.50 plus both calculational and measurement uncertainties), and as such already supports an FΔh increase.

Small Break LOCA. The Surry SBLOCA was reanalyzed as described in Section 6.0 of this report. The acceptance criteria of peak clad temperature and oxidation levels were not violated with an assumed FΔh of 1.65.

This review demonstrates that all of the necessary analyses and evaluations have been performed to support the proposed increase in FΔh to 1.62.

5.0 LOSS OF FLOW ACCIDENT REANALYSIS

5.1 Accident Description

A complete loss of forced reactor coolant flow may result from a simultaneous loss of electrical power to all three reactor coolant pumps. If the reactor is at power at the time of the accident, the immediate effect of a LOFA is a rapid increase in the coolant temperature. This increase could result in DNB with subsequent fuel damage if the reactor is not promptly tripped. Reactor protection is provided by either the pump underfrequency or undervoltage trip function.

5.2 Method of Analysis

The Virginia Power RETRAN models (6),(19) are used to perform reactor coolant system (RCS) transient analysis. The RETRAN models simulate the neutron kinetics, reactor coolant system, pressurizer, pressurizer relief and safety valves, pressurizer spray, steam generators and steam generator safety valves. The RETRAN code computes pertinent plant variables including temperature, pressure, and power level.

The Virginia Power COBRA models are used to perform a detailed thermal-hydraulic (T/H) analysis of the reactor core (4). COBRA solves the governing conservation and state equations to resolve the flow and energy fields within the reactor core itself. These results are used in turn to calculate the DNBR with the WRB-1 DNB correlations. COBRA can perform either steady state or transient DNBR analyses with conditions which have been supplied by the RETRAN code.

Key analysis details are listed in Table 5.2-1. All assumptions are consistent with or conservative with respect to those in the previously approved analysis. An increased Rated Thermal Power was assumed in anticipation of a possible submittal package seeking to increase this parameter in the future. An $F\Delta h$ of 1.56 was assumed, which is the proposed safety analysis limit for use in statistical DNBR analyses. Both the underfrequency and the undervoltage trip events were analyzed. A +3 pcm/^{o}F Moderator Temperature Coefficient was conservatively assumed although the actual full power MTC will be zero or negative. Delay times of 0.6 second and 1.5 seconds were assumed for the underfrequency and undervoltage trips respectively.

5.3 Results and Conclusions

The underfrequency trip LOFA was found to be the most limiting event for both fuel types. Transient power and flow are shown in Figures 5.3-1 and 5.3-2 respectively. The transient DNBR's for SIF are shown in Figure 5.3-3. Although a multiplier of 0.8 was applied to the transient DNBR's as a contingency factor, there was no violation of the Statistical DNBR limit.

TABLE 5.2-1
KEY ANALYSIS ASSUMPTIONS DETAILS

Initial Conditions	
Power	2546.0 MWt
Average Temperature	573.0 °F
RCS Flow Rate	273,000 gpm
Pressure	2250 psia
$F_{\Delta h}$ at Rated Power	1.56
1.55-Cosine Axial Power Profile	
Reactor Kinetics	
Moderator Temperature Coeff.	+3.0 pcm/°F (BOC)
Doppler Temperature Coeff.	-1.0 pcm/°F (BOC)
Power Reactivity (pcm)	$-19.2Q + 0.03Q^2$ (most negative)
β	0.0072 (BOC)
ℓ^*	26.0 μsec
Trip Reactivity Shape	Surry Improved Fuel

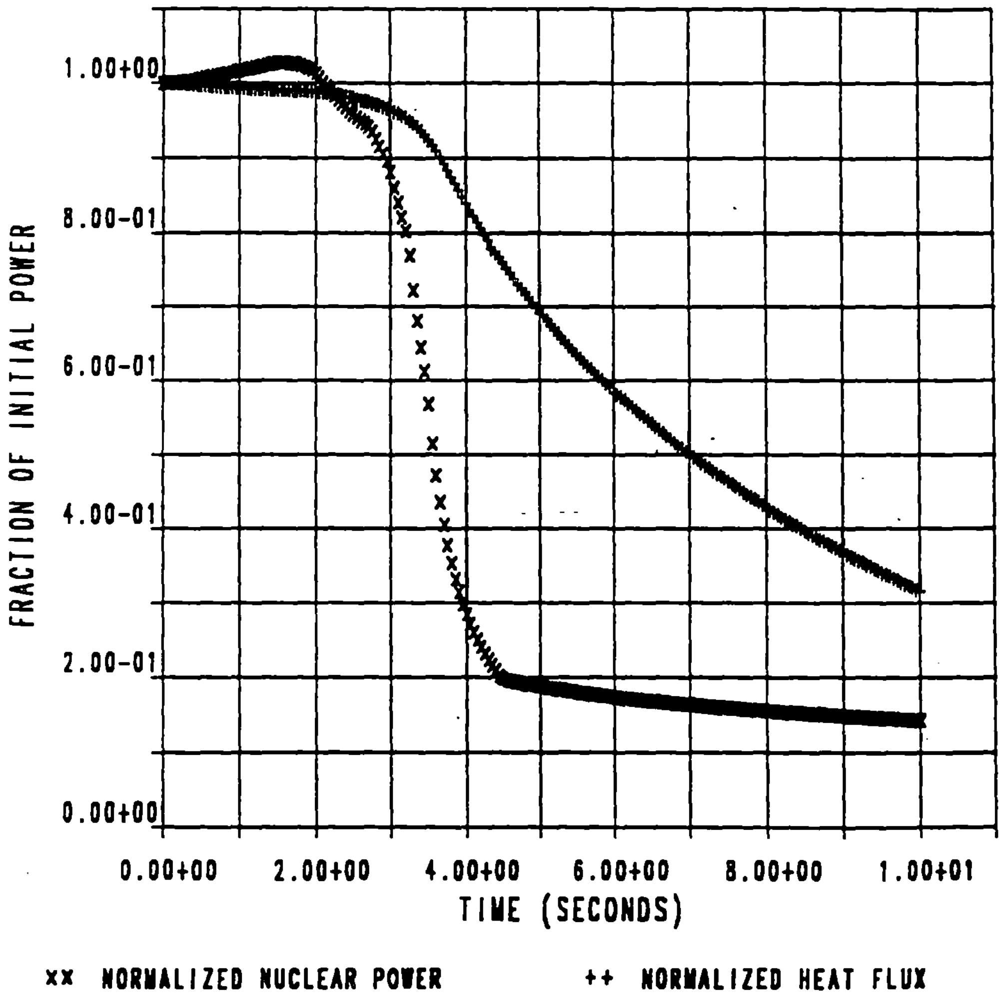


Figure 5.3-1 - LOFA Transient Power (Underfrequency Trip)

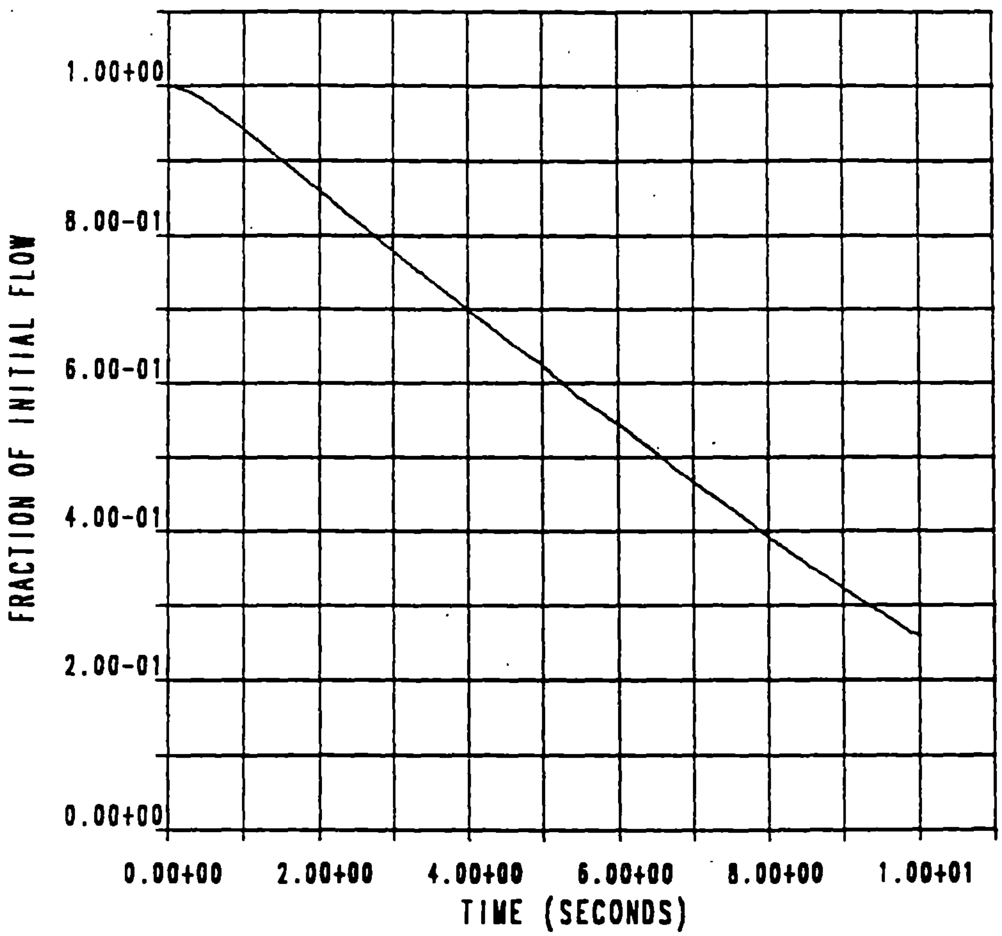


Figure 5.3-2 - LOFA Transient Flow (Underfrequency Trip)

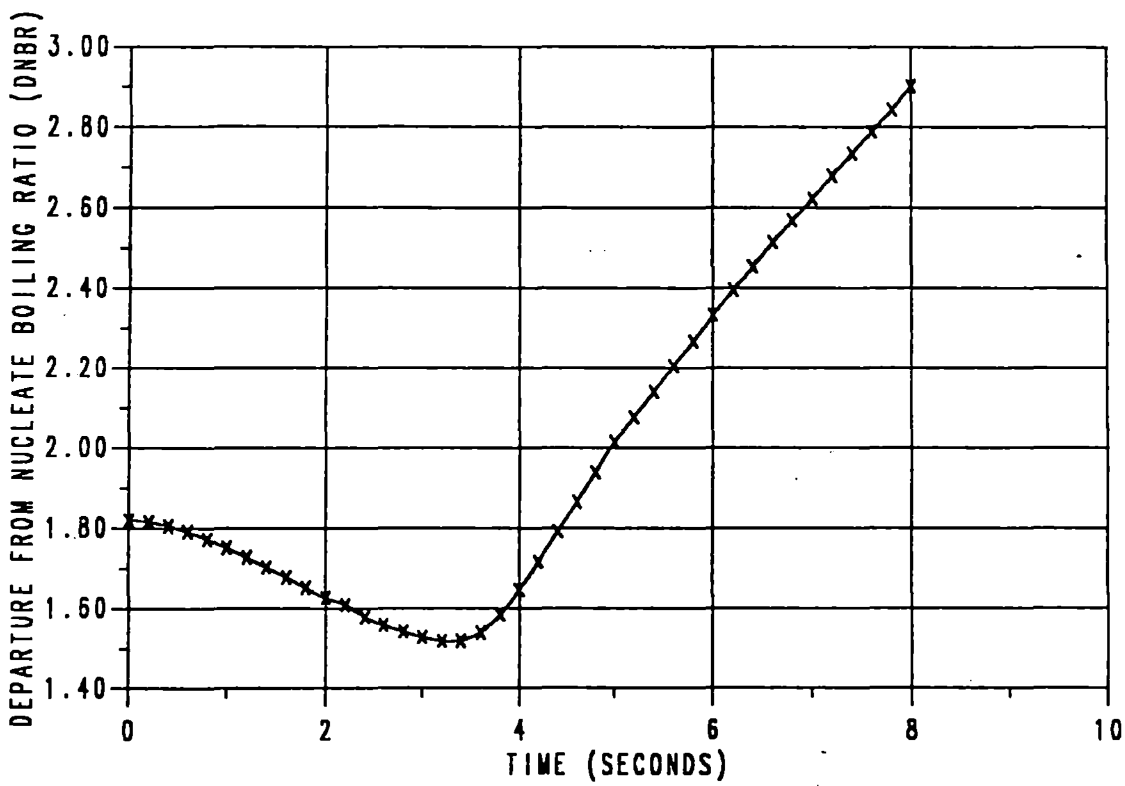


Figure 5.3-3 - LOFA Transient DNBR's (Underfrequency Trip)

6.0 SMALL BREAK LOCA REANALYSIS

6.1 GENERAL

A reanalysis of the Emergency Core Cooling System (ECCS) performance for the postulated small-break LOCA (SBLOCA) has been performed in compliance with Appendix K to 10 CFR 50. The results of this reanalysis are presented here, and are in compliance with 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Reactors." This analysis was performed with the NRC-approved NOTRUMP code (13) of the Westinghouse LOCA-ECCS evaluation model (14). The thermal behavior of the fuel was analyzed using the LOCTA-IV code (15). The analytical techniques used are in full compliance with 10 CFR 50, Appendix K.

As required by Appendix K of 10 CFR 50, certain conservative assumptions were made for the LOCA-ECCS analysis. The assumptions pertain to the conditions of the reactor and associated safety system equipment at the time that the LOCA is assumed to occur, and include such items as the core peaking factor and the performance of the Emergency Core Cooling System.

6.2 IDENTIFICATION OF CAUSES AND ACCIDENT DESCRIPTION

A LOCA can result from a rupture of the Reactor Coolant System (RCS) or of any line connected to that system up to the first isolation valve. The system boundaries considered in the LOCA analysis are defined in the UFSAR. Ruptures of small cross section will cause expulsion of the coolant at a rate that can be accommodated by the charging pumps. Breaks of greater size (up to 1 ft^2 area) are defined as small breaks, and are analyzed with the NOTRUMP computer code. A rupture in the Reactor Coolant System results in the discharge to the containment of reactor coolant and associated energy. The result of this discharge is a decrease in coolant pressure in the Reactor Coolant System and an increase in containment temperature and pressure. The reactor trip signal subsequently occurs when the pressurizer Low Pressure trip setpoint is reached. A safety injection system (SIS) signal is actuated when the Pressurizer Low Low Pressure setpoint is reached, activating the high head safety injection pumps. The SIS actuation and subsequent activation of the Emergency Core Cooling System, which results from the SIS signal, assumes the most limiting single failure of ECCS equipment.

Before the break occurs, the unit is in an equilibrium condition, i.e., the heat generated in the core is being removed via the secondary system. In the small break LOCA, the blowdown phase of the small break occurs over a long time period. Thus for a small break LOCA there are three characteristic stages: a gradual blowdown in which the decrease in water level is checked by the inventory replenishment associated with safety injection; core recovery; and long-term recirculation. The heat transfer between the reactor coolant system and the secondary system may be in either direction, depending on the relative temperature. For the case of continued heat addition to the secondary side, the secondary side pressure increases and the main safety valves may actuate to reduce the pressure. Makeup to the secondary side is automatically provided by the auxiliary feedwater system. Coincident with the safety injection signal, normal feedwater flow is stopped by closing the main feedwater control valves and tripping the main feedwater pumps. Emergency feedwater flow is initiated by starting the auxiliary feedwater pumps. The secondary side flow aids in the reduction of RCS pressure. When the reactor coolant system depressurizes to approximately 600 psia, the accumulators begin to inject borated water into the reactor coolant loops. The conservative assumption is then made that injected accumulator water spills in the broken loop and only the intact loops retain their accumulator water. This conservatism is again consistent with Appendix K of 10 CFR 50. Reflecting the loss of offsite power assumption, the reactor coolant pumps are assumed to be tripped at the time of reactor trip, and the effects of pump coastdown are included in the blowdown analysis.

6.3 ANALYSIS ASSUMPTIONS

As required by Appendix K of 10 CFR 50, certain conservative assumptions were made for the Small Break LOCA-ECCS analysis. The assumptions pertain to the conditions of the reactor and associated safety system equipment at the time that the LOCA is assumed to occur, and include such items as the core peaking factors, core decay heat and the performance of the Emergency Core Cooling System. Table 6.0-1 presents the values assumed for several key parameters in this analysis. Assumptions and initial operating conditions which reflect the requirements of Appendix K to 10CFR50 have been used in this analysis. These assumptions include:

- The break is located in the cold leg between the pump discharge and the vessel inlet.
- Minimum safeguards safety injection, including the assumption that safety injection flow in the broken loop is lost from the RCS, resulting in injection flow only to the intact loop cold legs.
- The accumulator in the broken loop also spills to containment.
- 120 percent of 1971 ANS decay heat is assumed following reactor trip.
- Initial power is 102% of the full core power, to account for the calorimetric uncertainty.

Several additional assumptions have been incorporated into the SBLOCA reanalysis described below. A core power of 2546 MWt was assumed, increased by 2% to account for calorimetric measurement uncertainty. The analysis also assumed the current Technical Specification maximum hot channel factor (FQ) value of 2.32. The inherent margin of the NOTRUMP ECCS evaluation model has been employed to accommodate these changes and to justify increasing the Technical Specifications limit for normalized FQ(z) in the upper half of the core. Finally, a maximum value of 1.65 has been assumed for the enthalpy rise hot channel factor (FΔh). This assumption, taken collectively with analogous assumptions in the large break LOCA analysis and core thermal limit generation, allows an increase in the FΔh Technical Specification limit.

The analysis was performed assuming a full core of the Surry Improved Fuel (SIF) assembly design. As described previously, the fresh fuel in reload cores for both Surry units is of the SIF design. Reload cores may also contain previously burned 15x15 STD Low Parasitic (LOPAR) fuel assemblies. As explained below, it is appropriate to model a mixed core containing both types of assemblies as a full core of SIF.

The only mechanism to cause a mixed core to have a greater calculated small break PCT than a full core of either fuel type is the possibility of flow redistribution caused by fuel assembly hydraulic resistance mismatch. The NOTRUMP evaluation model used to calculate core hydraulics during a small break has only one core channel. This is an acceptable modeling assumption, since the flow rate during this event is relatively low, which provides sufficient time to maintain flow equilibrium between fuel assemblies. Since such crossflow is not established during the small break event, mixed core hydraulic resistance mismatches are not a significant factor in the analysis.

The flow delivered by the high head safety injection system has been modelled assuming an imbalance between the flowrates in the three injection lines. This assumption increases the amount of flow assumed to be lost through the injection line on the broken loop, which minimizes the calculated flow delivered to the RCS. The effects of this assumption are included in the assumed injection flowrate, which is a function of RCS pressure. Flow testing and throttle valve adjustments (if necessary) are performed during each refueling outage to ensure that the actual system performance is bounded by the assumption in this analysis. The assumed imbalance is 40 gpm between the minimum and maximum flow branch lines, at the reference test RCS pressure of 0 psig.

6.4 ANALYSIS OF EFFECTS AND CONSEQUENCES

6.4.1 METHOD OF ANALYSIS

A small break LOCA analysis was performed using the NOTRUMP computer code following the methodology and the model delineated in WCAP-10079-P-A (13) and WCAP-10054-P-A (14). The NOTRUMP computer code is used for loss of coolant accidents due to small breaks less than one square foot. The code calculates the transient depressurization of the RCS as well as describing the mass and enthalpy of flow through the break.

