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UPDATED FEEDWATER NOZZLE FRACTURE MECHANICS ANALYSIS FOR EDWIN I. HATCH NUCLEAR POWER STATION UNIT 2

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GE NUCLEAR ENERGY

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ABSTRACT

This report provides a plant specific feedwater nozzle fracture mechanics assessment for the Hatch 2 plant. The results presented herein are an update to those documented in reports NEDO-30256 (Reference 1) and SASR 1290-HT2 (Reference 4) based on plant data collected during the Spring 1991 shutdown/startup sequence. The intent of this report is to show compliance with NRC requirements regarding feedwater nozzle crack growth, as specified in NUREG-0619 (Reference 2), and amended by NRC Generic Letter 81-11 (Reference 3). The results show that the growth of an assumed initial 0.25-inch crack would propagate to 0.96 inch (which is within the allowable value of one inch) during the 40-year design life of the plant using ASME Code, Section XI (Reference 10) methods. This result demonstrates full compliance with the crack growth analysis requirements of NUREG-0619.

1.0 INTRODUCTION

The Reference 1 report provided a plant specific feedwater nozzle fracture mechanics assessment for Edwin I. Hatch Nuclear Power Plant, Unit 2 (hereafter called Hatch 2) based on the original low flow feedwater controller in conjunction with the plant operating history obtained through 1982. That report was generated in response to Nuclear Regulatory Commission (NRC) requirements regarding feedwater nozzle crack growth. These NRC requirements are contained in NUREG-0619 (Reference 2), as amended by NRC Generic Letter 81-11 (Reference 3), which states that a fracture mechanics evaluation must predict an end-of-design-life crack size of one inch or less. The results of the Reference 1 report demonstrated that the growth of an assumed, initial 0.25-inch crack would propagate to greater than one inch 13 years after the initial plant startup date using ASME Code, Section XI methods.

The plant operational history used in the Reference 1 report was obtained from the first four years of plant operation (1979 - 1982). The number of events was obtained from plant operator log books, and representative thermal cycling magnitudes were developed for startup, shutdown and SCRAM events using plant recorder strip charts for the four year period. The magnitude of each type of event was assumed typical of plant operation, and the number of actual events which occurred between 1979 and 1982 was extrapolated to estimate the number of events the plant would see during its 40 year design life. With this information, a crack growth analysis was performed based on linear elastic fracture mechanics (LEFM) techniques.

Some recommendations were made in the Reference 1 report in an effort to improve the Hatch 2 low flow controller characteristics so that acceptable postulated crack growth results could be achieved. These included checking instrumentation and making adjustments or modifications to the low flow controller system. Subsequent to the completion of the Reference 1 report, Georgia Power Company (GPC) performed some significant modifications to the feedwater control system to improve the cycling characteristics. In addition, the tabulation of thermal events experienced by the plant was continued. As a result of these efforts, the crack growth issue required by NUREG-0619 was

again addressed in Reference 4.

The Reference 4 report documented an updated crack growth analysis using the thermal cycle definitions provided in Reference 1, but using new cycle count projections based on the operating history obtained between 1982 and 1990. This effort was initiated primarily because the original Reference 1 projection of the number of events the plant would likely experience in its 40 year design life (based on the operating history experienced during the first four years of plant operation) was shown to be excessively conservative based on the operating history obtained between 1982 and 1990. As a result, new crack growth projections were documented in Reference 4 to address this conservatism. The results demonstrated that the growth of an assumed, initial 0.25-inch crack would propagate to greater than one inch 32 years after the initial plant startup date using ASME Code, Section XI methods. In addition, a comparison between limited plant data from a 1989 startup and the duty used in Reference 1 was made in Reference 4. The results demonstrated that current plant cyclic duty was less severe than that previously assumed, as expected because of feedwater system modifications. Recommendations were made with respect to future work in this area so that acceptable crack growth for the 40 year design life of the reactor, and therefore full compliance with the requirements of NUREG-0619, could be shown.

The purpose of this report is to document an updated crack growth analysis using actual plant cyclic duty obtained during a Spring 1991 normal shutdown/startup sequence, and cycle counts obtained through Spring 1991. This analysis was initiated based on the recommendations made in Reference 4, as well as the Reference 4 demonstration that current plant cyclic duty was less severe than that previously assumed. The intent is to show full compliance with the requirements of NUREG-0619.

As documented in Reference 5, the methodology prescribed by NUREG-0619 is extremely conservative for the Hatch 2 feedwater nozzle evaluation. This conservatism results from the improved configuration of the Hatch 2 feedwater nozzles and sparger design as compared to the configurations on which NUREG-0619 was originally based.

NUREG-0619 was initiated in response to cracking observed in BWR feedwater nozzles. This cracking occurred in BWR plants with the loose (or slip) fit sparger design. In this configuration, cold incoming feedwater could leak past the juncture of the thermal sleeve and nozzle safe end and impinge upon the nozzle bore. Turbulent mixing of the hot water returning from the steam separators and dryers and the cold feedwater bypass leakage caused high frequency, thermal stress cycling of the feedwater nozzle bore surface metal. Both analytical and experimental studies confirmed that the observed cracking in BWR feedwater nozzles was initiated by the high frequency thermal cycling and the resulting fatigue. This bypass leakage was the basis for Generic Letter 81-11 requirements to modify the slip fit sparger designs.

In NUREG-0619, the NRC staff concurred that crack initiation was caused by high frequency thermal stress cycling. It was also noted in NUREG-0619 that stainless steel cladding contributed to fatigue cracking because the thermal stresses are higher than they would be in unclad metal. This is the basis of Generic Letter 81-11 requirements to remove the feedwater nozzle cladding.

It is known from analysis and from experience in repairing feedwater nozzles that an initiated crack will propagate due to high cycle fatigue to a depth of approximately 0.25 inch. At this depth, the amplitude of the high cycle thermal stresses attenuates to an insignificant level. On this basis, Generic Letter 81-11 requires that a 0.25-inch crack or flaw be assumed to exist and that licensees demonstrate that this postulated flaw will not grow to a size greater than one inch during the 40-year design life of the reactor.

The conservatism of NUREG-0619 as it relates to Hatch 2 is evident in several areas. First, Hatch 2 has had welded-in, single thermal sleeve spargers since the time of construction. Thus, the possibility of crack initiation due to bypass leakage is extremely small. Secondly, the Hatch 2 feedwater nozzles were installed as unclad. Thus, the contributing role of the stainless steel cladding in nozzle crack initiation has never existed.

Additionally, there have been routine non-destructive examinations, per NUREG-0619, of the Hatch 2 feedwater nozzles. The exam results indicate that

there has been no crack initiation or growth as predicted by NUREG-0619 methodology. For example, dye penetrant (PT) examinations of the accessible portions of the feedwater nozzle inner blend radius area on nozzles 2N4A and 2N4D were performed in 1985. Both inspections showed that no significant thermal fatigue cracks were present. There have also been several ultrasonic (UT) examinations of the inner blend radius and nozzle cylindrical bore section. These examinations have not detected any thermal fatigue crack indications. Consequently, there is no reason to suspect that fatigue cr are present in the Hatch 2 feedwater nozzles.

Based on the foregoing, the excessive conservatism of assuming an initial 0.25-inch flaw and the successful examination results should be considered when evaluating the results of this NUREG-0619 postulated crack growth analysis.

2.0 CYCLIC TEMPERATURE DEFINITIONS

The Reference 1 report provided thermal cycle definitions based on operating data from startup, shutdown and SCRAM events. These definitions were based on a review of plant recorder strip charts providing feedwater system temperatures and flow rates. Calculations were performed to determine the feedwater nozzle fluid temperature based on mixing of the reactor water cleanup (RWCU) and feedwater flows. A representative definition of a startup, shutdown and SCRAM event was developed which consisted of a series of temperature differences and a corresponding number of occurrences for each of these temperature differences.

Since the RWCU system was rerouted to both feedwater lines (as opposed to its original configuration of emptying into just one line), separate definitions were developed for the time periods representing before and after RWCU reroute. In addition, it was assumed that a plant shutdown was the mirror image of a startup (e.g., these events were identical with regard to cyclic duty). This assumption is conservative as the cycle definition for a shutdown is typically less severe than during a startup. Finally, only the nozzles experiencing RWCU injection prior to the RWCU reroute were analyzed as those nozzles experienced more severe duty. Thus, four cycle definitions were generated in the Reference 1 report and used in that crack growth analysis; these are repeated for convenience in this report in Tables 1 - 4.