NOTRUMP is a general one-dimensional network code consisting of a number of advanced features. Among these features are the calculation of thermal non-equilibrium in all fluid volumes, flow regime-dependent drift flux calculations with counter-current flooding limitations, mixture level tracking logic in multiple-stacked fluid nodes and regime-dependent heat transfer correlations. The NOTRUMP small break LOCA emergency core cooling system (ECCS) evaluation model was developed to determine the RCS response to design basis small break LOCAs and to address the NRC concerns expressed in NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants."

In NOTRUMP, the RCS is nodalized into volumes interconnected by flowpaths. The broken loop is modelled explicitly, with the intact loops lumped into a second loop. The transient behavior of the system is determined from the governing conservation equations of mass, energy and momentum applied throughout the system.

The use of NOTRUMP in the analysis involves, among other things, the representation of the reactor core as heated control volumes with an associated bubble rise model to permit a transient mixture height calculation. The multinode capability of the program enables an explicit and detailed spatial representation of various system components. In particular, it enables a proper calculation of the behavior of the loop seal during a loss-of-coolant accident.

The peak clad temperature in the core during a transient is calculated by utilizing the Westinghouse LOCTA-IV code (15) for a small break analysis. The transient thermal hydraulic NOTRUMP code writes data to a file for the LOCTA-IV code. The clad thermal analysis code uses the RCS pressure, core mixture level, normalized core power, and core exit mass flow rate from the thermal hydraulic code NOTRUMP as input.

The variation of assumed Safety Injection flow with break size is shown in Figure 6.0-1. The flow is smaller for the 6 inch case, to account for the possibility that the break (which in this case, has greater area than the SI line cross section) represents the complete severance of a safety injection line. This is assumed to cause the line to become detached from the RCS and spill its flow directly into the containment.

This analysis also employed a revised K(z) envelope, the hot channel factor normalized operating curve shown in Figure 6.0-2. K(z) is a multiplier on the allowable 3-D peaking factor FQ, and by nature cannot exceed 1.0. The current Surry K(z) curve consists of three linear, continuous segments, of which the uppermost is defined by small break LOCA concerns. However, in this small break LOCA analysis, the uppermost line segment has been eliminated by the use of a revised hot rod axial power shape (shown in Figure 6.0-3) in the LOCTA-IV code. This power shape has been chosen from a generic database of potential shapes achievable during power operation by assessing the characteristics which yield limiting small break LOCA results. The selected shape has been identified as the most limiting within the bounds of the proposed K(z) curve. For each reload core, the revised Technical Specifications K(z) curve will be confirmed to provide a limit which defines acceptable bounds for axial power shapes. This confirmation will be performed by assessing the impact of achievable shapes upon all transients for which axial power shape is a key analysis input.

6.4.2 ANALYSIS RESULTS

For this analysis, cases were run assuming 2 inch, 3 inch, 4 inch and 6 inch effective diameter cold leg breaks. Results of key parameters for the cases analyzed are presented in Figures 6.0-4A through 6.0-10D. Table 6.0-2 presents the time sequence of events, and Table 6.0-3 summarizes the peak clad temperature for each case analyzed. The 3 inch cold leg break was found to be the most limiting break size for a small break LOCA from the present analysis. The analysis resulted in a limiting peak clad temperature of 1851.8°F, a maximum local cladding oxidation level of

3.20%, and a total core metal-water reaction of less than 0.3%. The attached figures show the following:

- Pressurizer Pressure - Figures 6.0-4A through 6.0-4D show the calculated pressure for the different break sizes.
- Core Mixture Level - Figures 6.0-5A through 6.0-5D show that the core mixture level decreases, accompanied by the RCS depressurization, until the combined rate of the Safety Injection and the Accumulator Injection exceeds the break flow.
- Pumped SI Flow - Figures 6.0-6A through 6.0-6D show the sum of pumped safety injection flow to the intact loops.
- Core Exit Vapor Flow - Figures 6.0-7A through 6.0-7D show the core exit vapor flow.
- Hot Assembly Fluid Temperature - The fluid temperature in the hot assembly peaks at the same time as the clad temperature, with approximately the same magnitude, and is shown in Figures 6.0-8A through 6.0-8D.
- Hot Assembly Heat Transfer Coefficient - Figures 6.0-9A through 6.0-9D show the calculated heat transfer coefficient in the hot assembly.
- Peak Clad Temperature - Figures 6.0-10A through 6.0-10D show the calculated hot-spot clad temperature transient. The peak clad temperature for the limiting 3 inch break size is 1851.8°F at the 11.75 foot core elevation.

6.5 SMALL BREAK LOCA ANALYSIS CONCLUSIONS

The fuel clad heatup summary in Table 6.0-3 presents results that are well within the acceptance criteria specified by 10 CFR 50.46. The calculated peak clad temperature for the limiting 3 inch break is 1851.8°F, which is significantly less than the 2200°F limit. The maximum local metal water reaction is 3.20%, which is much less than the embrittlement limit of 17%. The total zirconium-water reaction is less than 0.3%, which is well below the 1% limit. The results show that the clad temperature transient has peaked and sufficiently stabilized while the core is still amenable to cooling. Consequently, it is concluded that the Surry ECCS will be capable of mitigating the effects of a small break LOCA with a maximum $F\Delta h$ of 1.65, even at a conservative core power of 2546 MWt. Although this increased power was assumed in the analysis, no Technical Specification change is being sought at this time for operation at this power.

TABLE 6.0-1

SIGNIFICANT INPUTS AND ASSUMPTIONS

SMALL BREAK LOCA ACCIDENT

Parameter	Value
Core Power (MWt), 102% of	2546
Total Peaking Factor, FQ	2.32
Core Enthalpy Rise Factor, FΔh	1.65
Fuel Enrichment (%)	4.1
Fuel Pellets	Chamfered
Fuel Assembly Array	15x15 SIF*
Accumulator Water Volume, (ft^3/accumulator)	1025
Accumulator Tank Volume, (ft^3/accumulator)	1450
Accumulator Gas Pressure, (psia)	580
Safety Injection Flow	Figure 6.0-1
Initial Loop Flow (lbm/sec)	9349.72
Vessel Inlet Temperature (°F)	541.54
Vessel Outlet Temperature (°F)	607.86
Reactor Coolant Pressure (psia)	2280
Steam Pressure (psia)	751.0
Steam Generator Tube Plugging (uniform)	15%
Low Pressurizer Pressure Setpoint (psia)	1840
Low-Low Pressurizer Pressure Setpoint (psia)	1715

* This analysis was performed assuming the SIF fuel product. It is also applicable to mixed cores of SIF and LOPAR fuel.

TABLE 6.0-2

SMALL BREAK LOCA TIME SEQUENCE OF EVENTS

Event	Time After Start of LOCA (sec) For Each Break Size (Effective Diameter)			
	2 inch	3 inch	4 inch	6 inch *
Break Opens	0.0	0.0	0.0	0.0
Reactor Trip Signal	30.06	13.04	7.67	4.86
Safety Injection Signal	45.71	18.63	10.2	6.54
Loop Seal Clearing	1047.	504.	262.	108.
Top of Core is Uncovered	1597.	839.	238	173. 1523.
Accumulator Injection Begins	N/A **	1261.	648.	274.
Peak Clad Temperature Occurs	3199.	1467.	720.	322.
Top of Core is Covered	N/A ***	N/A ***	3198.	458. 2472.

* The pair of values reported for core uncovery and recovery correspond to the two calculated periods of uncovery for this case.

** Transient case was terminated prior to anticipated accumulator injection.

*** Long-term RCS inventory recovery was established in the transient calculation prior to reaching this condition.

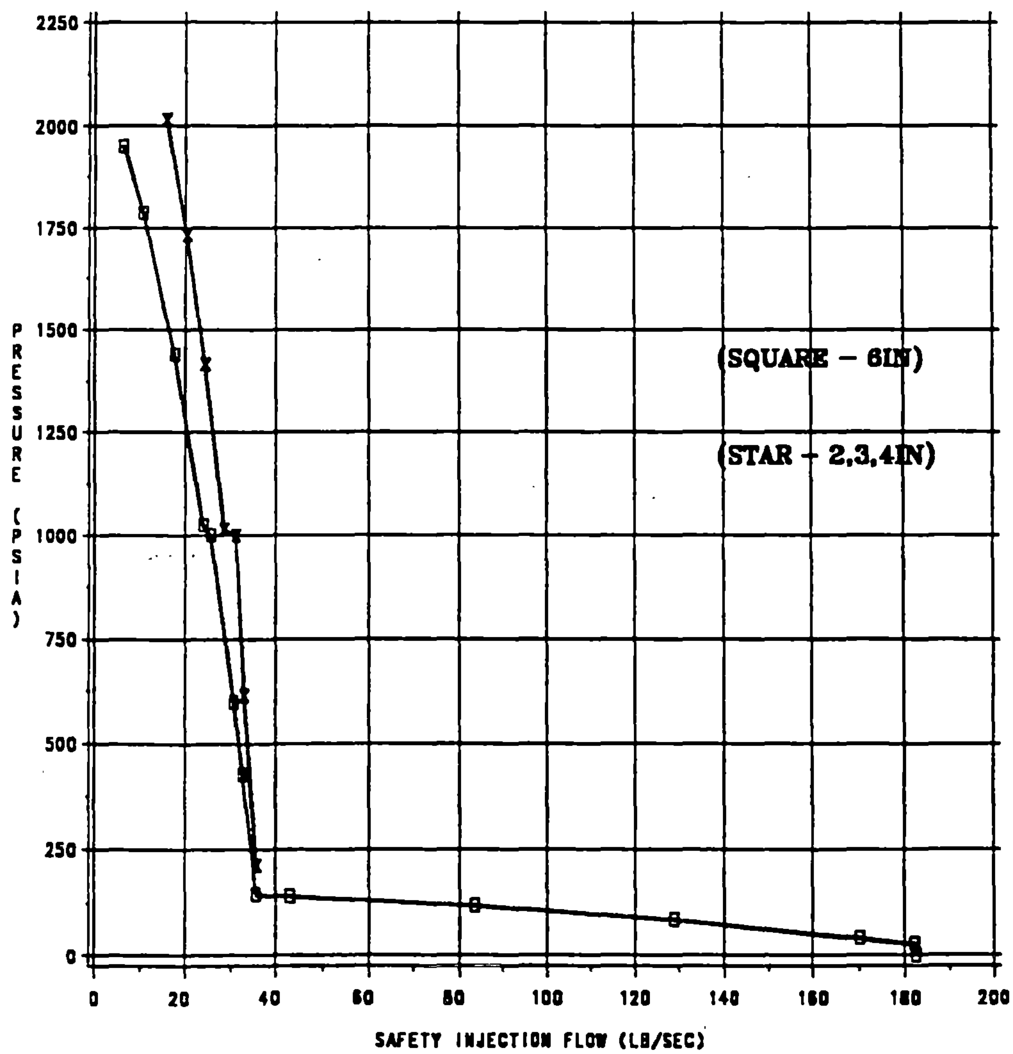
TABLE 6.0-3

SMALL BREAK LOCA RESULTS
FUEL CLADDING DATA

Parameter	Break Size (Effective Diameter)			
	2 inch	3 inch	4 inch	6 inch
Peak Clad Temperature (°F)	1252.5	1851.8	1399.5	1504.3
Peak Clad Temperature Location (ft)	11.75	11.75	11.25	10.75
Local Zr/H20 Reaction, Maximum (%)	0.30	3.20	0.19	0.21
Local Zr/H20 Reaction, Location (ft)	11.50	11.75	11.25	11.00
Total Zr/H20 Reaction (%)	< 0.3	< 0.3	< 0.3	< 0.3
Hot Rod Burst Time *	N/A	N/A	N/A	N/A
Hot Rod Burst Location *	N/A	N/A	N/A	N/A

* Burst was not calculated to occur.

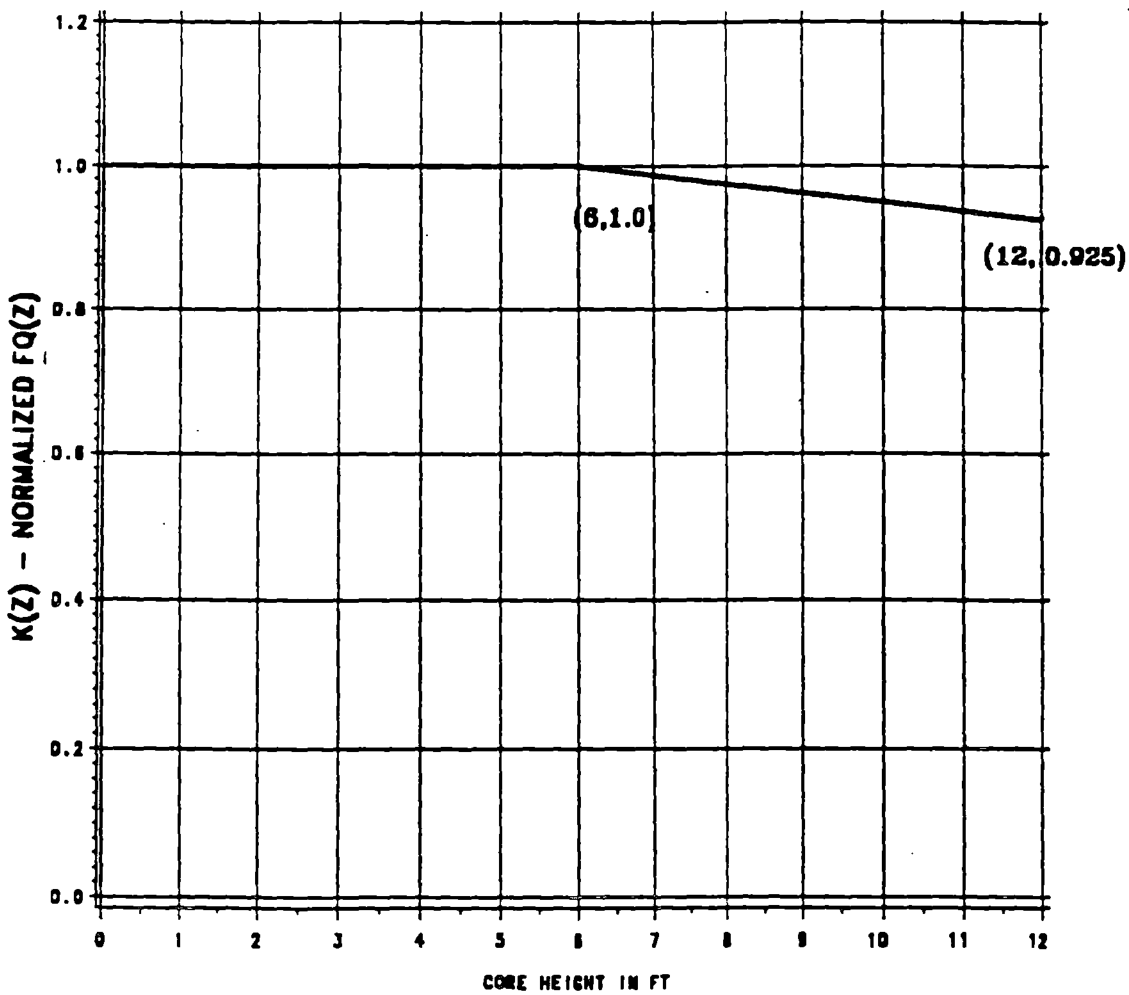
SURRY SMALL BREAK LOCA ANALYSIS
FDH=1.65 FQZ=2.32 SIF 15%SGTP



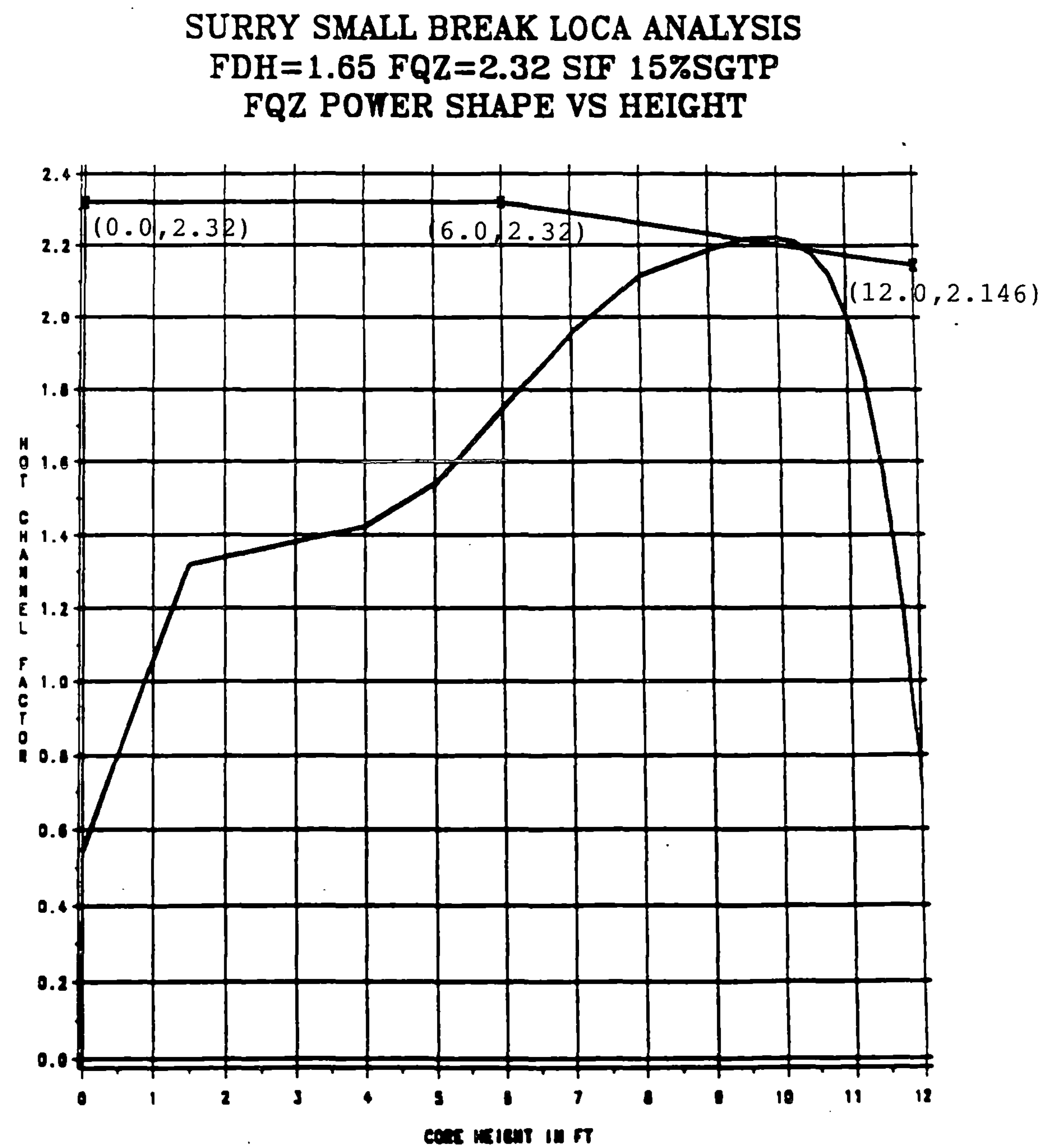
Safety Injection Flow (lb/sec)
FIGURE 6.0-1