In the Reference 4 analysis, the cycle definitions shown in Tables 1 - 4 were again used since those cycles were considered to be typical of plant operating duty and only limited new computer data were made available. This was conservative since the cycle definitions were determined to be more severe than those resulting from the modifications made to the feedwater system subsequent to 1983. Evidence of this conservatism was shown in Section 7.0 of the Reference 4 report.

For the current analysis, actual plant data were obtained from a normal shutdown/startup sequence before and after the Spring 1991 refueling outage. Special comput data acquisition equipment was implemented specifically for

this purpose, and existing plant instrumentation was tapped to acquire the data. All parameters which affect fluid flow through the feedwater nozzles were recorded by this equipment. This consisted of the following signals:

1. RWCU Temperature
2. RWCU Flow
3. Feedwater Temperature (Loops A and B)
4. Feedwater Flow (Loops A and B)
5. Reactor Pressure

These signals were recorded at a frequency of once every ten seconds, which is considered to be more than adequate for the purposes of this analysis. Data acquisition was started before any feedwater heaters were valved out of service during the shutdown, and recording continued for approximately a two day period (March 19-21, 1991) until cold shutdown was achieved. For the subsequent startup, data recording was initiated at the start of plant heatup, and ran continuously until all feedwater heaters were valved into service. This occurred over a period of approximately five days (May 30 - June 4, 1991). All data were permanently stored in data files and transmitted to GE (Reference 6) for incorporation into the crack growth analysis.

Again, calculations were performed on the collected computer data to determine the feedwater nozzle fluid temperature based on mixing of the reactor water cleanup (RWCU) and feedwater flows. Plots of the resulting data are provided in Appendix A. These data were then reduced into a series of temperature differentials grouped according to severity in 25°F increments, and a series of pressure differentials grouped according to severity in 100 psi increments. The temperature differentials consisted of starting at some temperature T_1, and proceeding to a final temperature T_2 where a change in direction of the temperature change occurred. The pressure differentials consisted of starting at some pressure P_1, and proceeding to a final pressure P_2 where a change in direction of the pressure change occurred. Any differentials less than 25°F or 100 psi were neglected to eliminate signal noise from consideration; these small magnitude cycles have an insignificant contribution to crack growth. A full cycle is defined in the crack growth

analysis as initially starting at some value T_0 (or P_0), changing to some other value T_1 (or P_1), and then returning to T_0 (or P_0). Therefore, the result of this data reduction was a series of half-cycles of different magnitudes.

A representative definition for the combined shutdown/startup event was developed for each feedwater loop which consisted of a series of temperature and pressure differences and a corresponding number of occurrences for each of these differences. Only minor differences existed between the two feedwater loops, and the definition which had the most temperature fluctuations (Loop A) was used for both loops. This shutdown/startup definition is provided in Table 5. Since SCRAM data were not available for computerized collection, the SCRAM definition previously used (from Table 4 - after RWCU reroute) was utilized for future cyclic duty projections in the crack growth analysis. This is considered conservative based on the comparison made between current plant duty and past plant duty in Section 7.0 of the Reference 4 report.

Therefore, for the purposes of calculating crack growth, three sets of data were considered: (1) the duty from Tables 1 and 2 for the period of operation before RWCU reroute, (2) the duty from Tables 3 and 4 for the period of operation after RWCU reroute and before feedwater system modifications, and (3) the duty from Tables 4 and 5 for the period of operation after feedwater system modifications until the end of the plant design life.

Table 1
Thermal Cycle Definition for a Startup Event
Before RWCU Reroute
(Obtained from Reference 1)

T_{max} (°F)	T_{min} (°F)	ΔT_n (°F)	Number of Cycles
424	100	324	1
436	137	299	1
436	180	256	12
436	212	224	2.5
436	233	203	4
395	220	175	3
323	182	141	1
265	136	129	3
266	144	122	7
272	157	115	4
272	163	109	9
316	213	103	7
267	173	94	7
244	164	80	6
216	152	64	12
228	170	58	9
231	180	51	32
257	215	42	7
201	171	30	16
			Total = 143.5

Notes: (1) One pressure cycle of 0 to 1000 psi is assumed.
(2) A shutdown event has the same cycle definition as this event.
(3) T_{max} = maximum fluid temperature, T_{min} = minimum fluid temperature, ΔT_n = nozzle temperature difference = T_{max}-T_{min}.

Table 2
Thermal Cycle Definition for a SCRAM Event
Before RWCU Reroute
(Obtained from Reference 1)

T_{max} (°F)	T_{min} (°F)	ΔT_n (°F)	Number of Cycles
436	327	109	6
436	268	168	4
436	225	211	8
436	174	262	4
436	147	289	3
			Total = 25

Notes:
(1) The above definition is for the cooldown portion of a SCRAM event only; the warmup portion has the same cycle definition as a startup event (Table 1).
(2) One pressure cycle of 0 to 1000 psi is assumed.
(3) T_{max} = maximum fluid temperature, T_{min} = minimum fluid temperature, ΔT_n = nozzle temperature difference = $T_{max}-T_{min}$.

Table 3
Thermal Cycle Definition for a Startup Event
After RWCU Reroute
(Obtained from Reference 1)

T_{max} (°F)	T_{min} (°F)	ΔT_n (°F)	Number of Cycles
436	102	334	1
436	119	317	1
436	143	293	8
436	169	267	4
436	182	254	5.5
436	205	231	2
335	185	150	1
268	142	126	1
545	436	109	1
226	137	89	14
217	134	83	3
214	134	80	10
225	150	75	1
229	161	68	3
208	145	63	6
196	143	53	5
174	128	46	5
197	155	42	13
178	144	34	36
207	183	24	10
172	156	16	11
			Total = 141.5

Notes:
(1) One pressure cycle of 0 to 1000 psi is assumed.
(2) A shutdown event has the same cycle definition as this event.
(3) T_{max} = maximum fluid temperature, T_{min} = minimum fluid temperature, ΔT_n = nozzle temperature difference = $T_{max} - T_{min}$.

Table 4
Thermal Cycle Definition for a SCRAM Event
After RWCU Reroute
(Obtained from Reference 1)

T_{max} (°F)	T_{min} (°F)	ΔT_n (°F)	Number of Cycles
436	339	97	1
436	268	168	5
436	212	224	4
436	179	257	8
436	134	302	7
			Total = 25

Notes:

(1) The above definition is fo. the cooldown portion of a SCRAM event only; the wa,mup portion has the same cycle definition as a startup event (Table 3).

(2) One pressure cycle of 0 to 1000 psi is assumed.

(3) T_{max} = maximum fluid temperature, T_{min} = minimum fluid temperature, ΔT_n = nozzle temperature difference = T_{max}-T_{min}.

Table 5

Thermal and Pressure Cycle Definition
for a Shutdown/Startup Event
(Obtained from Computer Data Acquisition During Spring 1991)

ΔT (°F)	Number of Half-Cycles
25 < ΔT ≤ 50	214
50 < ΔT ≤ 75	10
75 < ΔT ≤ 100	3
100 < ΔT ≤ 125	6
125 < ΔT ≤ 150	2
150 < ΔT ≤ 175	2
175 < ΔT ≤ 200	0
200 < ΔT ≤ 225	0
225 < ΔT ≤ 250	0
250 < ΔT ≤ 275	0
275 < ΔT ≤ 300	2
300 < ΔT	0
	Total = 239

ΔP (psi)	Number of Half-Cycles
100 < ΔP ≤ 200	3
200 < ΔP ≤ 300	0
300 < ΔP ≤ 400	0
400 < ΔP ≤ 500	1
500 < ΔP ≤ 600	0
600 < ΔP ≤ 700	0
700 < ΔP ≤ 800	0
800 < ΔP ≤ 900	0
900 < ΔP ≤ 1000	0
1000 < ΔP	2
	Total = 6

Notes:

(1) All temperature fluctuations less than 25°F, and all pressure fluctuations less than 100 psi in magnitude were neglected as the majority of these were attributed to signal noise and they have an insignificant impact on the crack growth evaluation.

(2) A half-cycle is defined as a change in temperature from T_1 to T_2 only (e.g., no return to T_1 associated with a full cycle). The same is true for pressure half-cycles.

3.0 PLANT OPERATING HISTORY

The Reference 1 report utilized plant operating history based on a review of the operator log books for the period of 1979 - 1982. During that four year period, a total of 79 startup/shutdown and SCRAM/return to service events were recorded. Extrapolating those events to 40 years, a total of 790 events was used in that crack growth analysis.

The Reference 4 report utilized the same event definitions used in Reference 1, but considered the number of thermal events experienced by the Hatch 2 reactor for the time period up to 1990, as provided in Reference 7. Extrapolating those events to 40 years, a total of 322 events was used in that crack growth analysis.

GPC has continued to keep track of the reactor thermal events since 1985. A summary of the thermal events experienced by the Hatch 2 reactor was provided for the time period of 1985 - 1991. These data were provided in the Reference 7 and 8 correspondences, and are shown in Table 6. This information demonstrates that events experienced during the 1979 - 1982 time period occurred more frequently than in the 1985 - 1991 time period. Therefore, for conservatism, the 1979 - 1982 events were used for estimating the number of events which occurred during the 1982 - 1985 time period where adequate cycle count information was not available. The 1985 - 1991 events were used to predict cycle counts for the remaining 40-year plant life.

The information shown in Table 6 was used to determine an updated projection for the number of events for the 40 year design life of the plant. This new projection is shown in Table 7. It is seen that the extrapolated number of events for the 40 year design life of the reactor (102 + 198 = 300) is significantly lower than the 790 events previously obtained in the Reference 1 report. It is also lower than the 322 events projected in the Reference 4 report. These differences demonstrate continued improvement in plant operation with respect to thermal duty, and are consistent with field observations made regarding cyclic duty. Reference 9 documents the fact that there are typically more thermal events during the initial years of plant

operation to account for "learning curve" effects. The methods used in determining the projected values in Table 7 are consistent with those outlined in Reference 9.

The projected number of events shown in Table 7 are used in the updated crack growth analysis contained herein.

Table 6
Plant Operating History Summary
1985 - 1991
(Obtained from References 7 and 8)

Year	Number of Startups	Number of Shutdowns	Number of SCRAMs
1985	2	2	2
1986	4	4	5
1987	1	1	4
1988	3	3	6
1989	1	1	0
1990	0	0	2
1991	2	2	0
Totals:	13	13	19

Notes:
(1) In 1985, the period covered was between 8/1/85 and 12/31/85.
(2) In 1991, the period covered was between 1/1/91 and 7/31/91.
(3) The total time period covered by this data (8/1/85 - 7/31/91) was assumed to be equal to 6.0 years.

Table 7

Projected Number of Events for the 40 Year Design Life

Time Period	No. of Years	Number of Startups	Number of Shutdowns	No. of SCRAMs
1979 - 1985	6	28	28	90
1985 - 1989	4	11	11	17
remaining life	30	63	63	91
Totals:	40	102	102	198

Notes:

(1) For the time period of 1979 - 1985, the following calculations were performed to determine the number of cycles:

# of startups from Reference 1 = 79 - 60 = 19

# of years of data in Reference 1 = 4

# of years for 1979 - 1985 time period = 6

# of startups for 1979 - 1985 time period = 19 x (6/4) = 28

Also: #shutdowns = 19x(6/4) = 28 and #SCRAMS = 60x(6/4) = 90

(2) For the time period of 1985 - 1989, the number of cycles was obtained from Table 6 (References 7 and 8).

(3) For the remaining life, the following calculations were performed to determine the number of cycles:

# of startups during 1985 - 1991 = 13

# of years of data during 1985 - 1991 = 6

# of years for remaining life = 34

# of startups for 1991 - 40-year life = 13 x (34/6) = 74

# of startups during 1985 - 1989 = 11

# of startups for remaining life = 74 - 11 = 63

Also: #shutdowns = 13x(34/6)-11 = 63 and #SCRAMS = 19x(34/6)-17 = 91

(4) The cyclic definitions before RWCU reroute (Tables 1 - 2) will be conservatively applied to the entire 1979 - 1985 time period in the crack growth analysis. The definitions after RWCU reroute (Tables 3 - 4) are applied to the 1985 - 1989 time period. The shutdown/startup definition obtained during Spring 1991 (Table 5) and the SCRAM definition after RWCU reroute (Table 4) are applied to the remaining life period.

(5) One SCRAM event is considered to include both the SCRAM and the following return to service.

4.0 FINITE ELEMENT ANALYSIS RESULTS

A detailed finite element model of the Hatch 2 feedwater nozzle configuration was developed in the Reference 1 report in order to develop temperature distributions as well as thermal and pressure stresses for subsequent use in the crack growth analysis. Those results were also used in the Reference 4 analysis, and remain valid for use in the current analysis. As a result, they are used without modification herein. The stresses for the most limiting nozzle corner section as given in Reference 1 are shown in Table 8.

The pressure stresses shown in Table 8 are for the case of a 1,000 psi vessel pressure, scaled to account for the differences of modeling a three dimensional structure with a two dimensional model. The scaled pressure stresses shown in Table 8 are linearly scaled to the ΔP described in the thermal cycle definitions shown in Tables 1 - 5.

The thermal stresses shown in Table 8 are those resulting from a step transient of 450°F (temperature ramped from 550°F to 100°F in 9 seconds). The stresses shown are those which are present in the nozzle structure 45 seconds after the temperature drop. In the Reference 1 report, these stresses were used for all temperature cycles in the crack growth analysis with the exception of one (corresponding to a drop in feedwater temperature to the steady state condition present at shutdown). For this one cycle per event, another stress profile corresponding to the stress state present 180 seconds after the analyzed temperature drop was used. It was found in the Reference 4 analysis that using this 180-second stress profile for one cycle per event had an insignificant impact on the crack growth analysis results. Therefore, for simplicity, only the stress profile corresponding to 45 seconds after the analyzed temperature drop was used in the Reference 4 crack growth analysis. That same thermal stress profile was also used in the crack growth analysis contained herein, as identified in Table 8. The thermal stresses shown in Table 8 (which developed from a ΔT of 450°F) are linearly scaled to the ΔT described in the thermal cycle definitions shown in Tables 1 - 5. The scaled thermal stresses are subsequently used along with the pressure stresses in the crack growth analysis contained herein.

Table 8
Thermal and Pressure Stresses for the Limiting Nozzle Cross Section (Obtained from Reference 1)

Distance from Inside Surface (inches)	Pressure Hoop Stress Ratioed by 1.5987 (psi)	Thermal Hoop Stress (t = 45 seconds) (psi)
0.000	33589	44541
0.075	33015	41319
0.225	31926	34551
0.400	30716	27370
0.600	29421	20086
0.850	27881	12762
1.150	26302	6011
1.500	24310	1113
1.954	22754	-2503
2.693	19039	-3528
3.284	17001	-3313
3.874	15136	-2815
4.465	13373	-2316
5.056	11653	-1856
5.647	9902	-1419
6.247	8040	-994
6.833	5944	-568
7.127	4860	-356

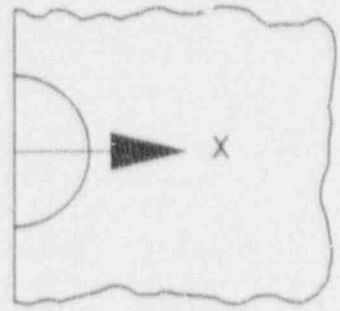
Notes: (1) The pressure hoop stresses were scaled by the 1.5987 factor to account for the differences of modeling a three dimensional structure with a two-dimensional model (see Reference 1).

5.0 STRESS INTENSITY FACTOR CALCULATIONS

As in References 1 and 4, stress intensity factors were calculated using solutions for standard stress distributions in half and quarter space. The stress intensity solutions for specific crack geometries are shown in Figure 1. It was recognized that the solution for a 3-D nozzle corner crack lies in between the half and quarter space solutions, so those solutions were averaged to obtain the nozzle solution. Further support of this assumption is provided in Appendix B.