TS FIGURE 3.12-8

HOT CHANNEL FACTOR NORMALIZED
OPERATING ENVELOPE
SURRY POWER STATION

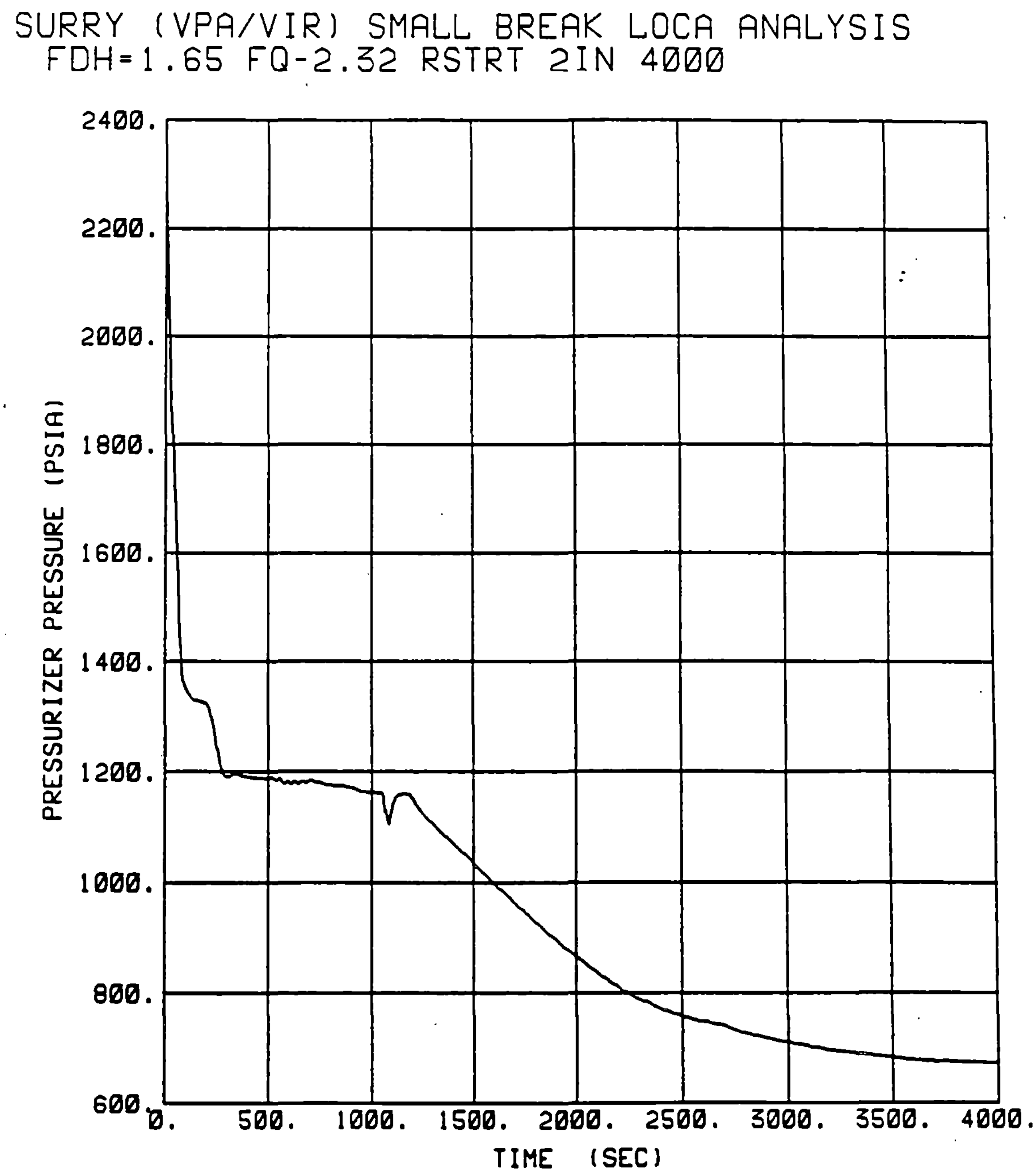


Hot Channel Factor Normalized, Operating Envelope
FIGURE 6.0-2

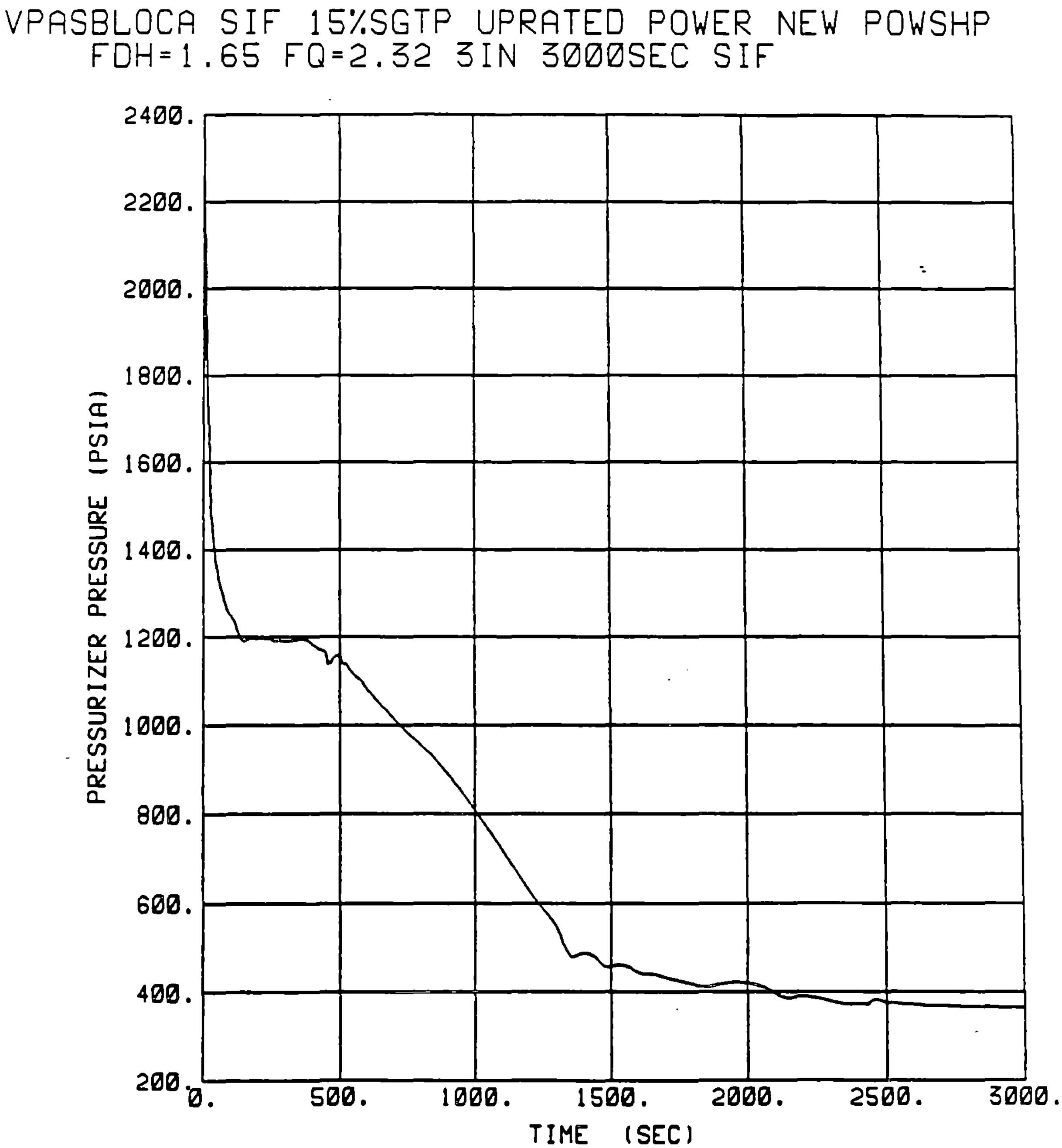


Note: The hot rod axial power shape slightly exceeds the K(z) limit. This is a conservative result of scaling calculations for the limiting power shape which require conservation of integrated power along the rod.

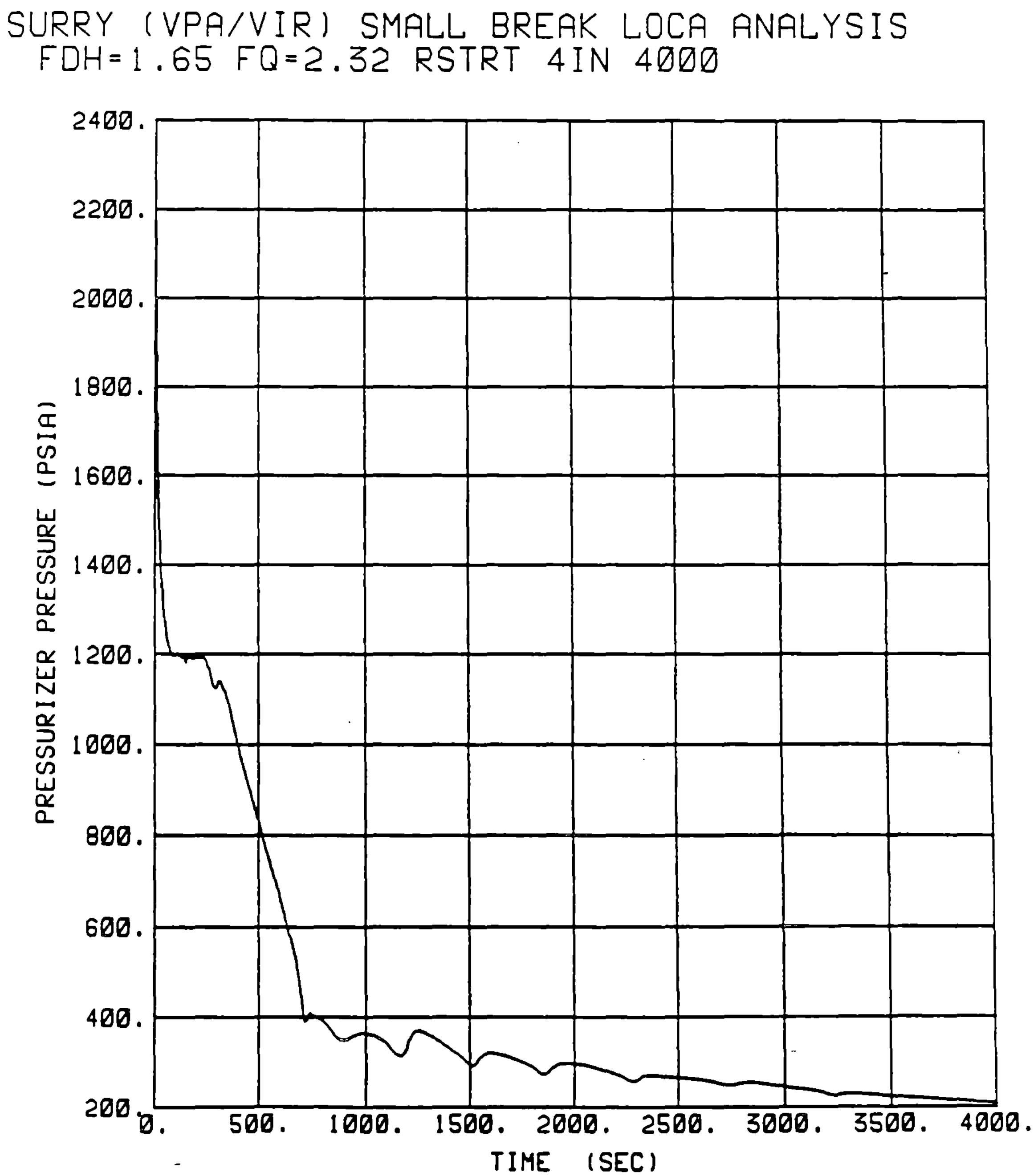
Hot Rod Power Shape Used In LOCTA-IV
FIGURE 6.0-3