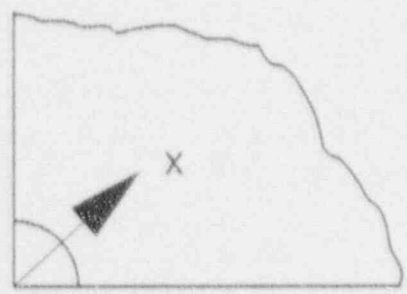
The pressure and thermal stress distributions shown in Table 8 were fit to third order polynomials using a standard least squares procedure. The fit was checked to ensure accuracy for the region of interest (at least the first 1" of cross section). The polynomial coefficients were then substituted into the simulated three dimensional nozzle corner crack stress intensity factor expression of Figure 1. The resulting stress intensity factor versus crack depth results are plotted in Figures 2 and 3.

The stress intensity relationships used to generate Figures 2 and 3 are used in the updated crack growth analysis contained herein.



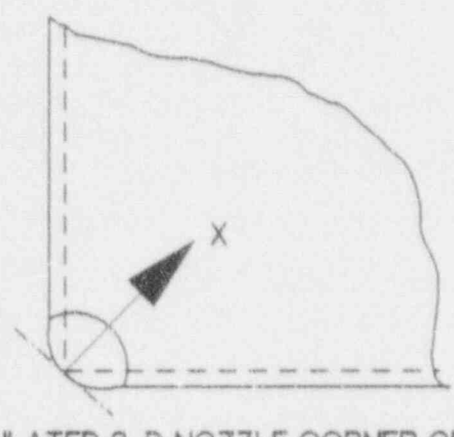
SEMI-CIRCULAR CRACK IN HALF-SPACE

$$K_I = \sqrt{\pi a}\,[0.688A_0 + 0.522(2a/\pi)A_1 + 0.434(a^2/2)A_2 + 0.377(4a^3/3\pi)A_3]$$



QUARTER-CIRCULAR CRACK IN QUARTER SPACE

$$K = \sqrt{\pi a}\,[0.723A_0 + 0.551(2a/\pi)A_1 + 0.462(a^2/2)A_2 + 0.408(4a^3/3\pi)A_3]$$



SIMULATED 3-D NOZZLE CORNER CRACK

$$K_I = \sqrt{\pi a}\,[0.706A_0 + 0.537(2a/\pi)A_1 + 0.448(a^2/2)A_2 + 0.393(4a^3/3\pi)A_3]$$

Figure 1: Boundary Integral Equation/Influence Function Magnification Factors for a BWR Feedwater Nozzle

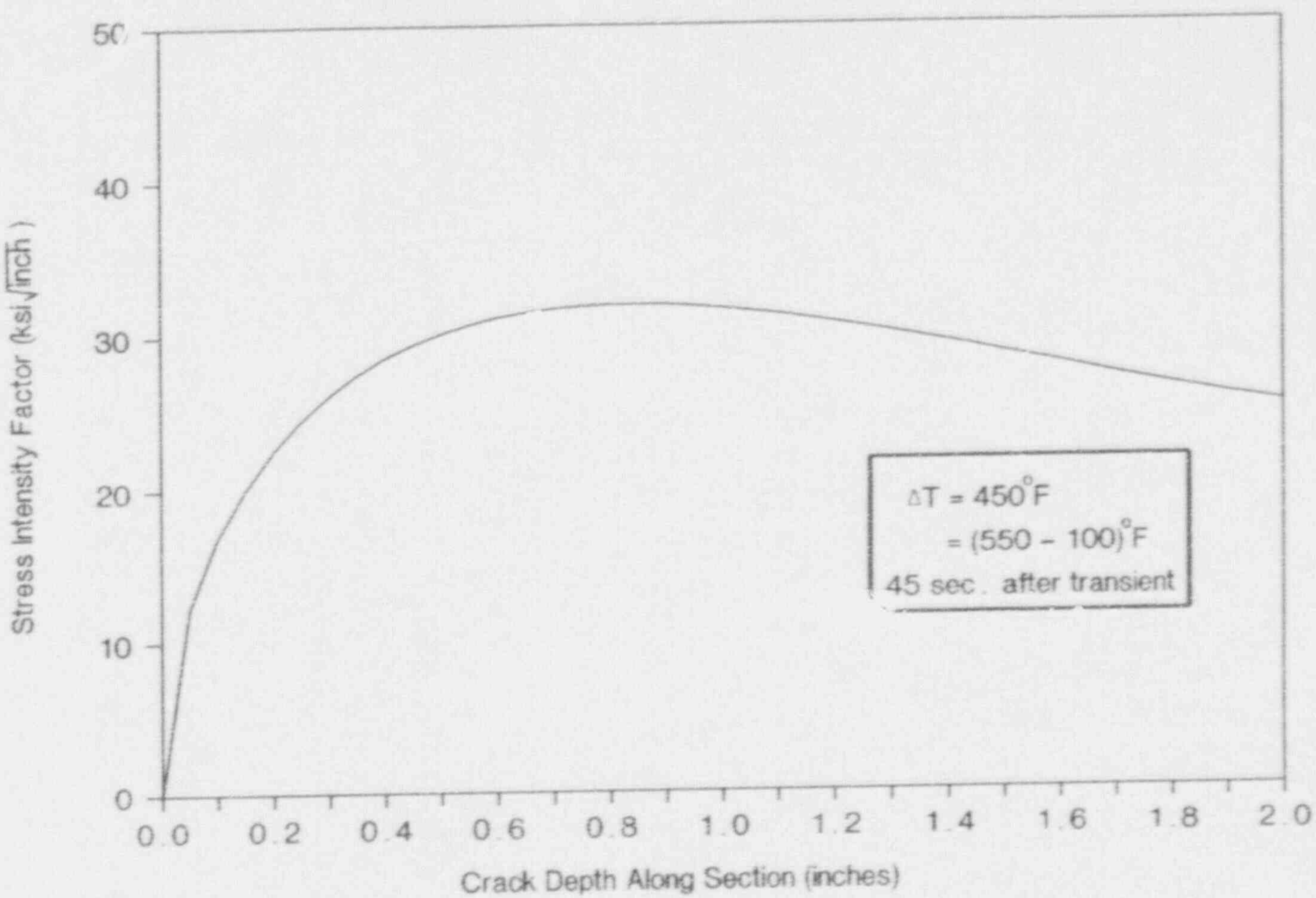


Figure 2: Stress Intensity vs. Crack Depth
(Thermal Stresses, 45 seconds)

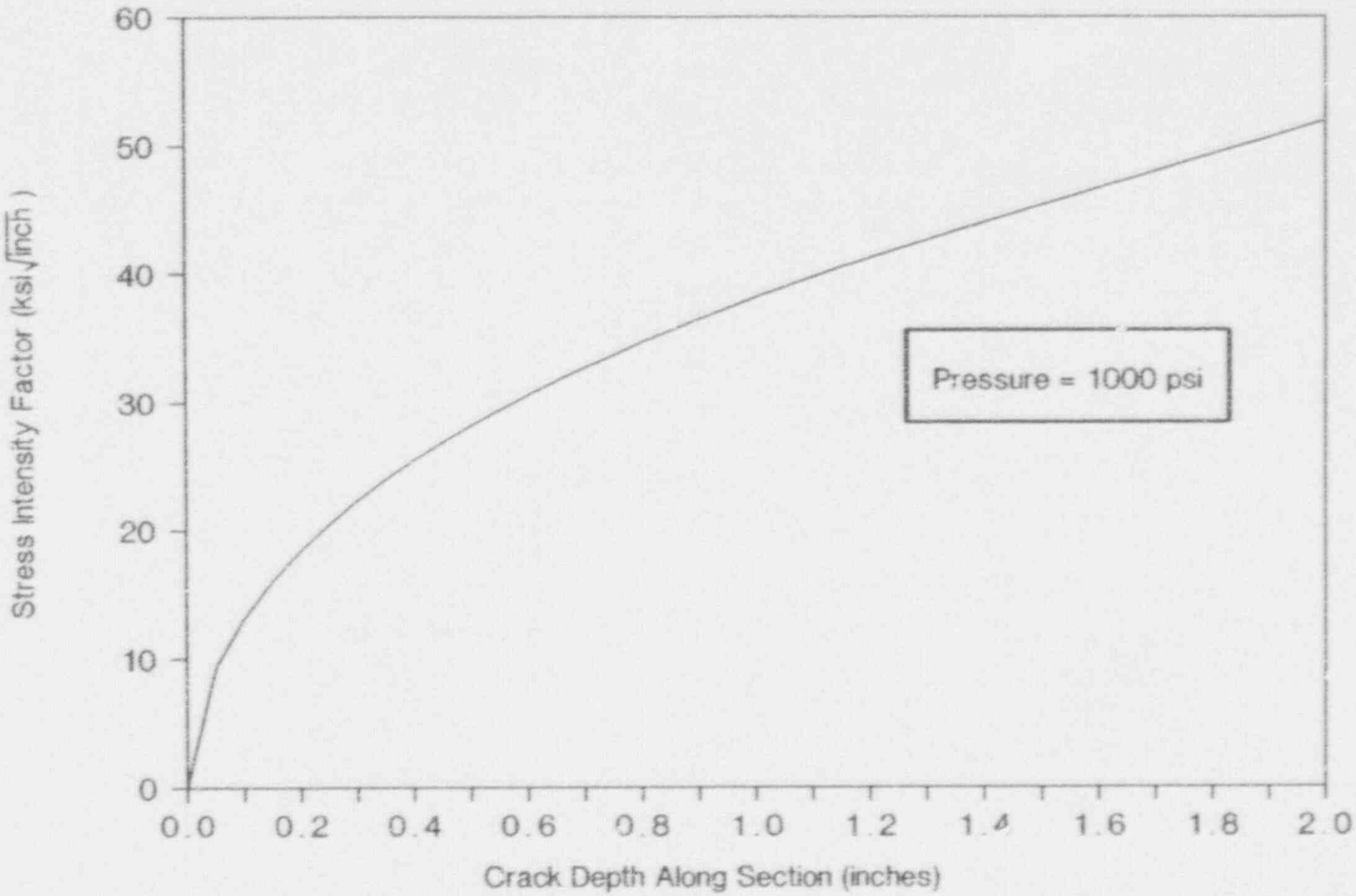


Figure 3: Stress Intensity vs. Crack Depth
(Pressure Stresses)

6.0 CRACK GROWTH EVALUATION

The fatigue crack growth rate data for low alloy steel from Section XI of the ASME Code (Reference 10) were used for the crack growth analysis. This curve is the same as was used in the Reference 1 and 4 reports. In the Reference 1 report, a best fit relationship was also used in the crack growth analysis. The best fit results were reported for information and comparative purposes only, and provided significantly improved results over those obtained from an ASME Code, Section XI approach. Nevertheless, the best fit relationship was not considered in the Reference 4 analysis since the ASME Code curves are the accepted criteria for analyses of this type. Therefore, the crack growth relationships from the ASME Code are used exclusively in the updated crack growth analysis contained herein.

The procedure for calculating the crack propagation is as follows: For each cycle, the maximum and minimum stress and the number of occurrences were calculated. From this, the stress intensity factor range and the corresponding R-ratio were calculated for each cycle. Using this and the ASME Code crack growth relationships, the incremental crack growth was calculated for each cycle. The crack size was updated and the procedure was repeated for all cycles until all events had been analyzed. The pattern of thermal events was assumed to be as follows for the analysis:

1979 - 1985 ==> 7 sets of:
- 4 startup/shutdown events (Table 1)
  followed by
- 13 SCRAM/return to full power events (Table 2)

1985 - 1989 ==> 6 sets of:
- 2 startup/shutdown events (Table 3)
  followed by
- 3 SCRAM/return to full power events (Table 4)

remaining life ==> 3 sets of — 21 startup/shutdown events (Table 5) followed by 30 SCRAM/return to full power events (Table 4)

Thus, the total number of events analyzed was as follows:

	Startups	Shutdowns	SCRAMs
1979 - 1985	7 x 4 = 28	7 x 4 = 28	7 x 13 = 91
1985 - 1989	6 x 2 = 12	6 x 2 = 12	6 x 3 = 18
remaining life	3 x 21 = 63	3 x 21 = 63	3 x 30 = 90
Totals:	103	103	199

These total number of analyzed events can be compared to the totals projected in Table 7. Although conservative, the effects of analyzing the one extra startup/shutdown event (103 vs. 102) and the one extra SCRAM event (199 vs. 198) are considered insignificant.

The results of the crack growth evaluation are shown in Figure 4. These results show that, using the ASME Code, Section XI crack growth relationships, a postulated 0.25 inch initial depth crack (as specified in NUREG-0619) becomes 0.96 inch deep at the end of the 40-year plant design life. As anticipated, these results are significantly improved from the Reference 1 results, and further improved from the Reference 4 results. This improvement is due to the significantly lower number of cyclic events projected for the 40 year design life of the plant, as well as improved cyclic thermal duty resulting from RWCU reroute and feedwater system modifications.

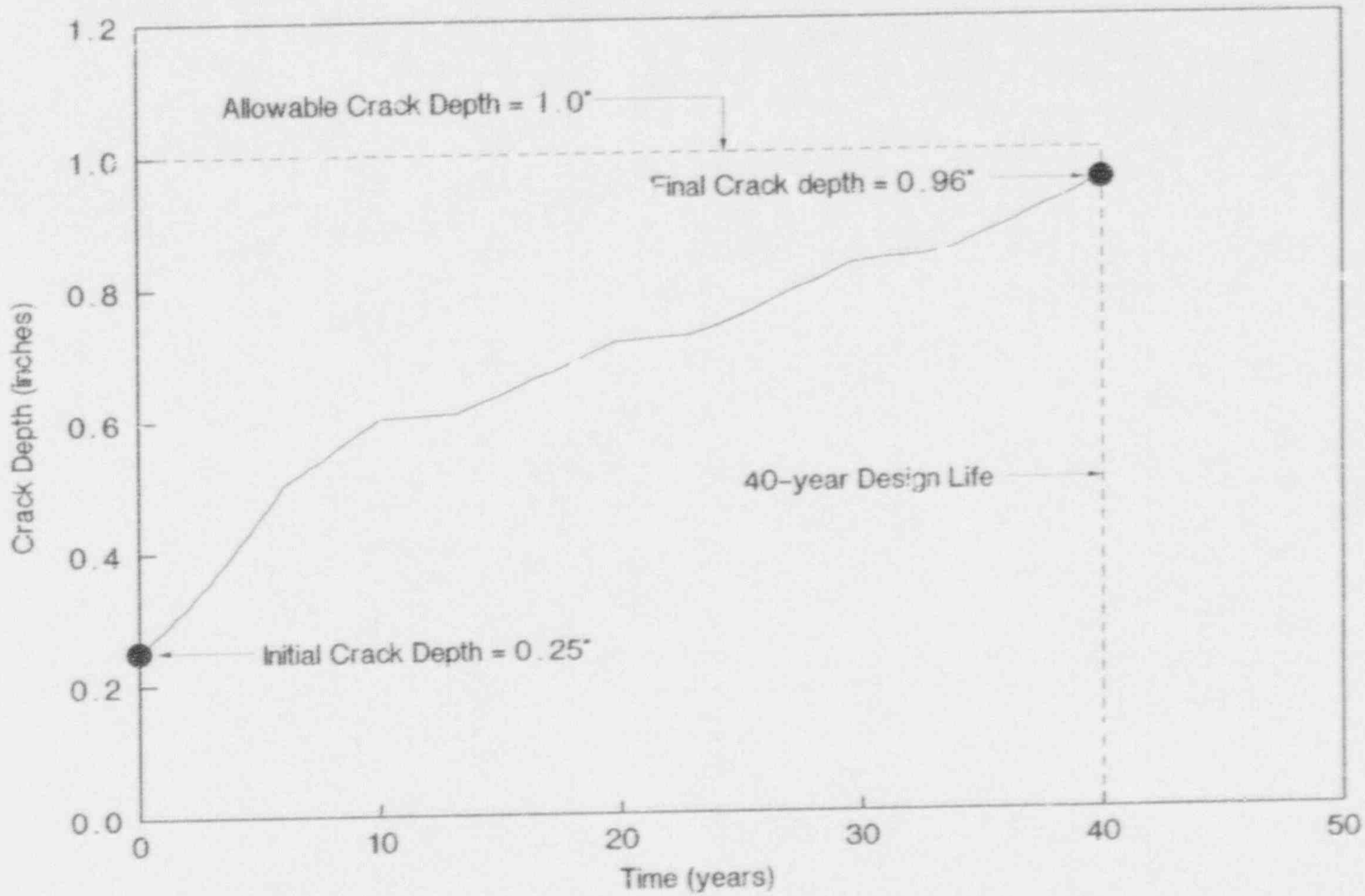


FIGURE 4: Updated Crack Growth Analysis Results

7.0 SUMMARY AND CONCLUSIONS

The Reference 1 and 4 crack growth analyses for Hatch 2 were reevaluated using an updated cycle count projection and data obtained during the Spring 1991 normal shutdown/startup sequence. Application of previously determined cycle severity (for the period 1979 - 1989), current cycle severity (for the remainder of the plant design life), thermal and pressure stresses, and ASME Code Section XI crack growth relationships to the cycle projection resulted in the postulated crack growth of an assumed 0.25 inch initial depth flaw to 0.96 inch at the end of the 40-year design life of the plant.