Pressurizer Pressure (psia), 2 inch Break
FIGURE 6.0-4A

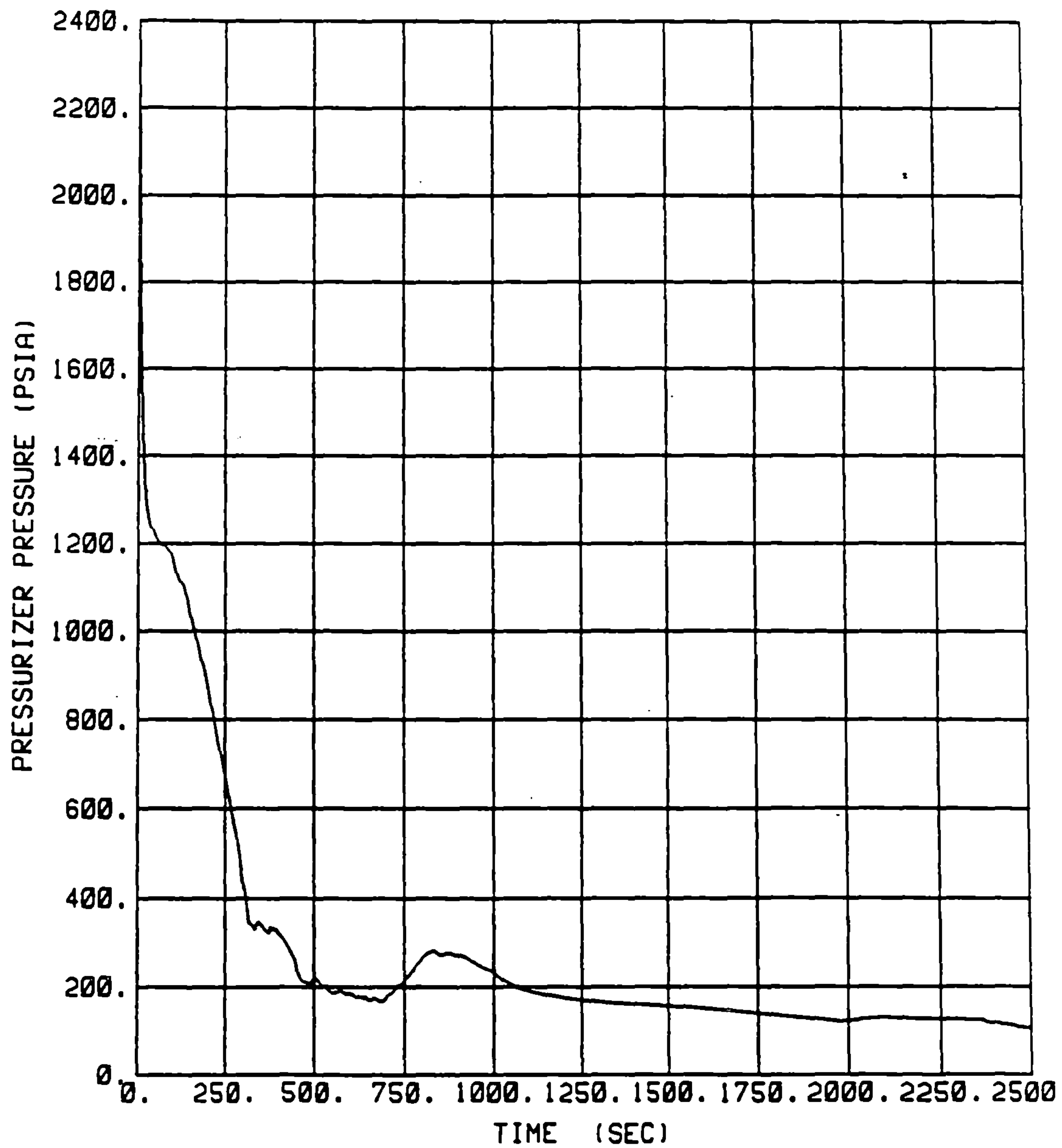


Pressurizer Pressure (psia), 3 inch Break
FIGURE 6.0-4B

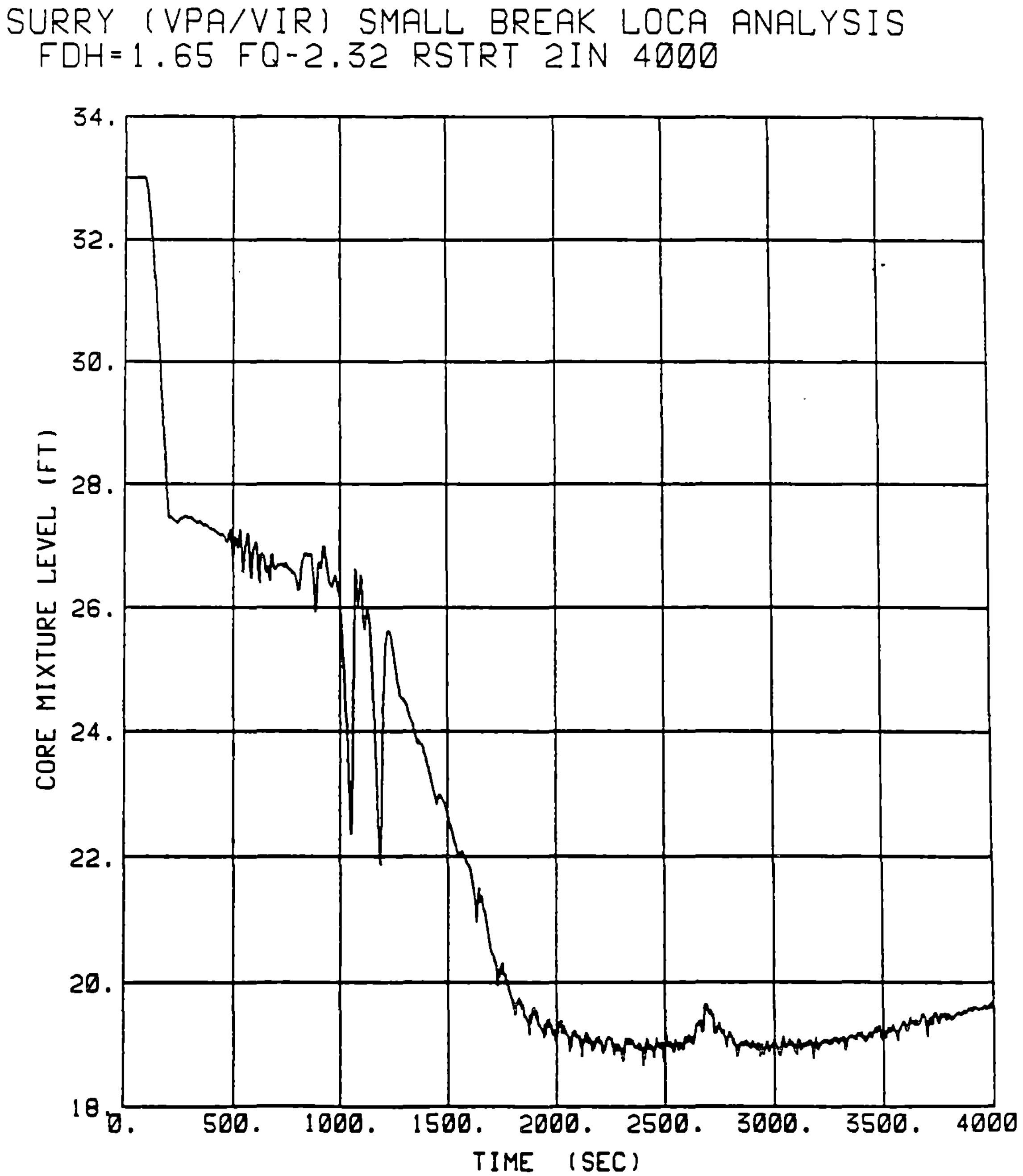


Pressurizer Pressure (psia), 4 inch Break
FIGURE 6.0-4C

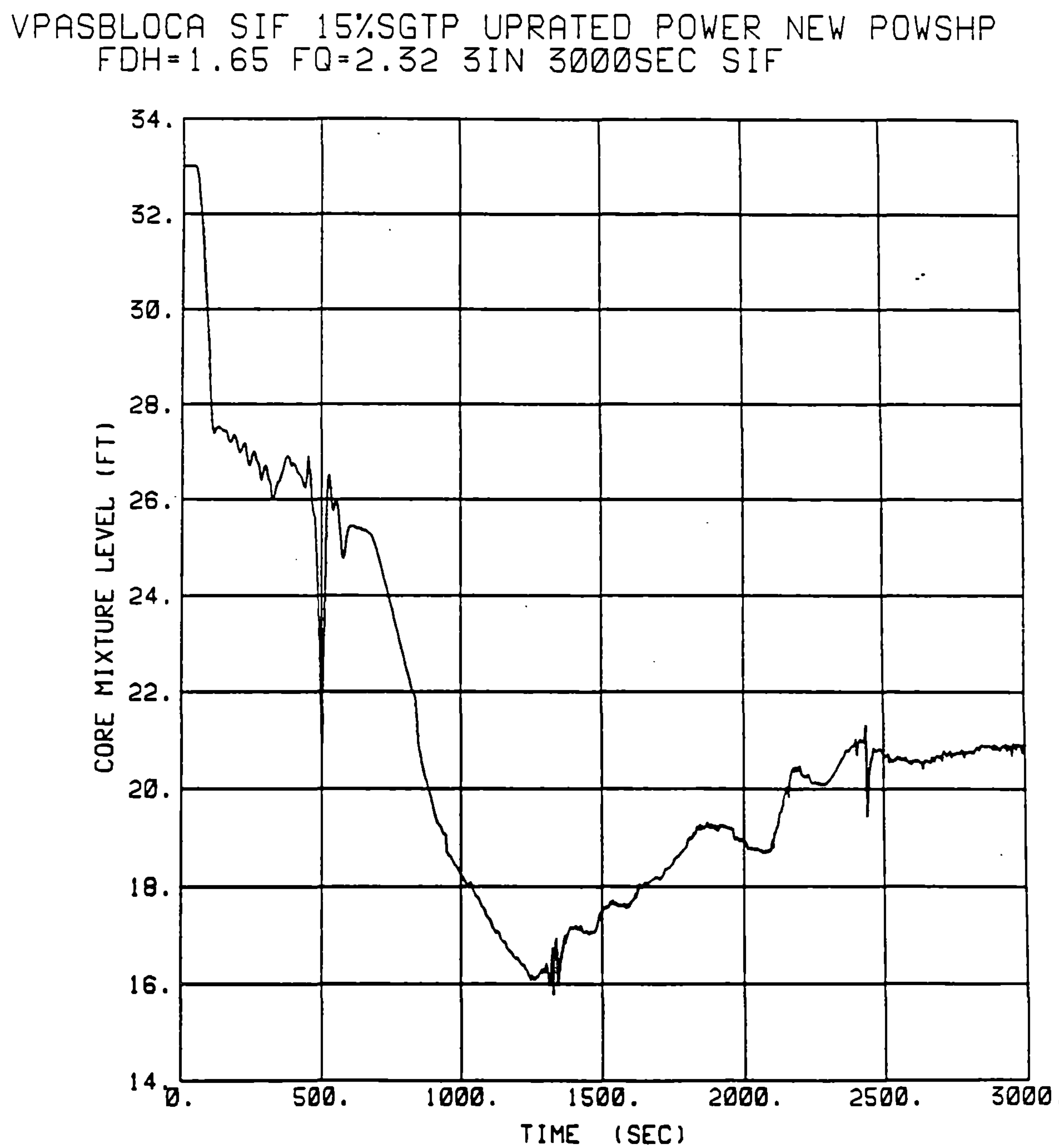
SURRY (VPA/VIR) SMALL BREAK LOCA ANALYSIS
FDH=1.65 FQ-2.32 RSTRT 6IN 2500



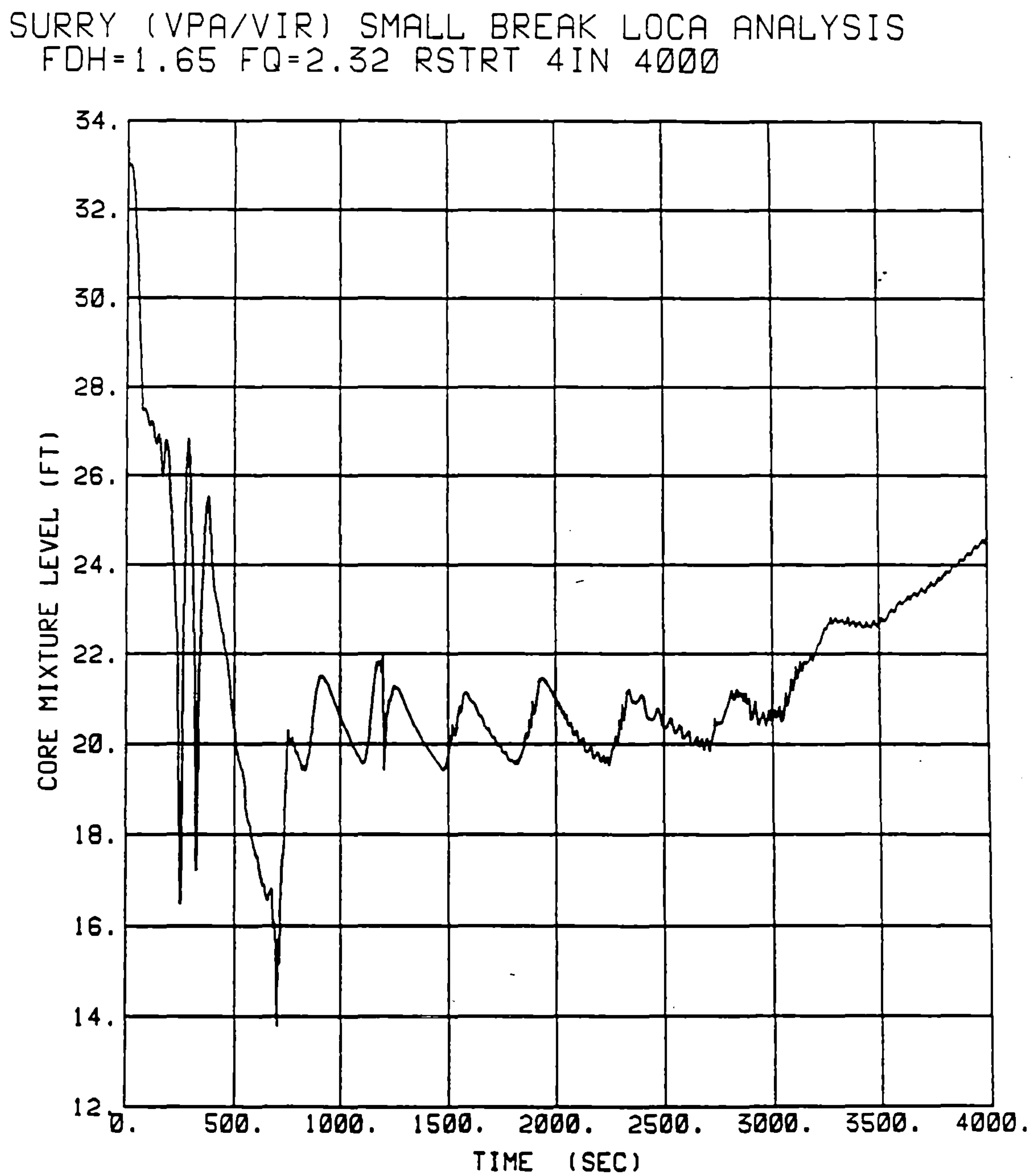
Pressurizer Pressure (psia), 6 inch Break
FIGURE 6.0-4D



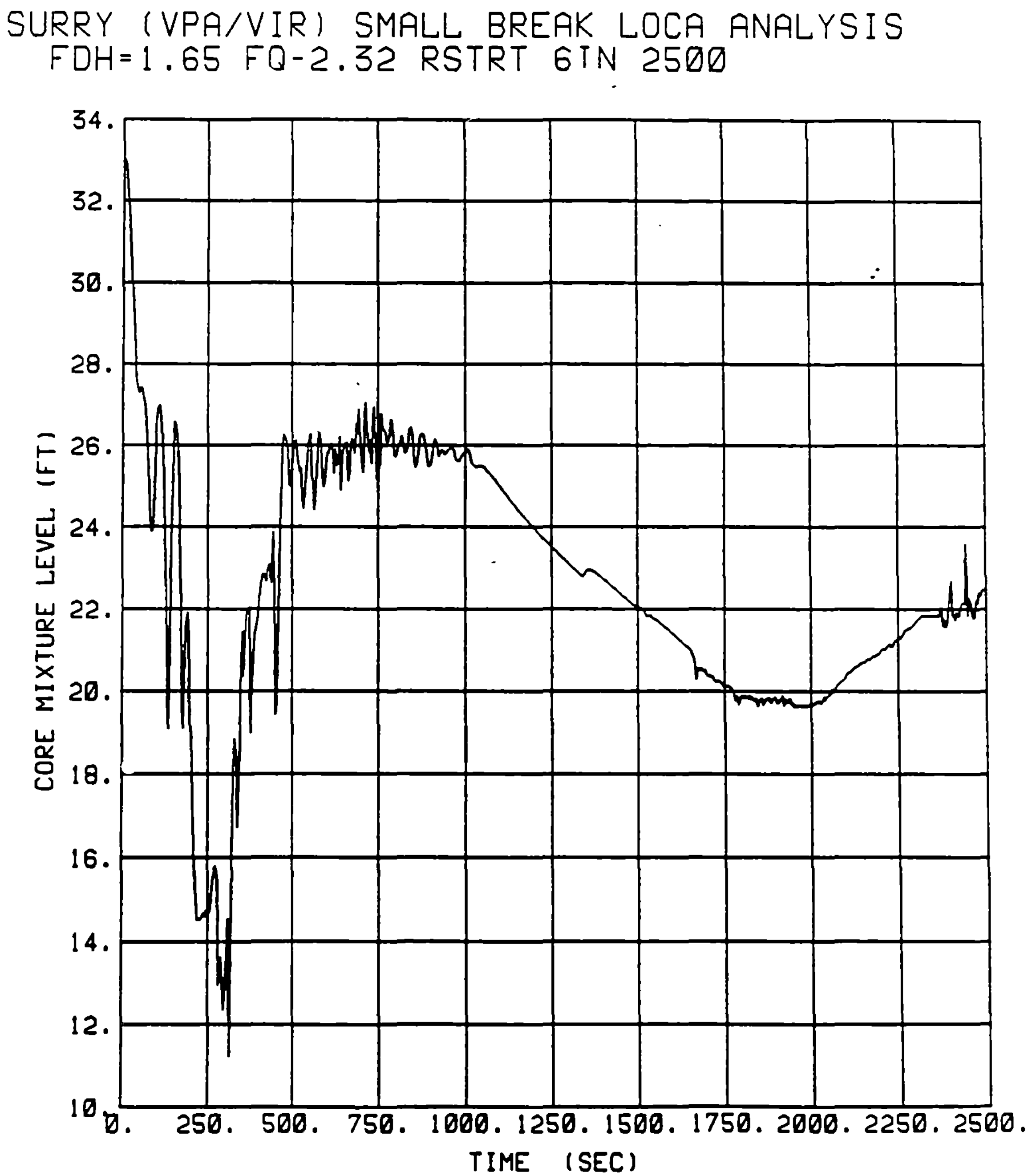
Core Mixture Level (ft), 2 inch Break
FIGURE 6.0-5A



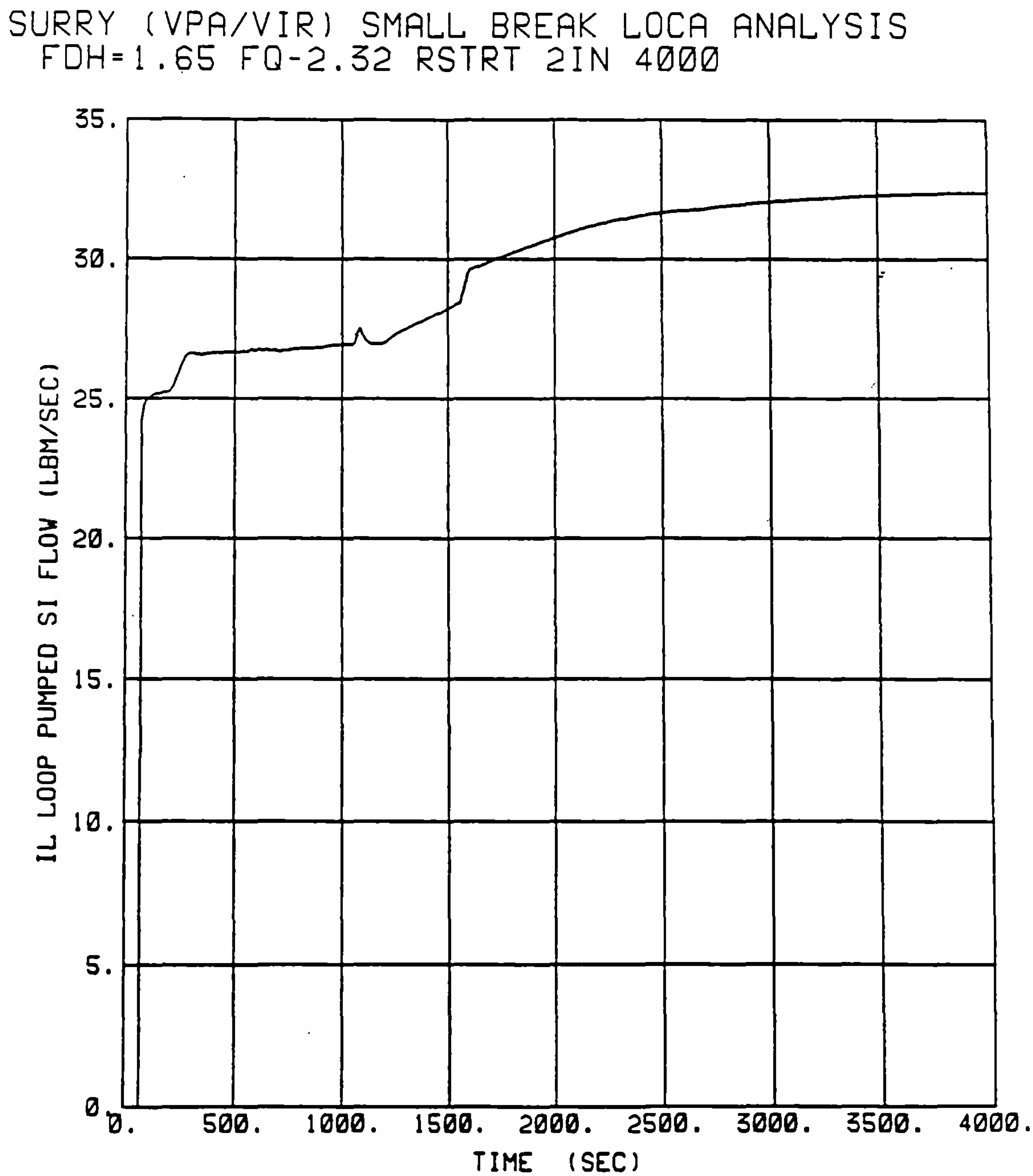
Core Mixture Level (ft), 3 inch Break
FIGURE 6.0-5B



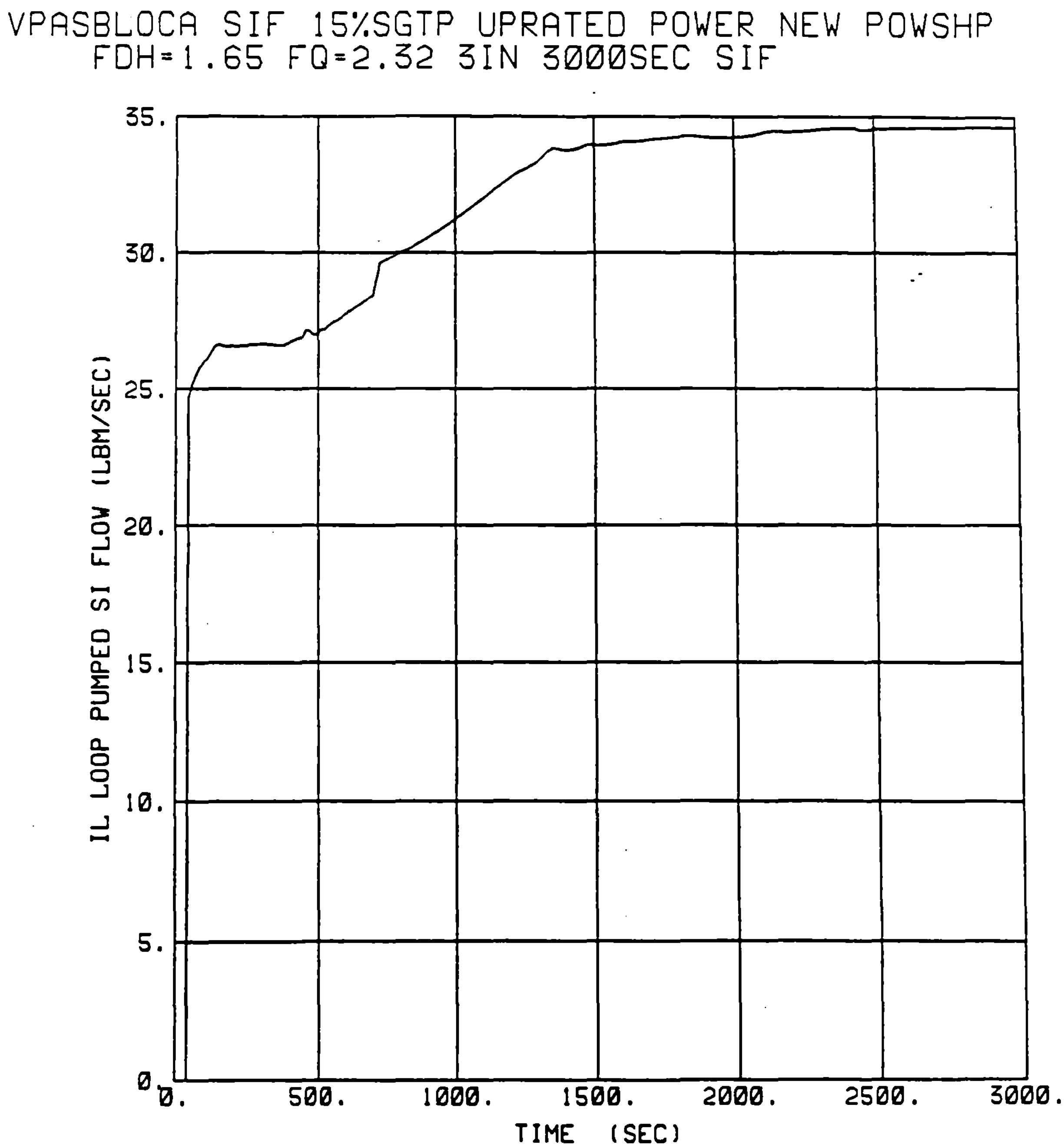
Core Mixture Level (ft), 4 inch Break
FIGURE 6.0-5C