The results of this report show that the crack growth analysis requirements of NUREG-0619 are fully satisfied. Further confirmation of the Hatch 2 nozzle integrity has been provided in the past from ultrasonic test (UT) and dye penetrant test (PT) inspections performed at intervals specified in NUREG-0619. These examinations, which have occurred over the past several refueling outages at Hatch 2, have revealed no indications. A program of continued thermal duty monitoring and inspections consistent with the provisions of NUREG-0619 will ensure continued integrity of the nozzle region and provide further validation of these results.

8.0 REFERENCES

[1] D.J. Liffengren, et. al., NEDC-30256, "Edwin I. Hatch Nuclear Power Station, Unit 2, Feedwater Nozzle Fracture Mechanics Analysis", General Electric Company, August 1983.

[2] NUREG-0619, "BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking", Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, November 1980.

[3] Generic Letter 81-11 to all Power Reactor Licensees from Darrell Eisenhut, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, February 28, 1981.

[4] SASR 1290-HT2, Revision 0, "Updated Feedwater Nozzle Fracture Mechanics Analysis for Edwin I. Hatch Nuclear Power Station, Unit 2," G.L. Stevens, GE Nuclear Energy, December 1990.

[5] SASR 86-38, DRF 137-0010, "Response to the NRC Questions with Regard to the Hatch 2 Feedwater Nozzle NUREG 0619 Report", G.L. Stevens, General Electric Company, June 16, 1986.

[6] Letter from William F. Garner (Southern Company Services, Inc.) to D.R. Madison (Southern Nuclear Operating Company), "E.I. Hatch Nuclear Plant - Units 2, Feedwater Shutdown Data Acquisition Data," File REA HT-90764, EWO H527 AS, Log REA-1-6-0621, June 28, 1991.

[7] Letter from S.J. Bethay (GPC) to R.P. Daly (GE), "Hatch Unit 2 Feedwater Nozzle Fracture Mechanics Analysis", Log HL-1378, November 20, 1990.

[8] Letter from S.J. Bethay (GPC) to R.P. Daly (GE), "Hatch Unit 2 Feedwater Nozzle Fracture Mechanics Analysis", Log HL-1767, August 8, 1991.

[9] Services Information Letter (SIL) Number 318, "BWR Reactor Vessel Cyclic Duty Monitoring", December 1979.

[10] ASME Boiler and Pressure Vessel Code, Section XI, "Rules for In-Service Inspection of Nuclear Power Plant Components", Appendix A, "Analysis of Flaws", 1989 Edition.