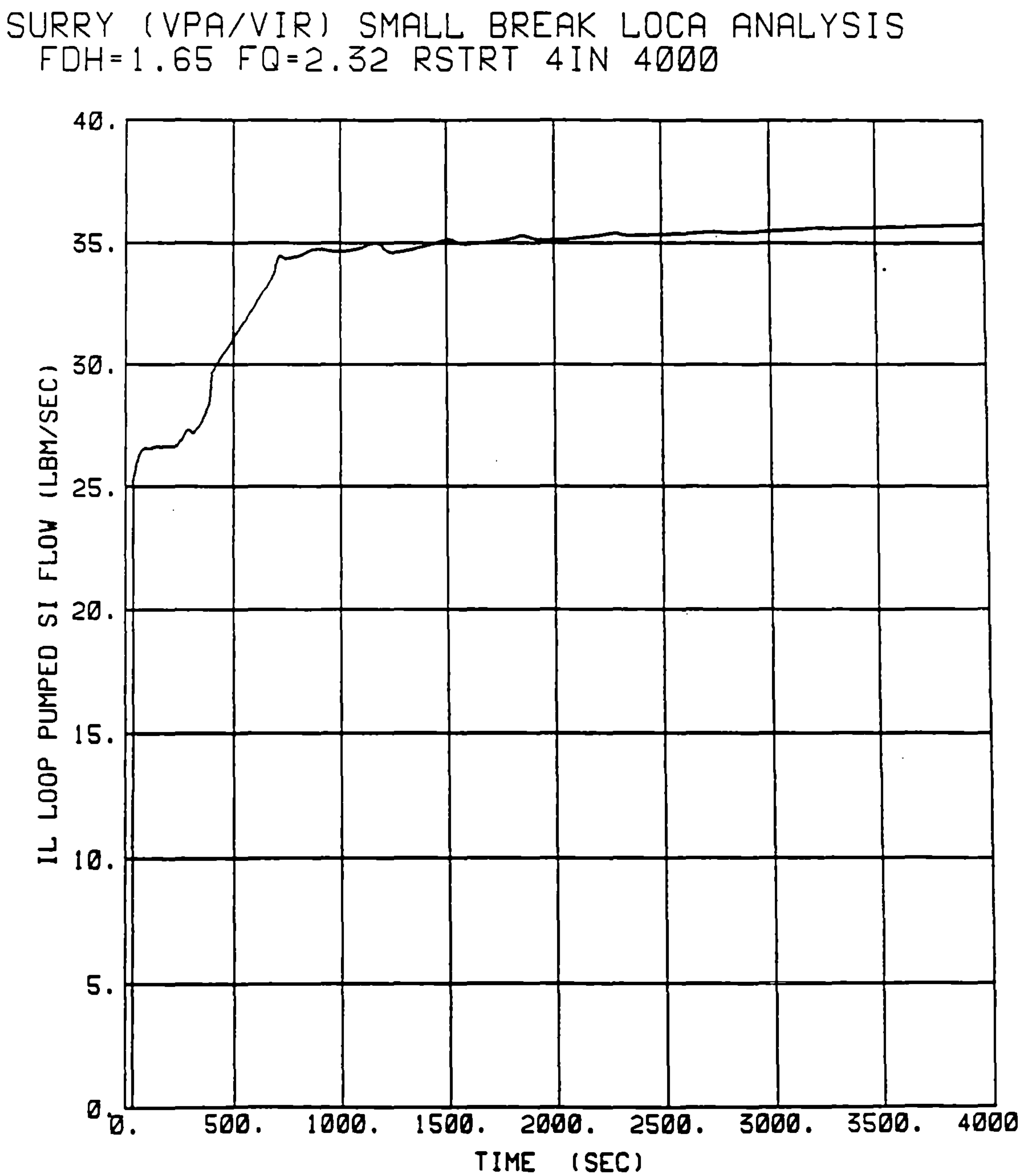
Core Mixture Level (ft), 6 inch Break
FIGURE 6.0-5D



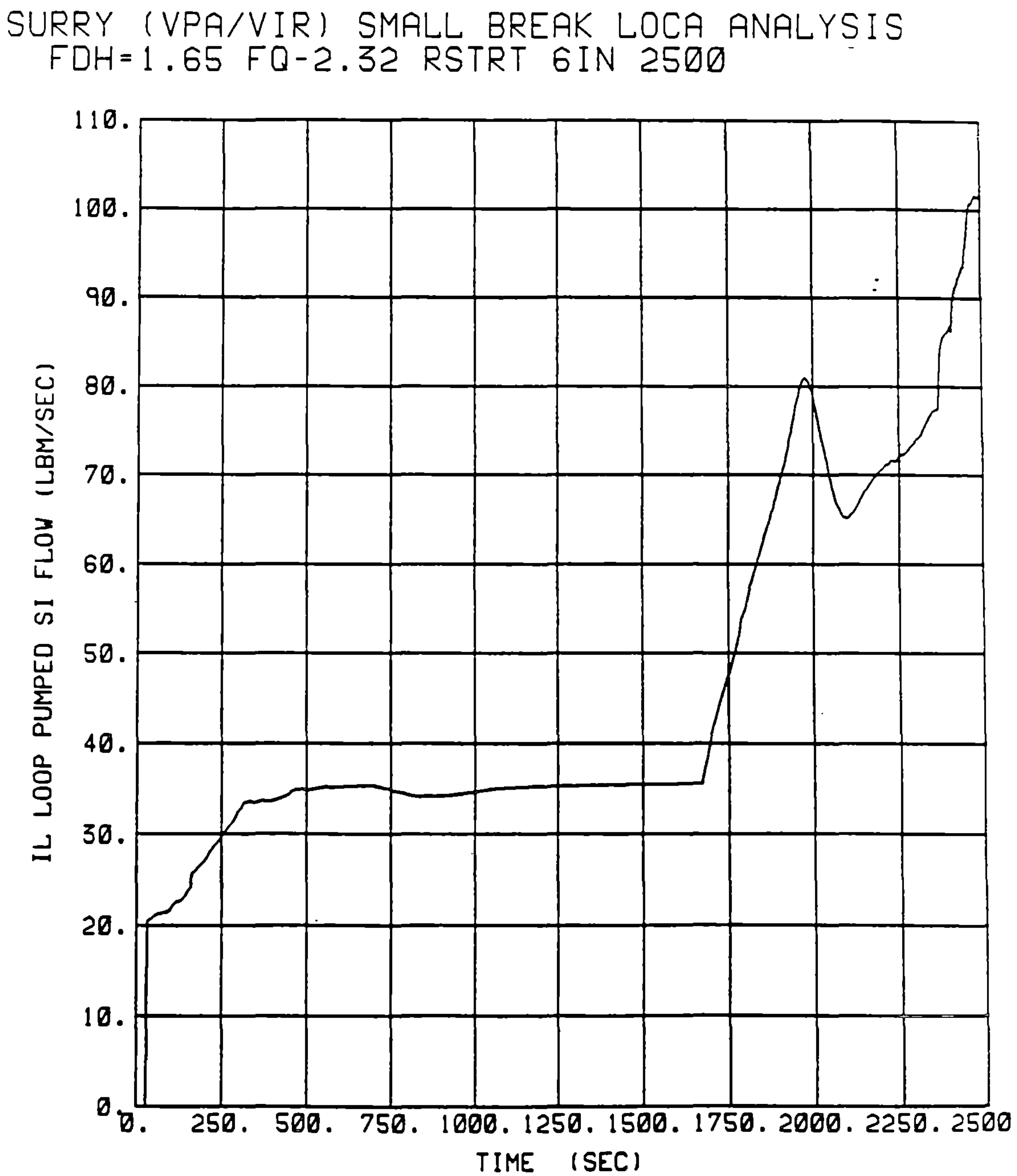
Intact Loop Pumped SI Flow (lbm/sec), 2 inch Break
FIGURE 6.0-6A



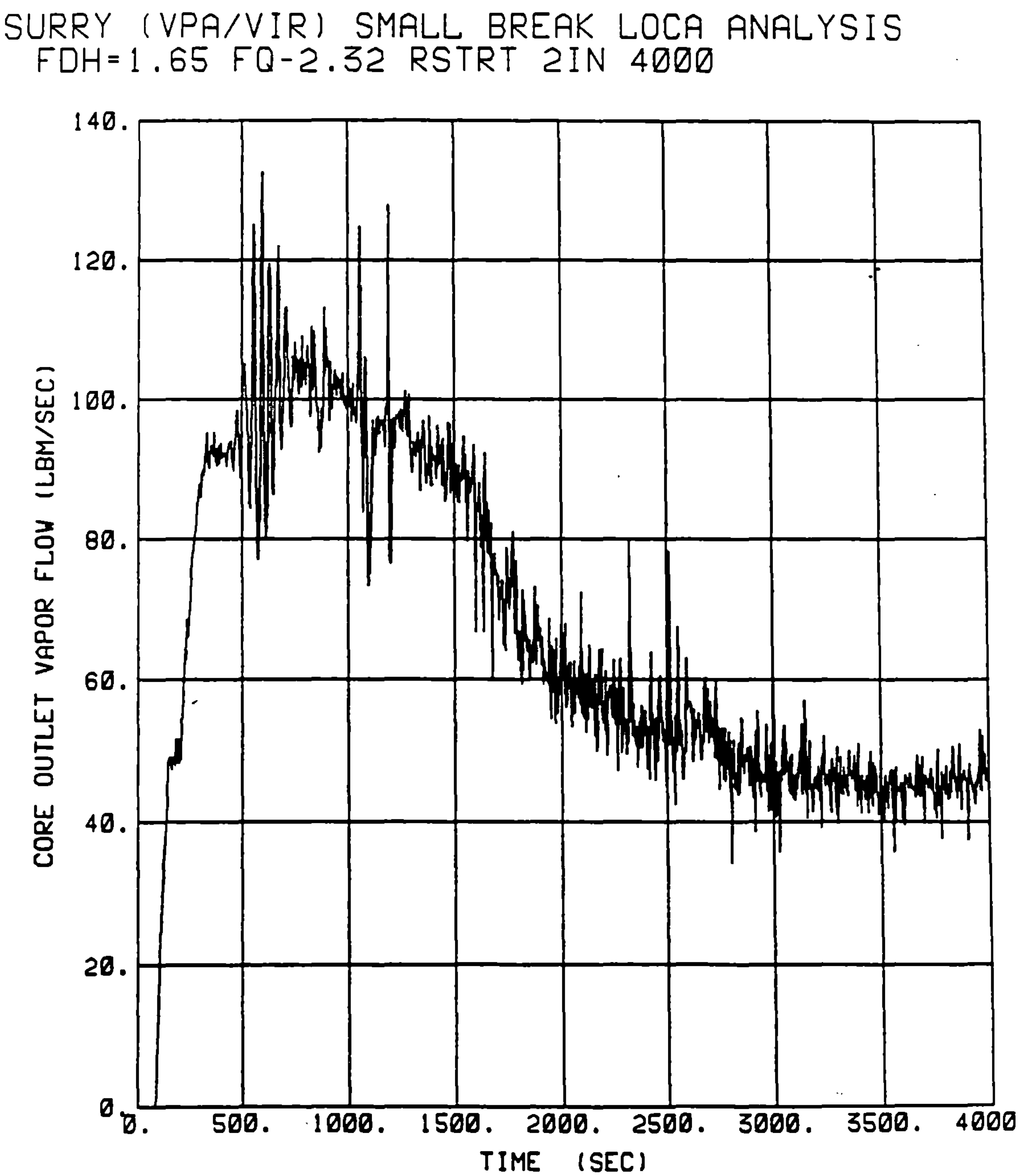
Intact Loop Pumped SI Flow (lbm/sec), 3 inch Break
FIGURE 6.0-6B



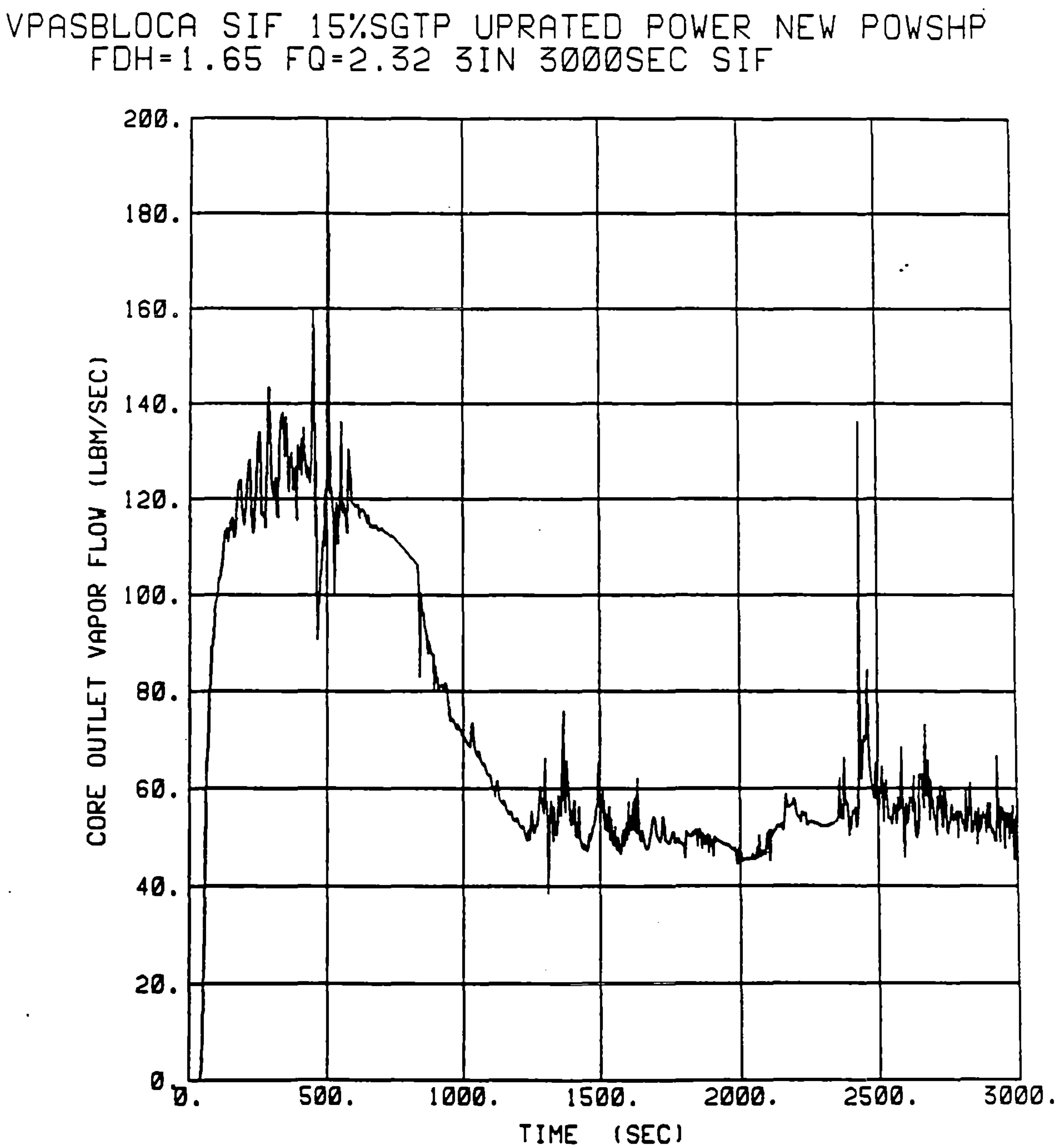
Intact Loop Pumped SI Flow (lbm/sec), 4 inch Break
FIGURE 6.0-6C



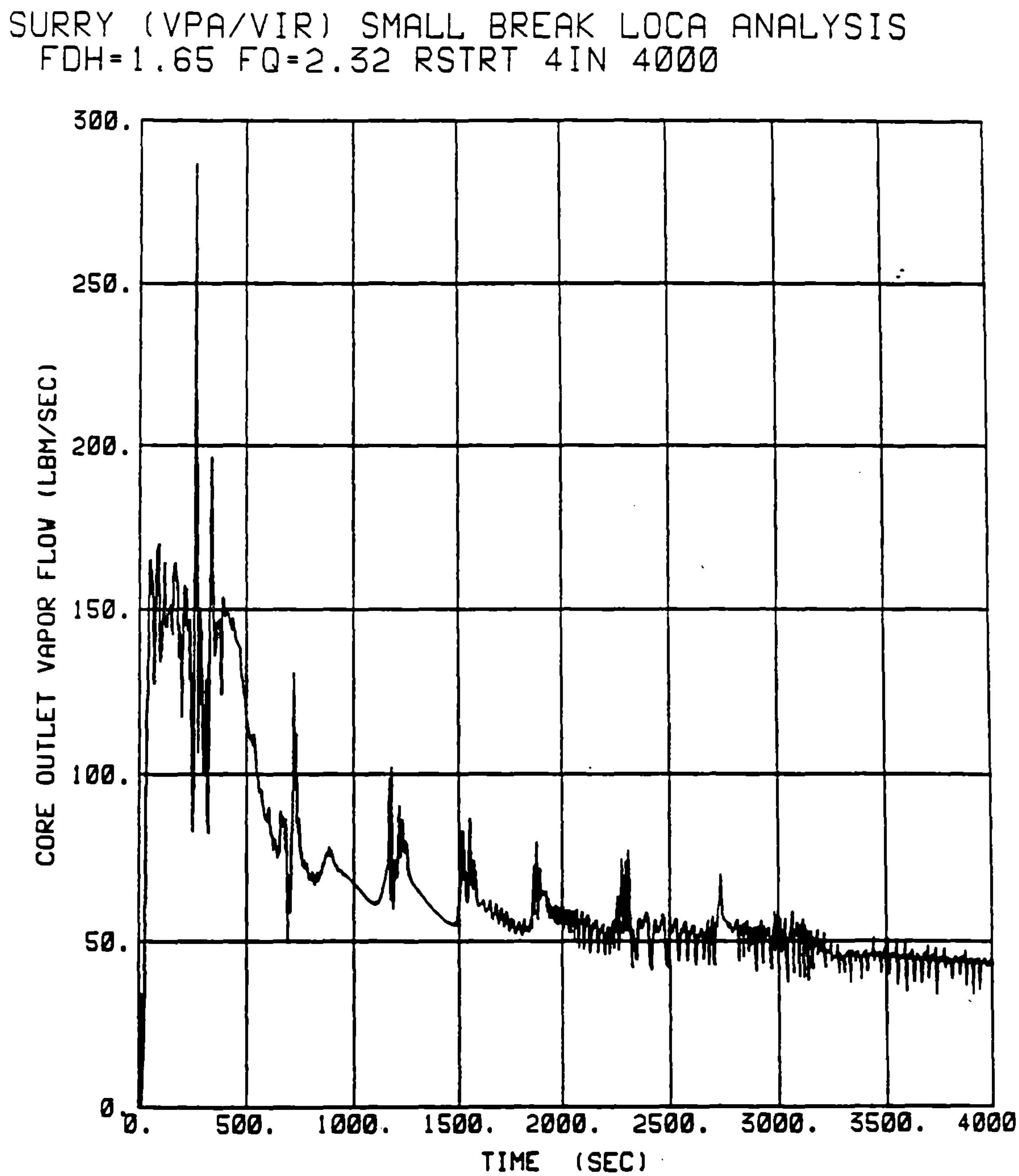
Intact Loop Pumped SI Flow (lbm/sec), 6 inch Break
FIGURE 6.0-6D



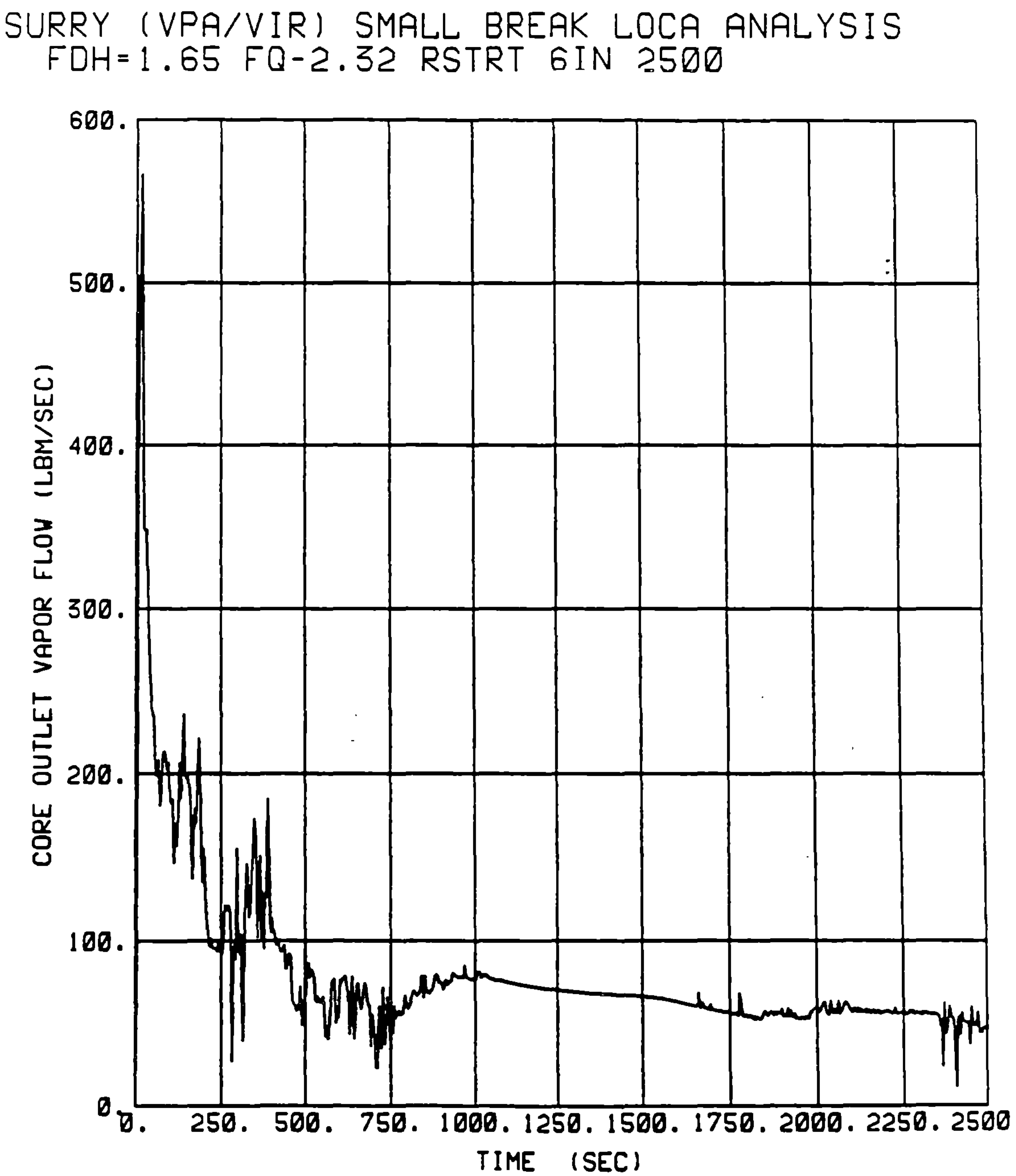
Core Exit Vapor Flow (lbm/sec), 2 inch Break
FIGURE 6.0-7A



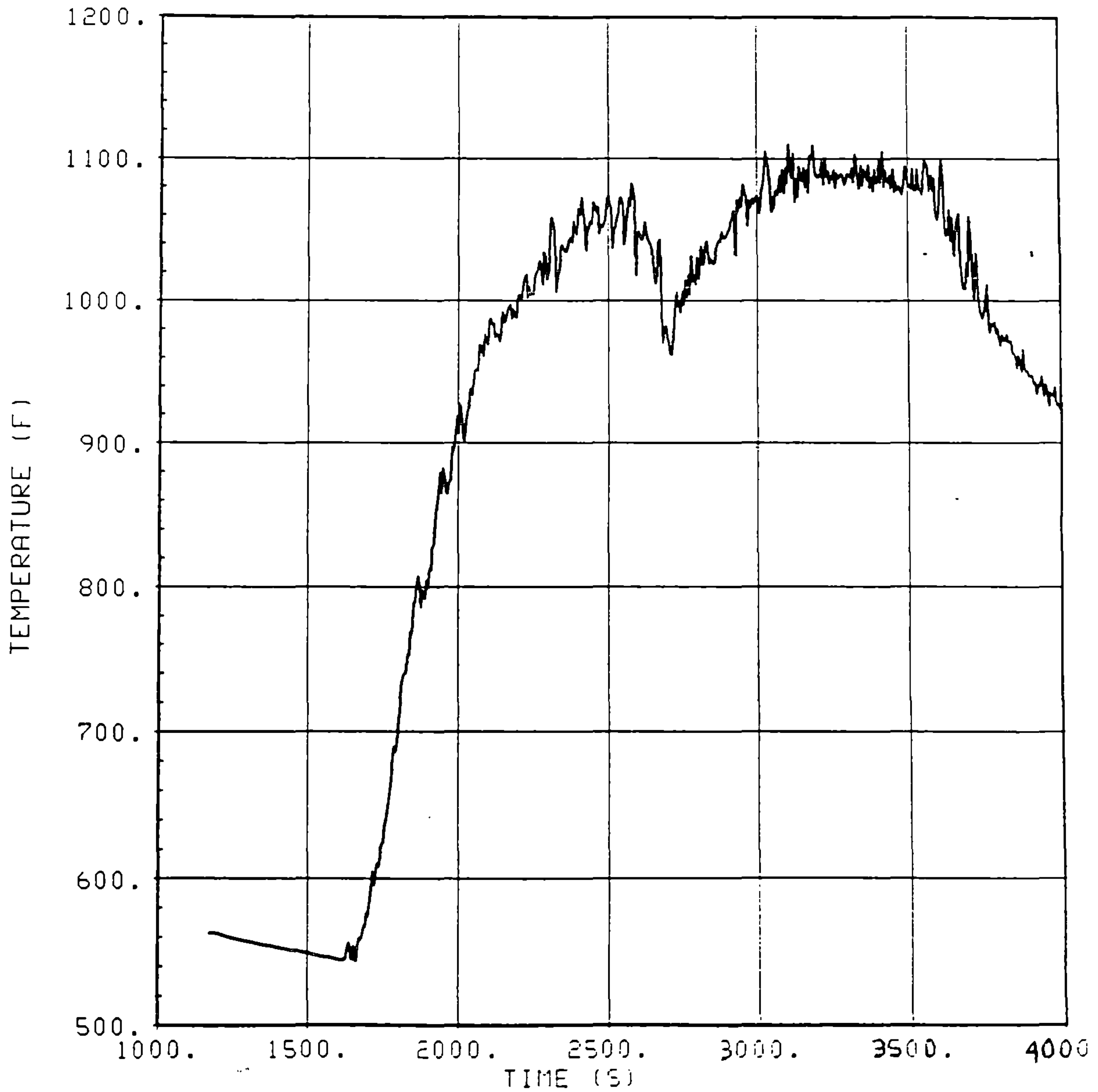
Core Exit Vapor Flow (lbm/sec), 3 inch Break
FIGURE 6.0-7B



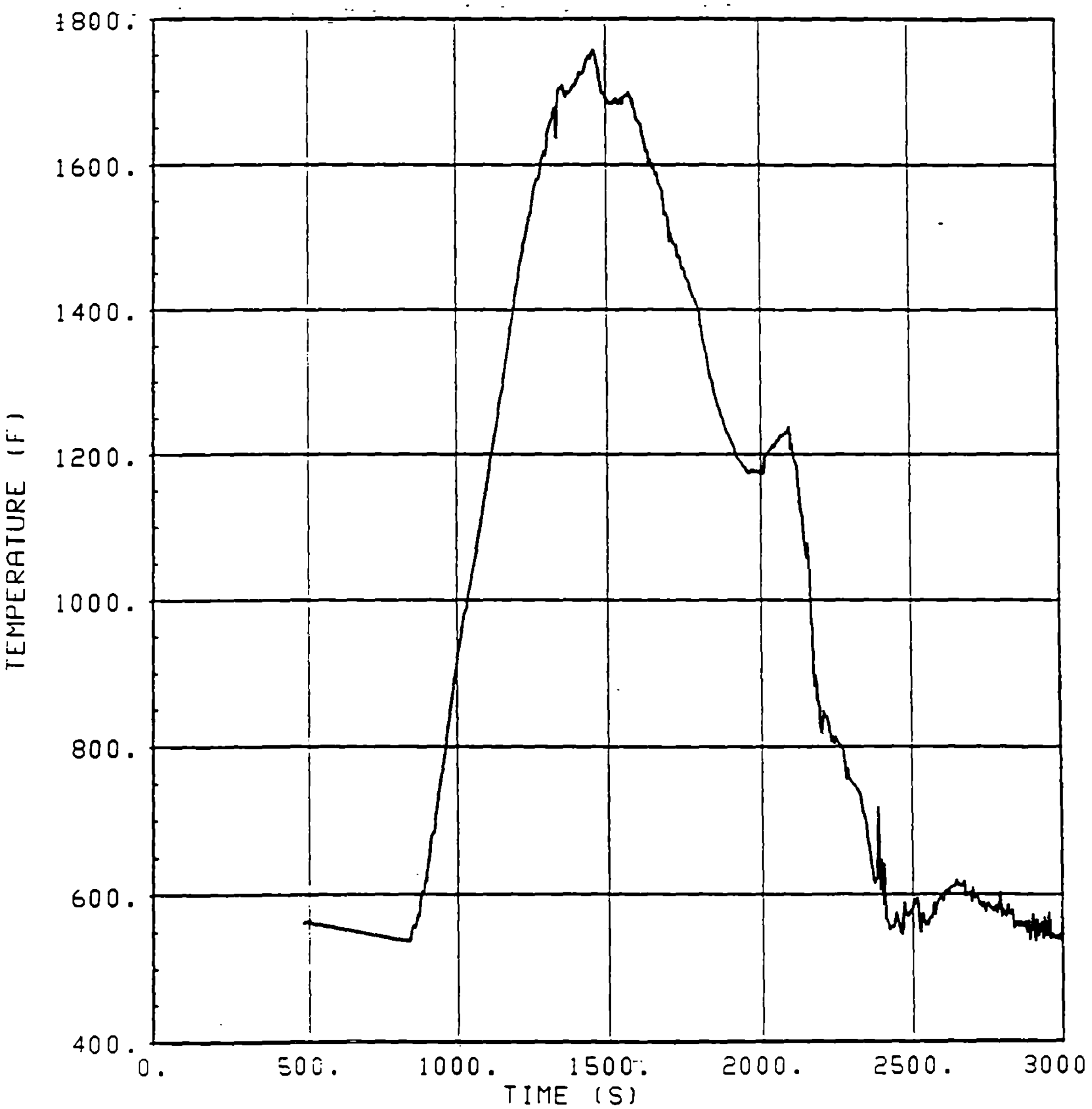
Core Exit Vapor Flow (lbm/sec), 4 inch Break
FIGURE 6.0-7C



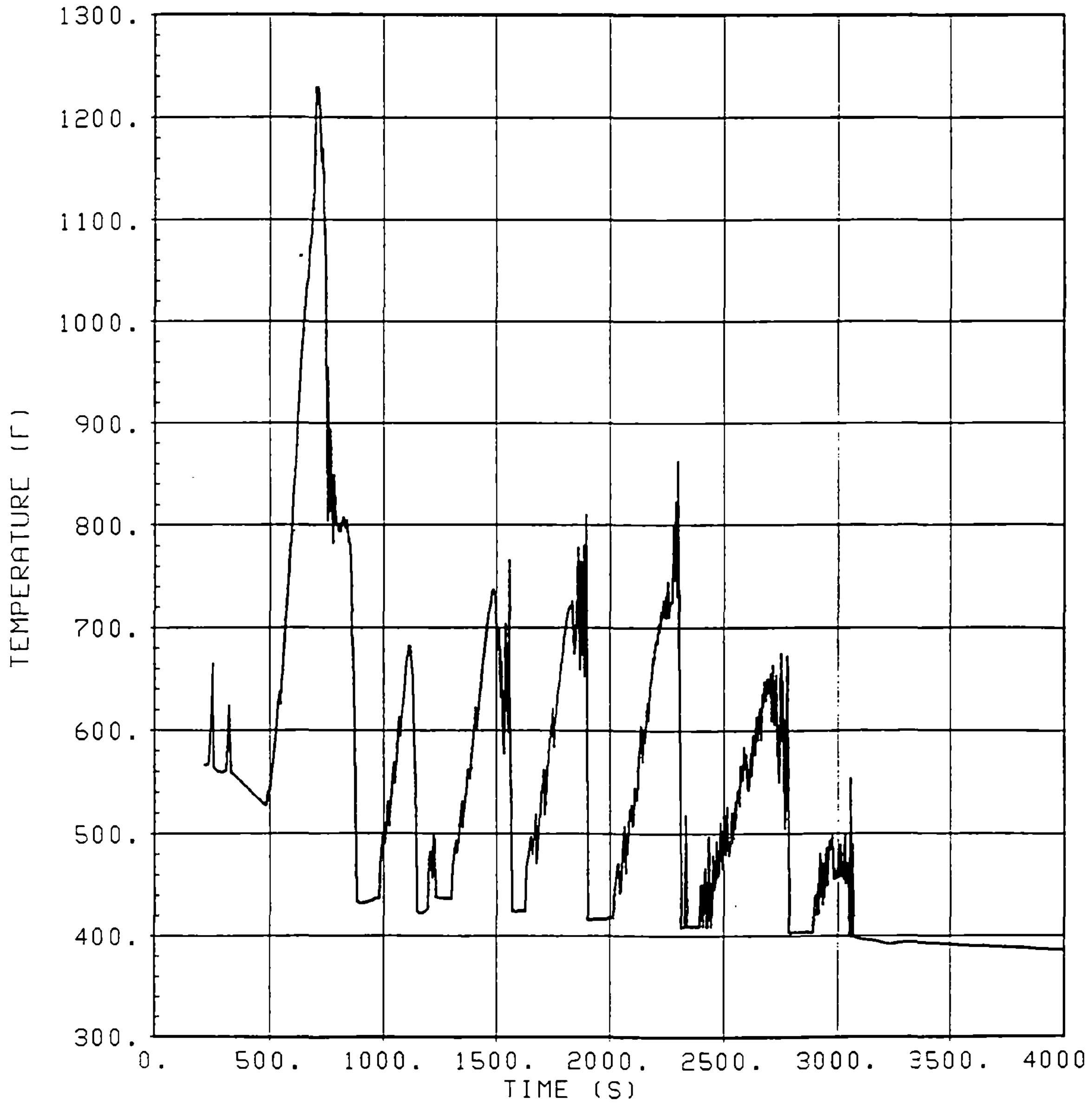
Core Exit Vapor Flow (lbm/sec), 6 inch Break
FIGURE 6.0-7D



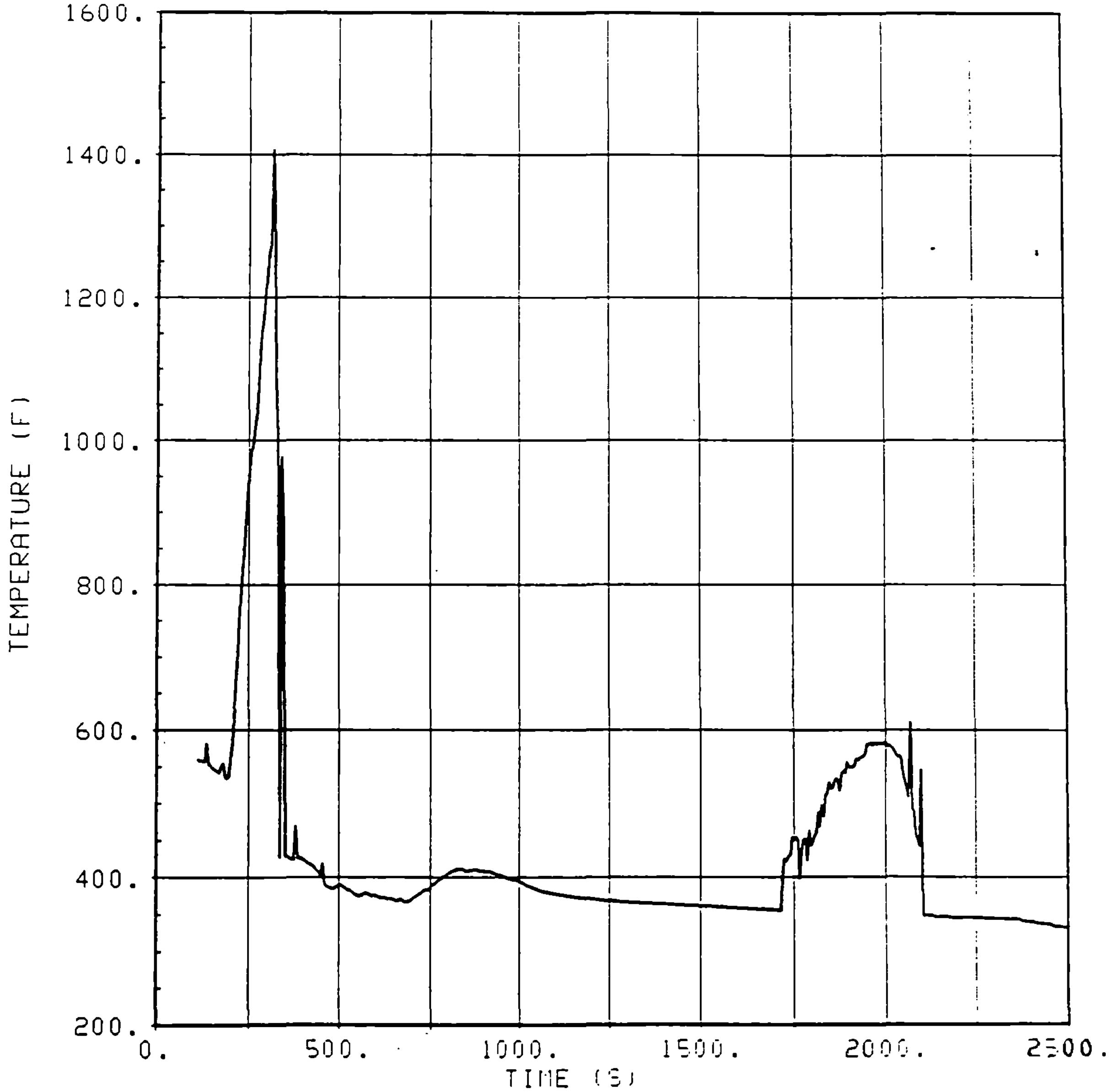
Fluid Temperature in Hot Assembly (°F), 2 inch Break
FIGURE 6.0-8A



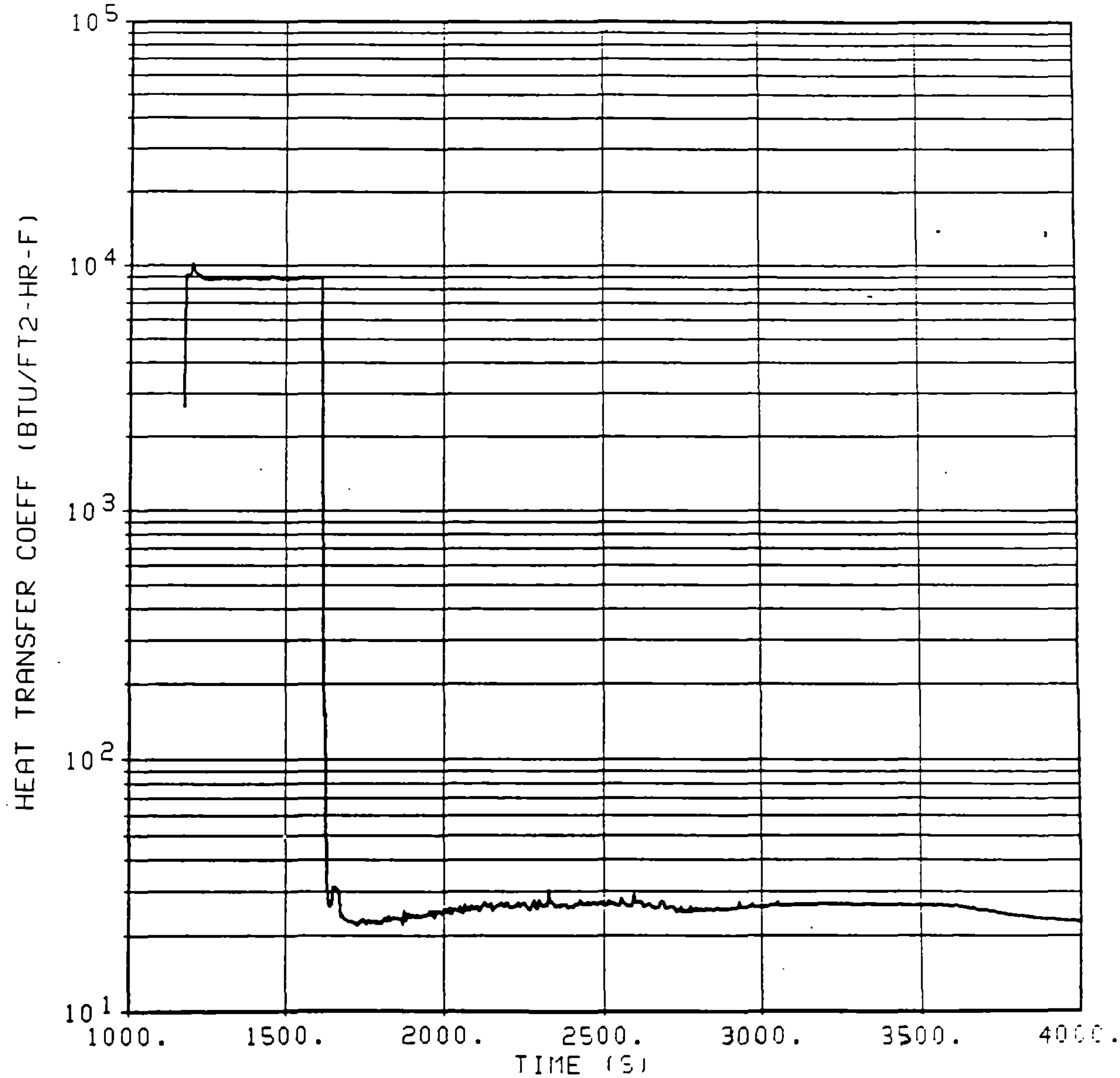
Fluid Temperature in Hot Assembly (°F), 3 inch Break
FIGURE 6.0-8B



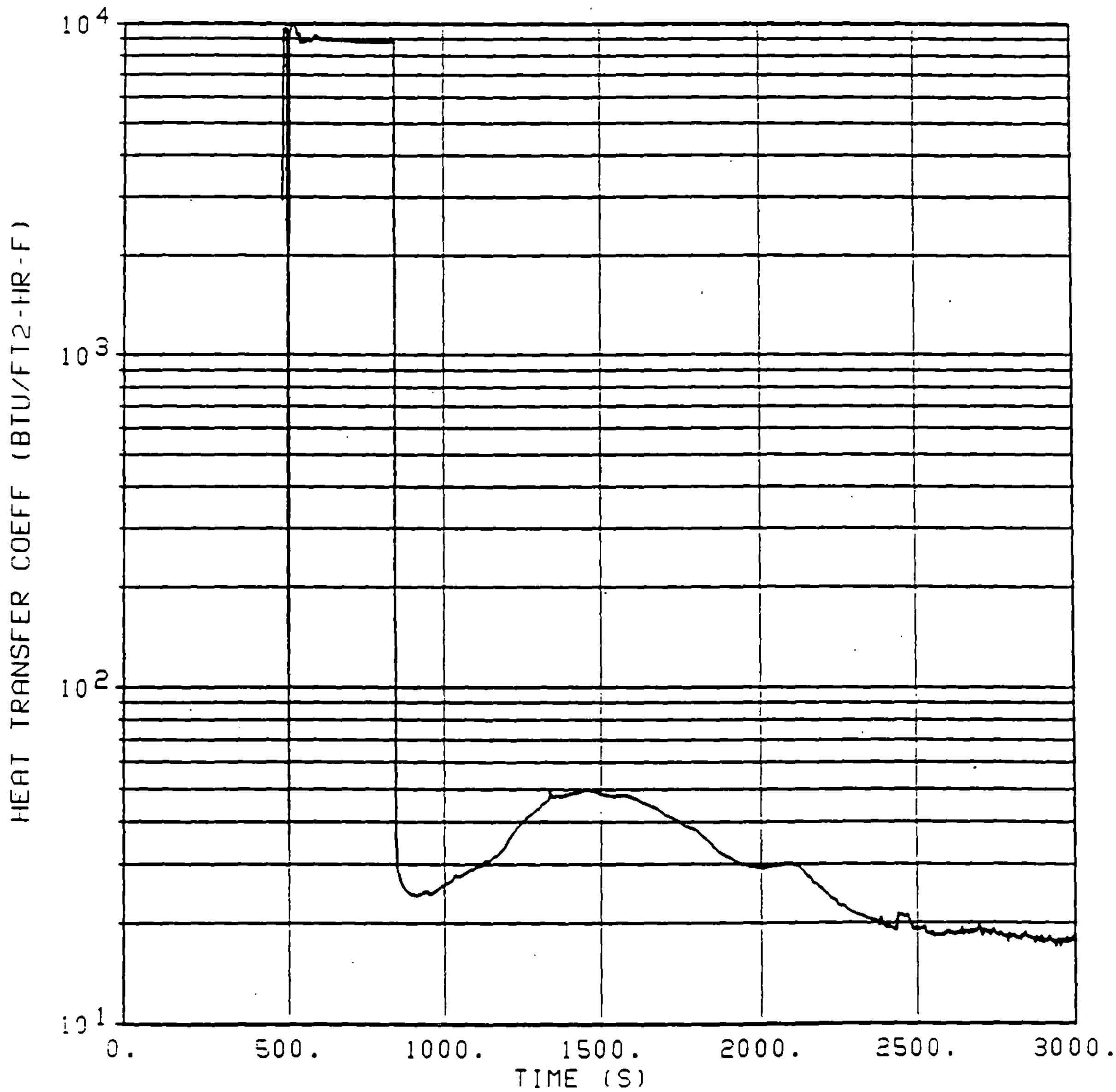
Fluid Temperature in Hot Assembly (°F), 4 inch Break
FIGURE 6.0-8C



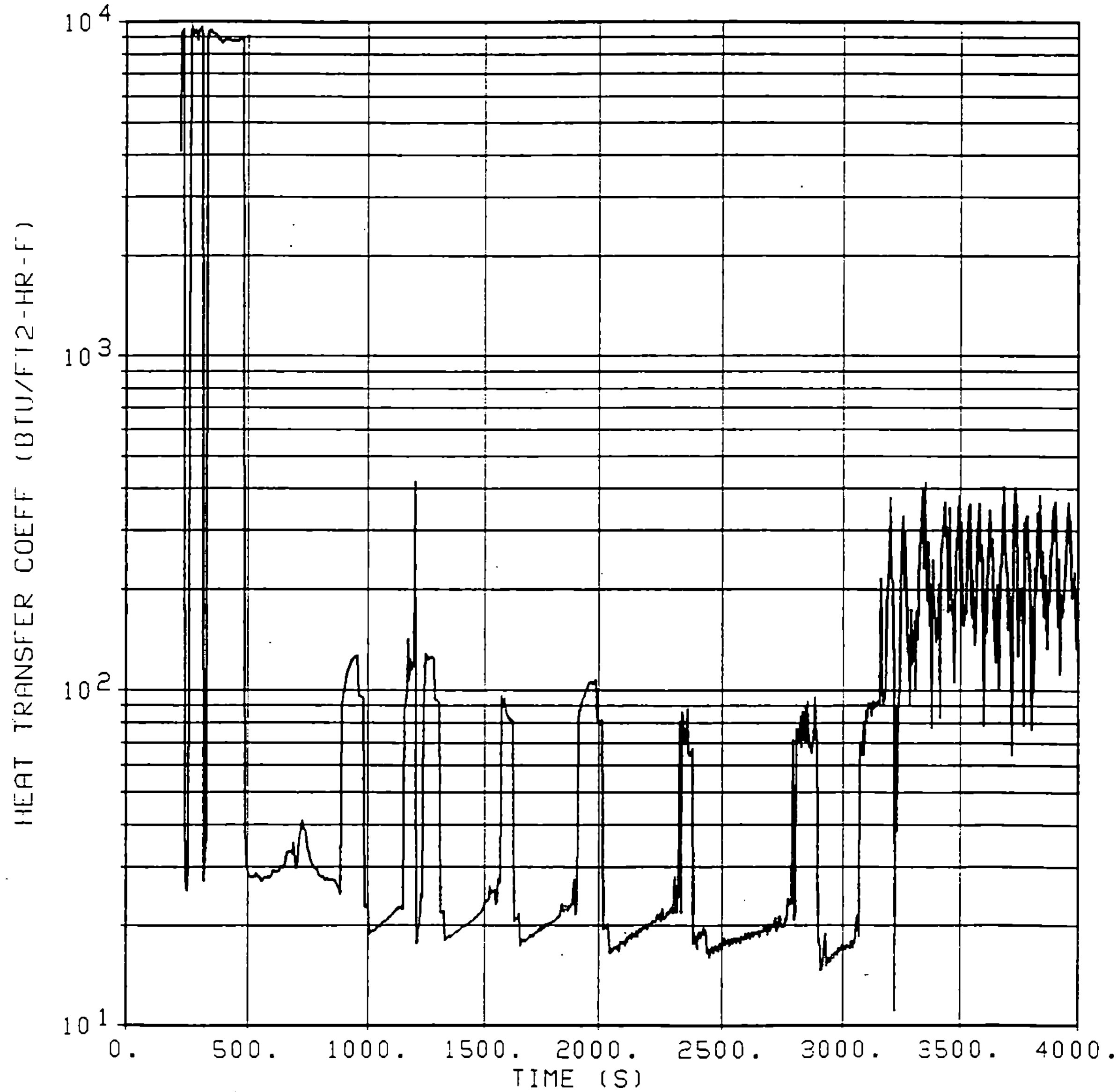
Fluid Temperature in Hot Assembly (°F), 6 inch Break
FIGURE 6.0-8D



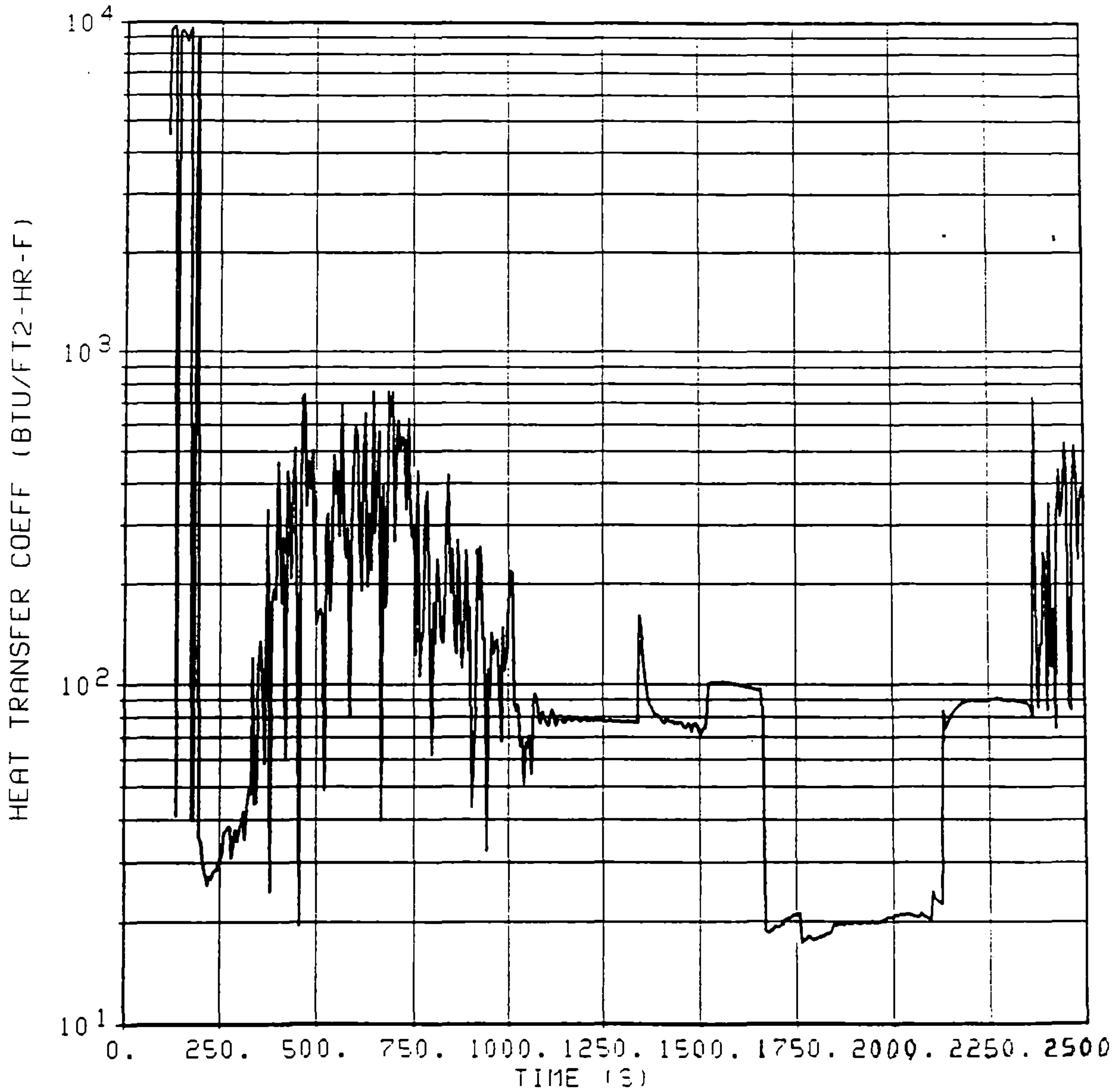
Heat Transfer Coefficient (BTU/hr-ft^2), 2 inch Break
FIGURE 6.0-9A



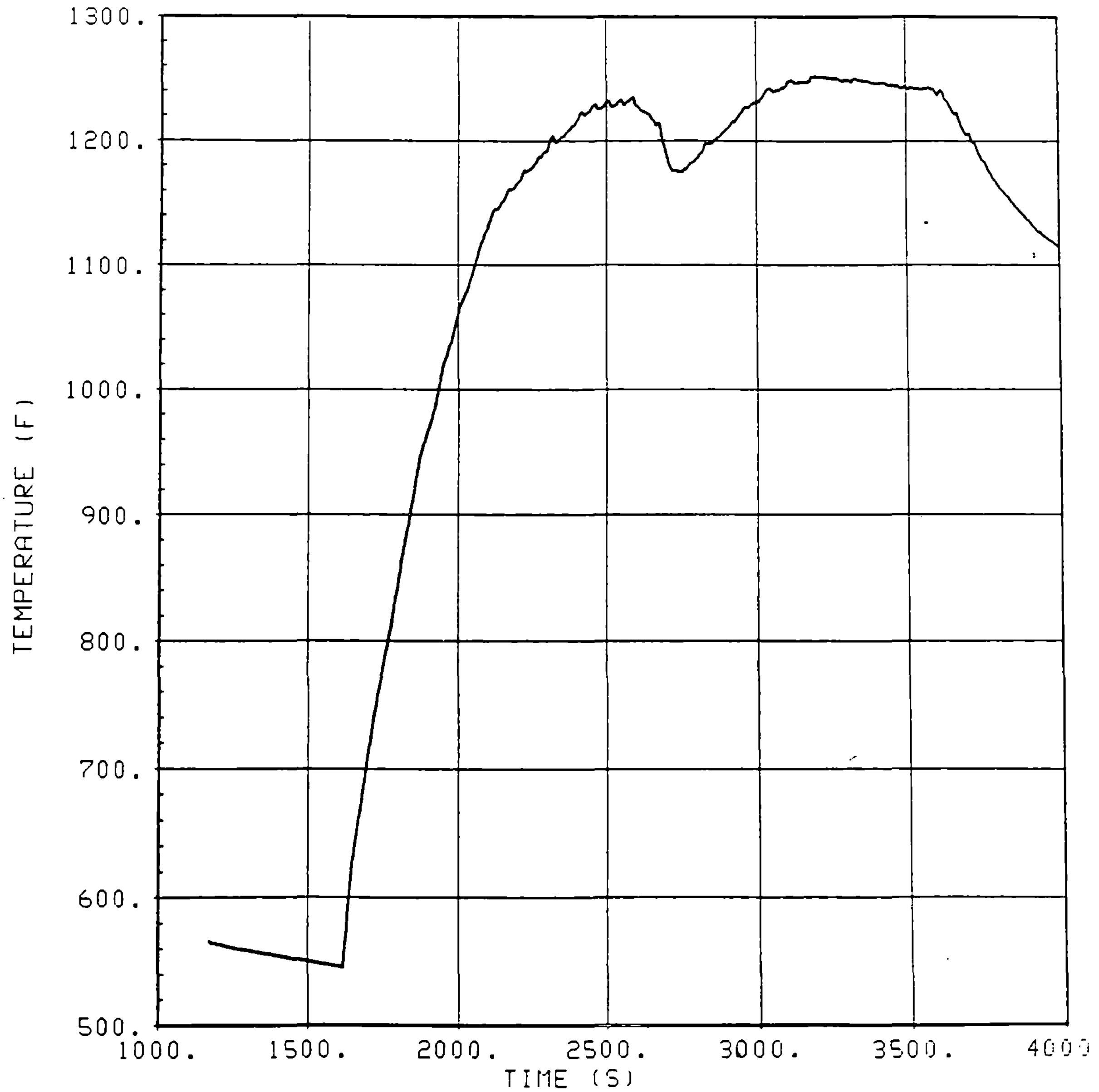
Heat Transfer Coefficient ($BTU/hr-ft^2$), 3 inch Break
FIGURE 6.0-9B



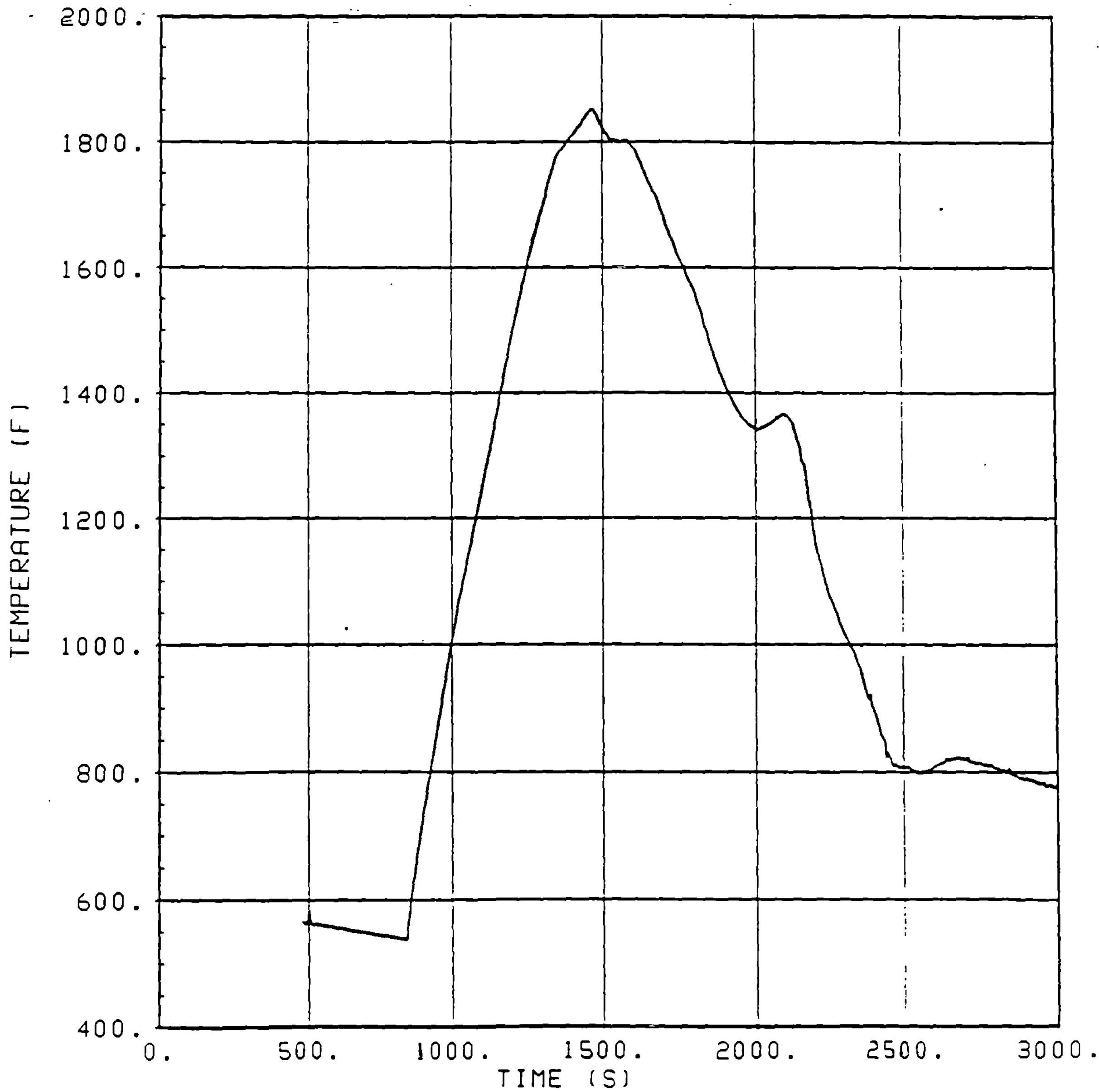
Heat Transfer Coefficient (BTU/hr-ft²), 4 inch Break
FIGURE 6.0-9C



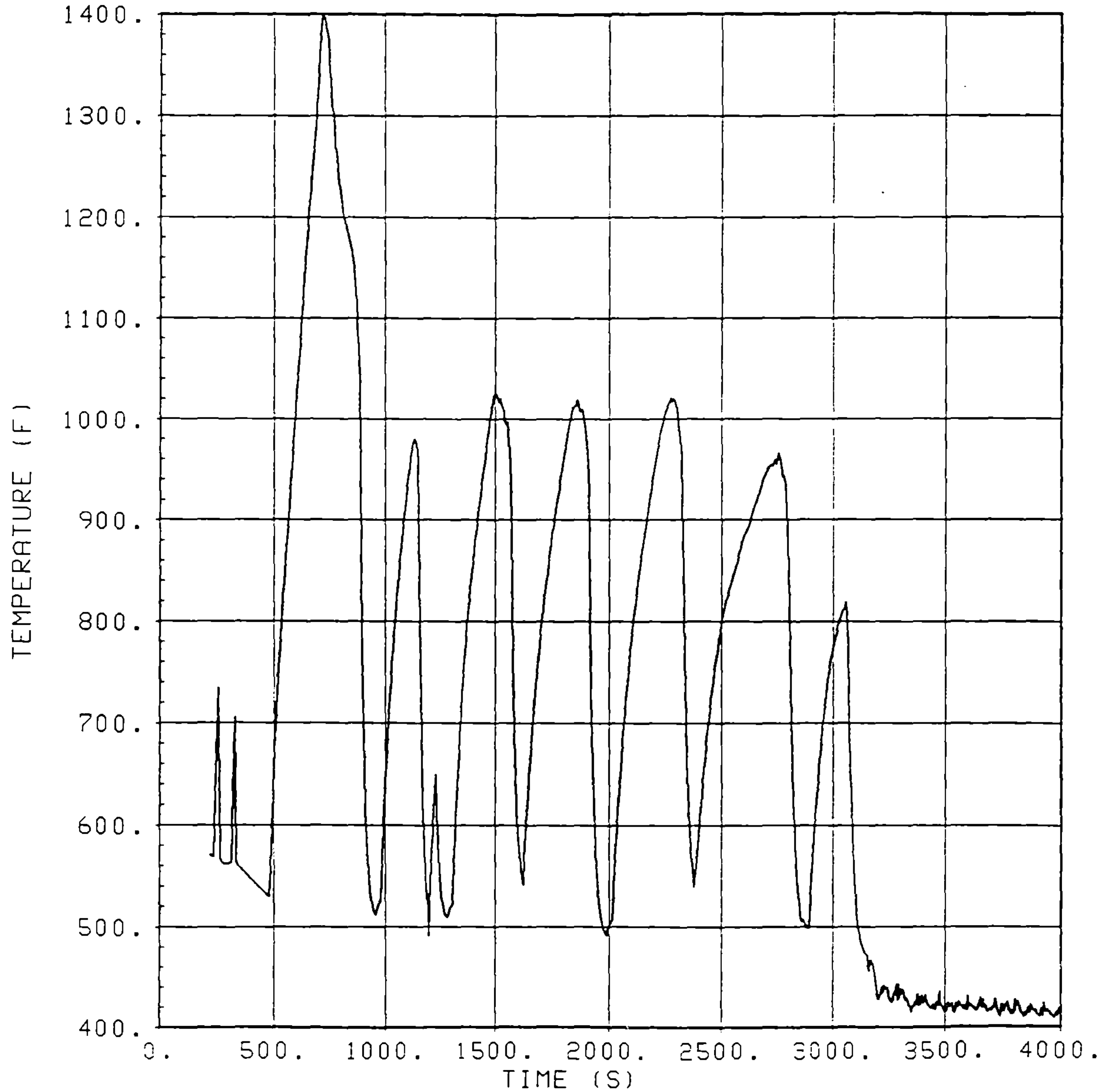
Heat Transfer Coefficient (BTU/hr-ft²), 6 inch Break
FIGURE 6.0-9D



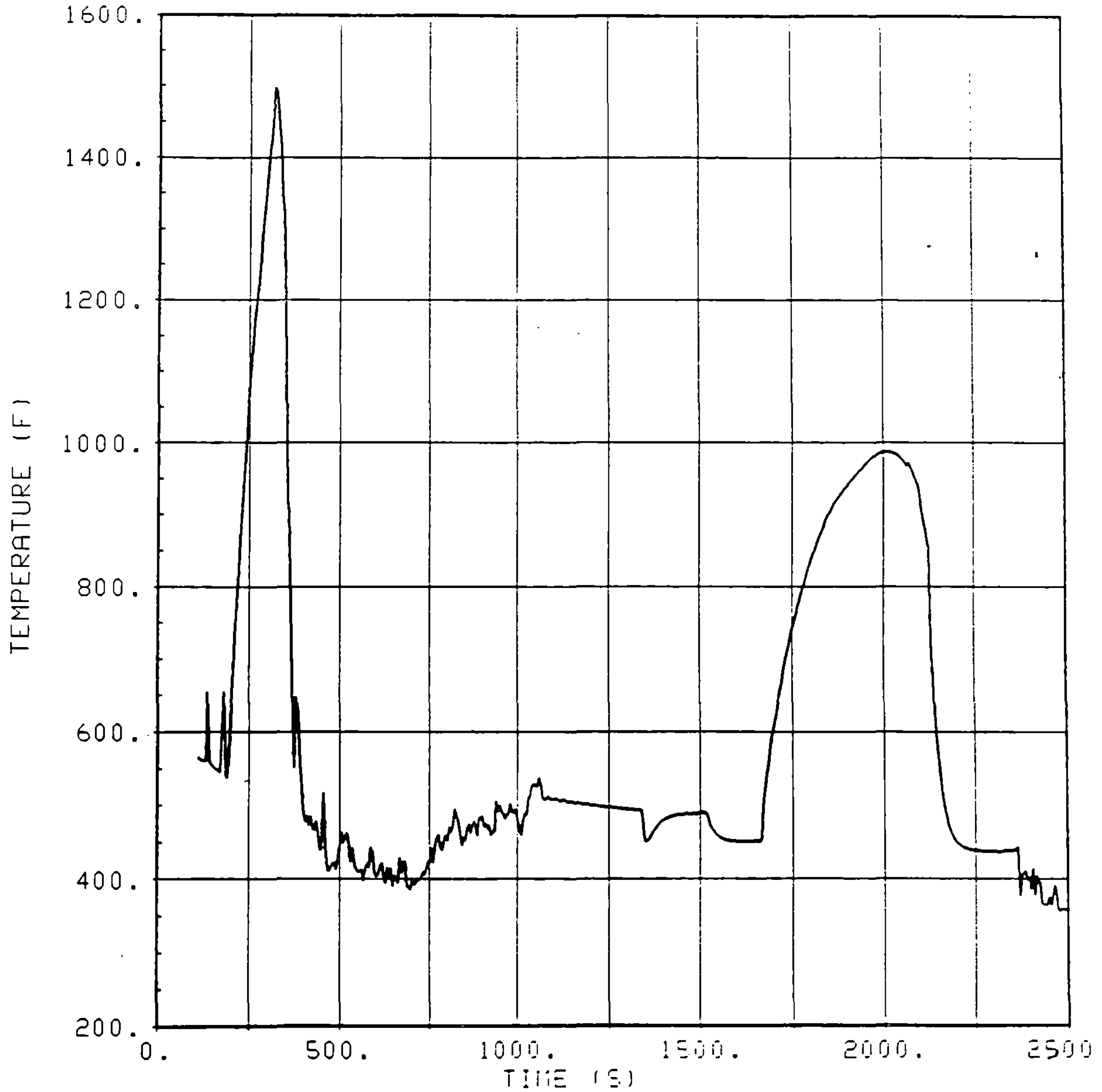
Hot Rod Clad Average Temperature (°F), 2 inch Break
FIGURE 6.0-10A



Hot Rod Clad Average Temperature (°F), 3 inch Break
FIGURE 6.0-10B



Hot Rod Clad Average Temperature (°F), 4 inch Break
FIGURE 6.0-10C



Hot Rod Clad Average Temperature (°F), 6 inch Break
FIGURE 6.0-10D

7.0 CONCLUSIONS

Virginia Power's Statistical DNBR Evaluation Methodology has been used to derive a Statistical DNBR Limit of 1.27 for SIF/WRB-1. This limit provides peak rod DNB protection with at least 95% probability at a 95% confidence level and 99.9% DNB protection for the full core. Retained DNBR margin was added to the limit to yield a design Statistical DNBR Limit of 1.46 for SIF.

The Loss of Flow accident was reanalyzed with the new Methodology. This reanalysis and the reanalysis of the SBLOCA were performed with a safety analysis $F\Delta h$ which corresponds to a nuclear design limit of 1.50, an increase from the previous limit of 1.435. The results of these reanalyses, and a review of the other FSAR accidents, demonstrate that safety margins are preserved even with the $F\Delta h$ increase.

The current Technical Specifications OTΔT/OPΔT setpoints and F(ΔI) function have been shown to provide bounding CTL protection for both SIF and LOPAR with a 1.62 $F\Delta h$.

The proposed Technical Specifications present an $F\Delta h$ limit of 1.56 reflecting a statistical treatment of the measurement uncertainty for this parameter. Uncertainties in $F\Delta h$ for LOPAR assemblies will continue to be applied deterministically in reload core design calculations.

As described in Sections 4.3, core thermal limit protection has been shown to be provided by the existing protection setpoints and F(ΔI) function for both fuel types. No revised reactor protection system setpoints are necessary to support implementation of the Statistical DNBR Evaluation Methodology or the proposed FΔh increase.

New core thermal limits have been developed to accommodate the increased maximum FΔh based on a deterministic treatment of uncertainties. Suitable Technical Specifications changes derived from the Stat DNB implementation analysis and increased FΔh have been prepared.

A DNB Parameter Surveillance requirement is included as a new Technical Specification for Surry for the implementation of the Statistical DNBR Evaluation Methodology. This new specification ensures that the RCS Tavg, pressurizer pressure, and RCS flow are maintained within the range of values assumed in safety analysis. The DNB parameter surveillance requirement is patterned after the Westinghouse Standard Technical Specifications and includes a flow calorimetric which must be performed once per cycle.

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(2) Letter from W. L. Stewart (VP) to Document Control Desk (NRC), "Virginia Electric and Power Company North Anna Power Station Units 1 and 2 and 2 Proposed Technical Specifications Change," Serial No. 87-231, dated June 17, 1987.

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(7) Letter from C. P. Patel (NRC) to W. L. Stewart (VP), "Surry Units 1 and 2 - Issuance of Amendments (TAC Nos. 64841 and 64842)," Serial No. 87-774, NAF File 7.1.3, dated December 10, 1987.

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(16) Letter from W. L. Stewart (Vepco) to Document Control Desk (NRC), "Virginia Electric and Power Company; Surry Power Station Units 1 and 2; Proposed Technical Specifications Change; Surry Improved Fuel Assembly," Serial No. 87-188, dated May 26, 1987.

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(19) Letter from W. L. Stewart (Virginia Power) to H. R. Denton (NRC), "Surry and North Anna Power Stations Reactor System Transient Analyses," Serial No. 85-753, dated November 19, 1985.

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ATTACHMENT 3

VIRGINIA ELECTRIC AND POWER COMPANY
PROPOSED TECHNICAL SPECIFICATION REVISION

$F_{\Delta h}$ INCREASE/STATISTICAL DNBR METHODOLOGY - SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

BASIS FOR NO SIGNIFICANT SAFETY HAZARDS DETERMINATION

An increased enthalpy rise hot channel factor (FΔh) and implementation of the approved Statistical DNBR Evaluation Methodology are being pursued for Surry Units 1 and 2. An increased design limit FΔh is needed at this time primarily to accommodate the increased radial power factors resulting from the planned installation of flux suppression inserts (FSIs) in Surry 1. The FSIs are being installed to reduce the peripheral core power and, hence, the fast neutron flux near vessel locations which are experiencing neutron irradiation embrittlement.

The implementation of Stat DNB provides increased thermal-hydraulic margin at Surry by treating key DNBR analysis uncertainties in a less restrictive, but appropriately conservative, statistical manner. The analyses supporting the implementation of Stat DNB have been performed in accordance with the method described in the approved Statistical DNBR Methodology Topical Report (VEP-NE-2-A). This submittal extends this approved method, already implemented for North Anna, to the Surry units.

The proposed changes do not involve a significant hazards consideration because operation of Surry Units 1 and 2 in accordance with this change would not:

- involve a significant increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report. Neither the FΔh limit nor the Statistical DNBR Methodology make any contribution to the potential accident initiators and, thus, cannot increase the probability of any accident. The key safety analysis parameters discussed in this report bound the current operating characteristics of both Surry fuel types. The reanalyses used approved safety analysis procedures, including conservative modelling of system accident response, to ensure that adequate margin to the design limits was preserved. Therefore, neither the accident probability nor the consequences of any accident can increase as a result of the implementation of the Statistical Methodology or FΔh increase. Further, the addition of a full core DNB design limit for SIF/WRB-1 provides increased assurance that the consequences of a postulated accident, which includes a radioactive release, would be minimized because the overall number of fuel rods in DNB would not exceed the 0.1% level.

- create the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report. Since the implementation of the proposed reanalyses and FΔh limit increase requires no hardware changes (e.g., alterations in plant configuration), operation with these changes does not create the probability for any accident which has not already been evaluated in the Final Safety Analysis Report.

- involve a significant reduction in a margin of safety. The margin of safety is the margin between the design limit (e.g., the DNBR limit or a LOCA clad temperature limit) and the point of actual fuel failure. This margin is preserved by insuring that none of the design limits are surpassed for any FSAR accident. The increased FΔh limit serves to increase margin to reactor vessel material embrittlement limits, since it facilitates the installation of flux suppression inserts in the core periphery. Appropriate evaluations or analyses have verified that none of the design limits have been violated for any FSAR transient, so that there has been no reduction in the margin of safety.

Therefore, pursuant to 10 CFR 50.92, based on the above considerations, it has been determined that these changes do not involve a significant safety hazards consideration.