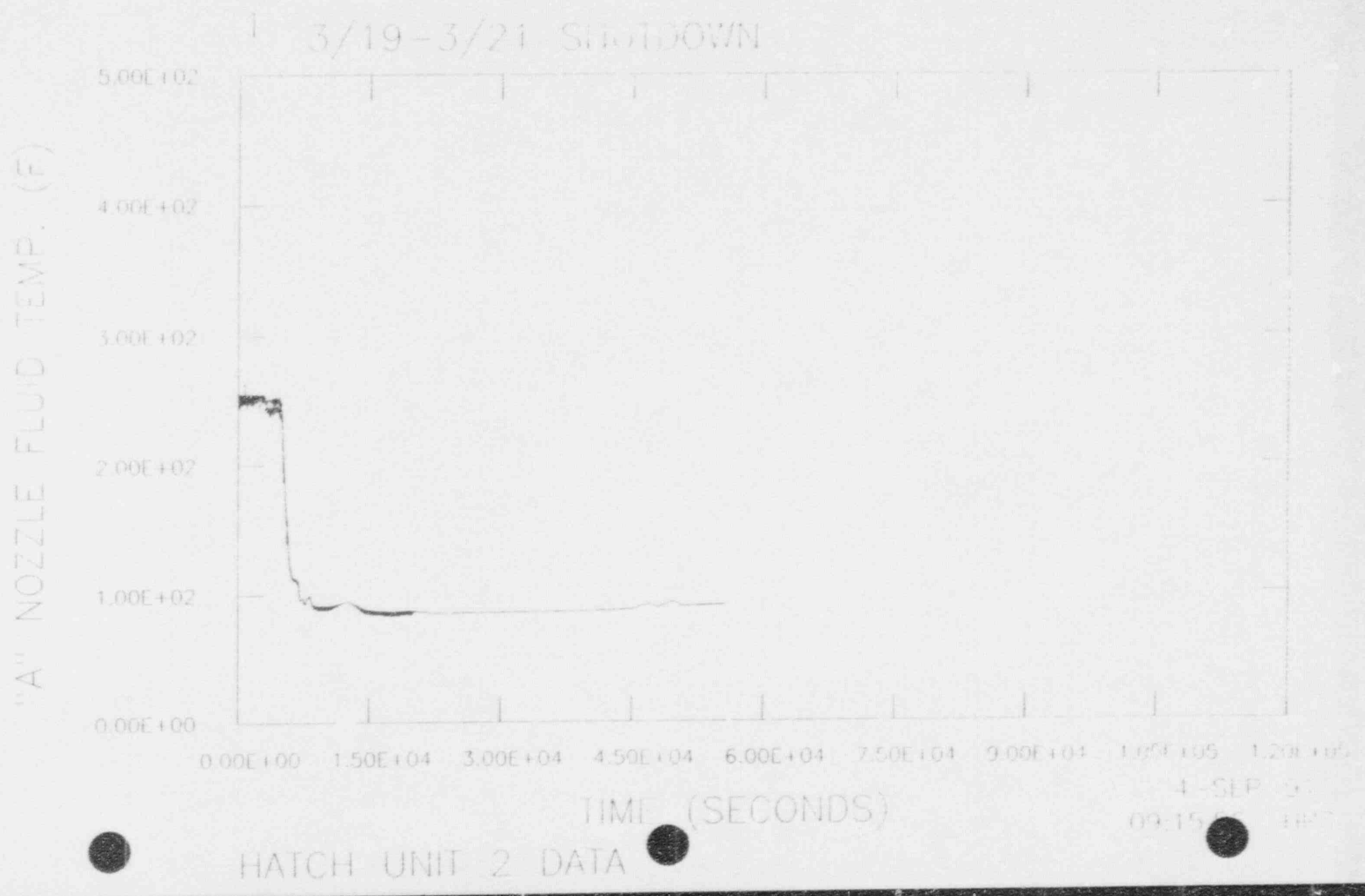
APPENDIX A

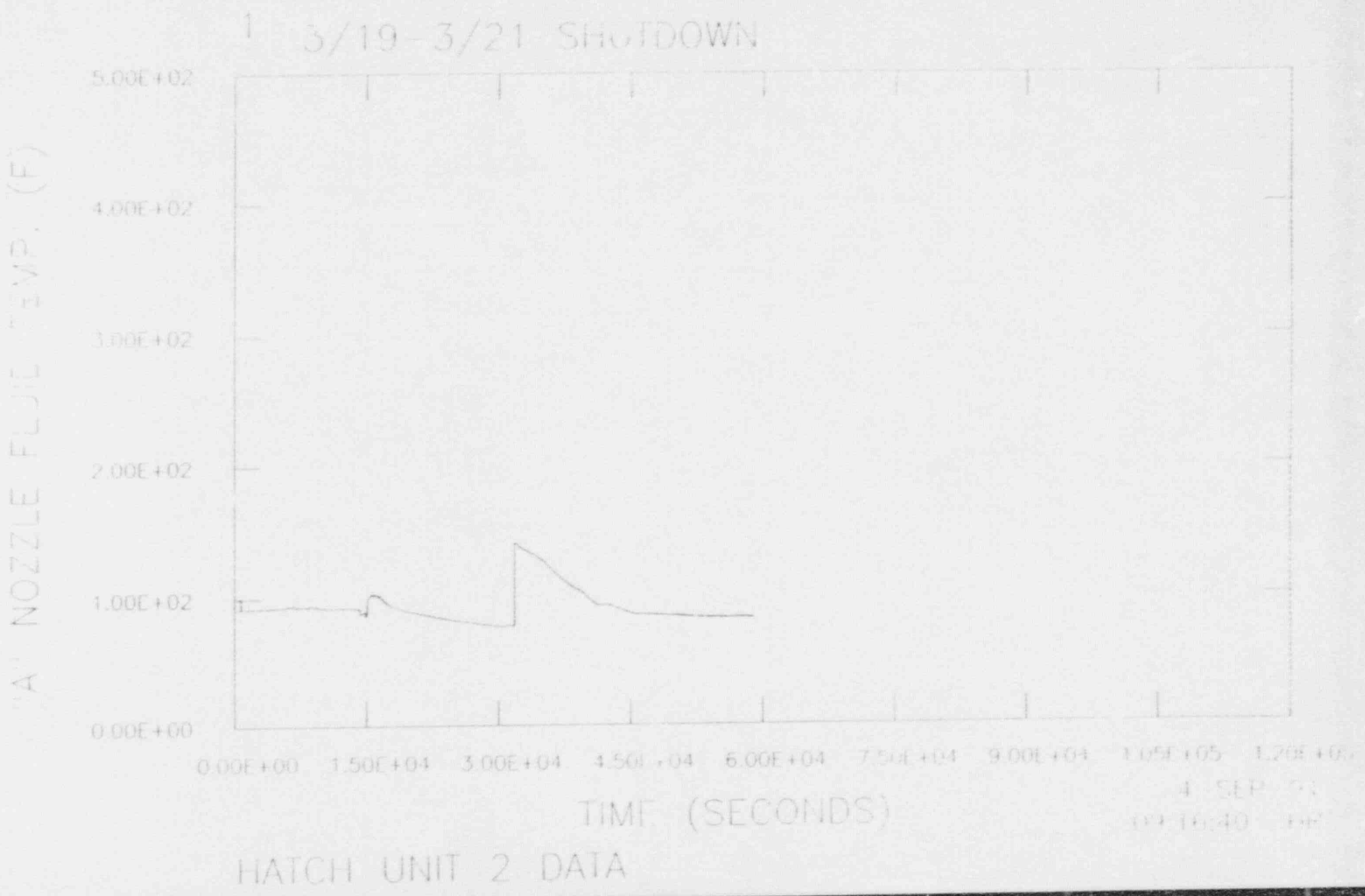
FEEDWATER NOZZLE TEMPERATURE, PRESSURE AND FLOW DATA

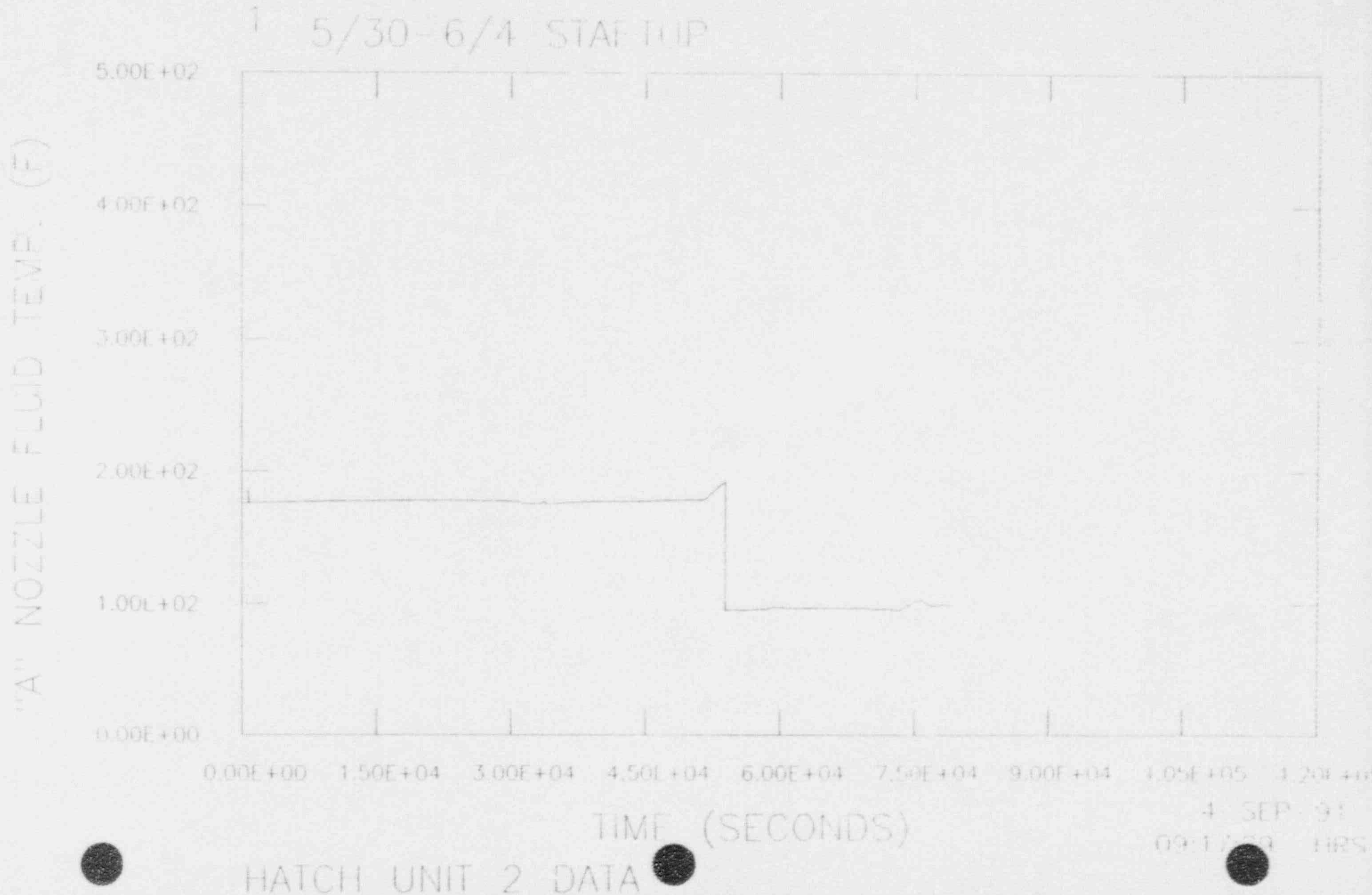
This Appendix contains plots of the calculated (mixed) nozzle fluid temperature and loop flow as a function of time for both Loops A and B for the Spring 1991 startup/shutdown sequence at Hatch 2. Also included are plots of reactor pressure as a function of time for the same time period. The time period covered by the plots corresponds to the following dates:

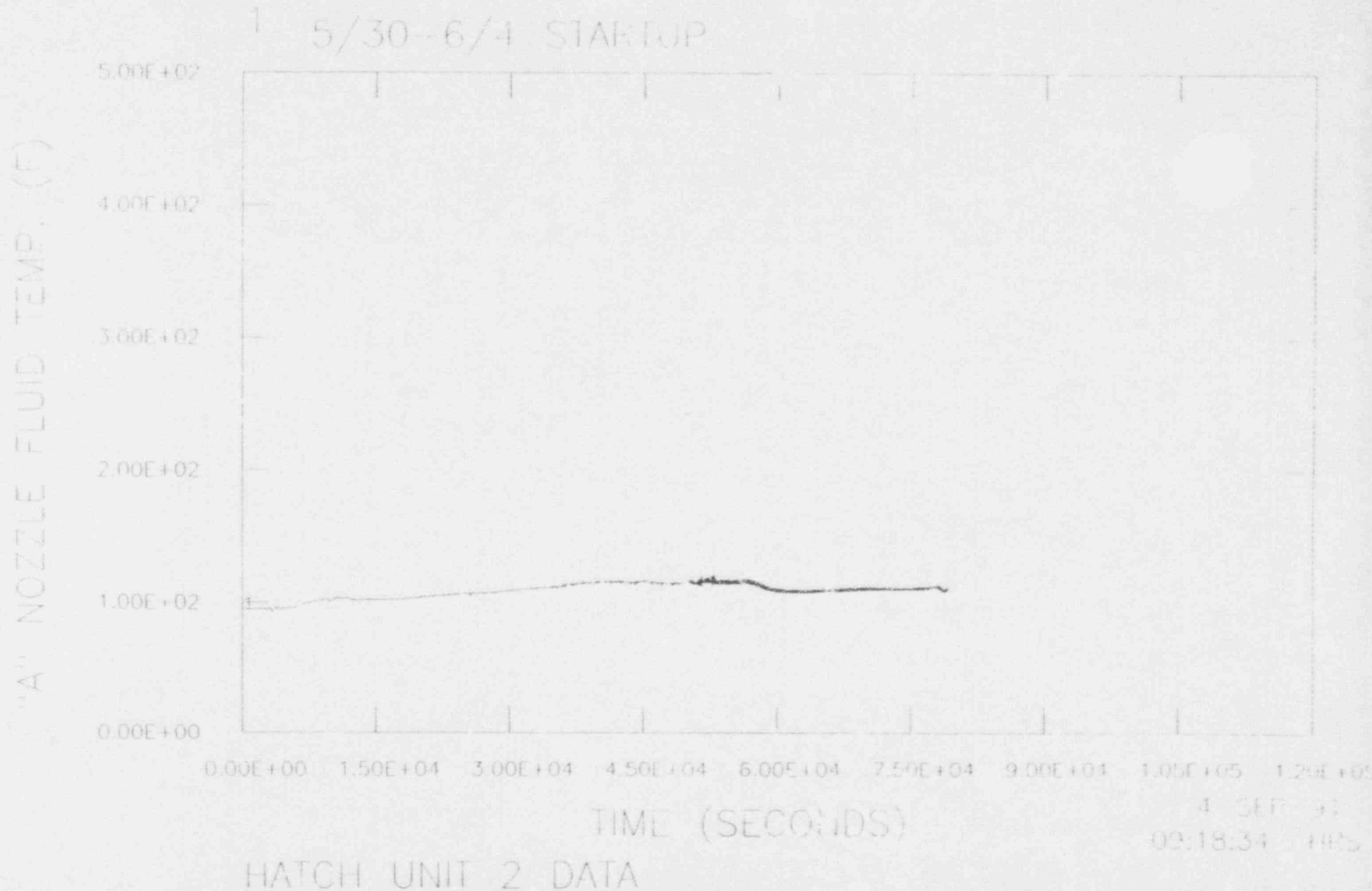
No.	Plot Identifier	Event	Period Covered
1	SD-1	3/19 - 3/21 Shutdown	3/19/91 @ 23:27 to 3/20/91 @ 15:00
2	SD-2	3/19 - 3/21 Shutdown	3/20/91 @ 15:00 to 3/21/91 @ 07:23
3	SU-1	5/30 - 6/04 Startup	5/30/91 @ 14:00 to 5/31/91 @ 12:00
4	SU-2	5/30 - 6/04 Startup	5/31/91 @ 15:00 to 6/01/91 @ 10:00
5	SU-3	5/30 - 6/04 Startup	6/01/91 @ 10:00 to 6/02/91 @ 08:00
6	SU-4	5/30 - 6/04 Startup	6/02/91 @ 08:00 to 6/03/91 @ 06:00
7	SU-5	5/30 - 6/04 Startup	6/03/91 @ 06:00 to 6/04/91 @ 04:00
8	SU-6	5/30 - 6/04 Startup	6/04/91 @ 04:00 to 6/04/91 @ 15:13

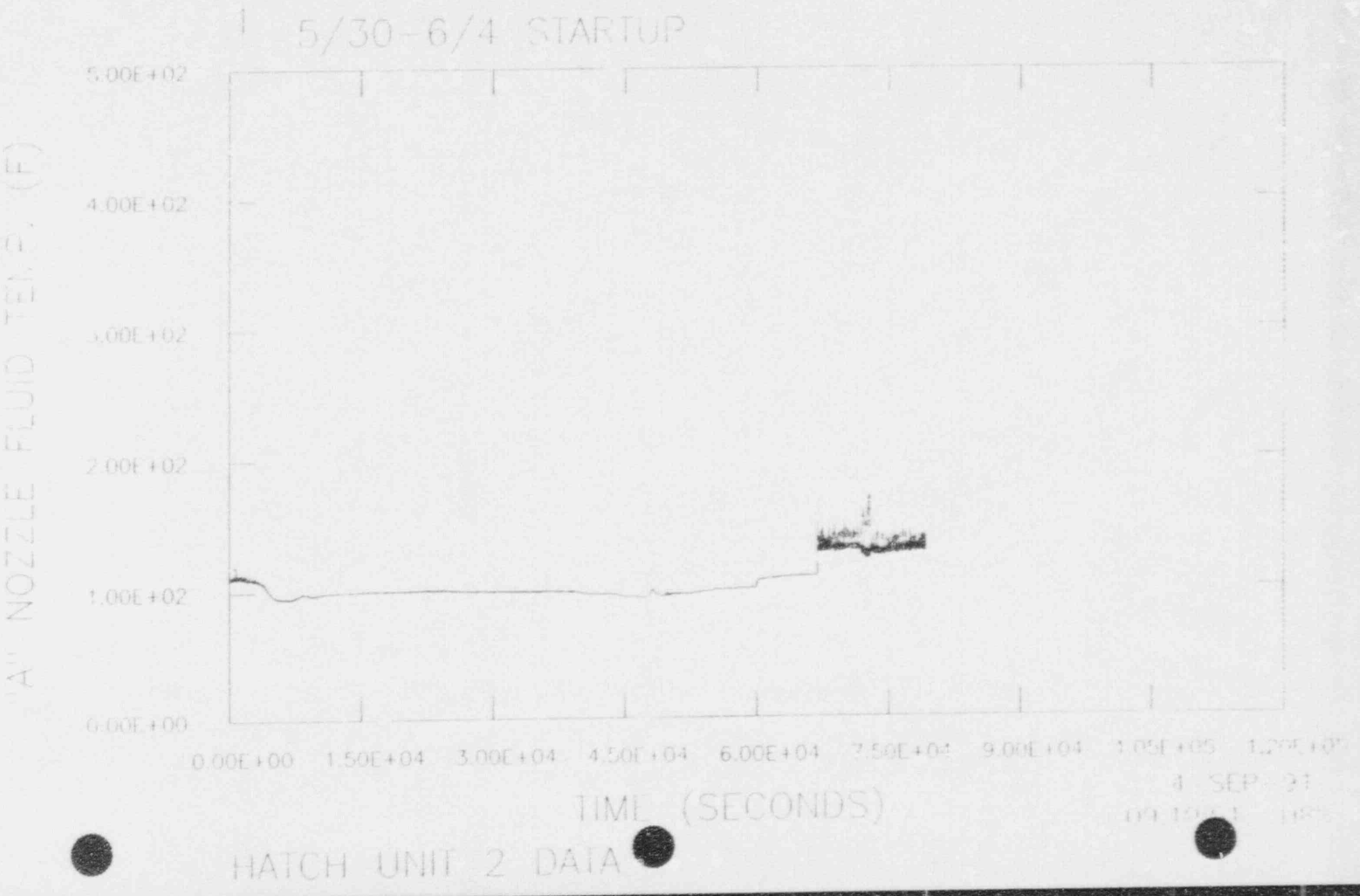
A.1 Loop A Mixed Fluid Temperature Plots

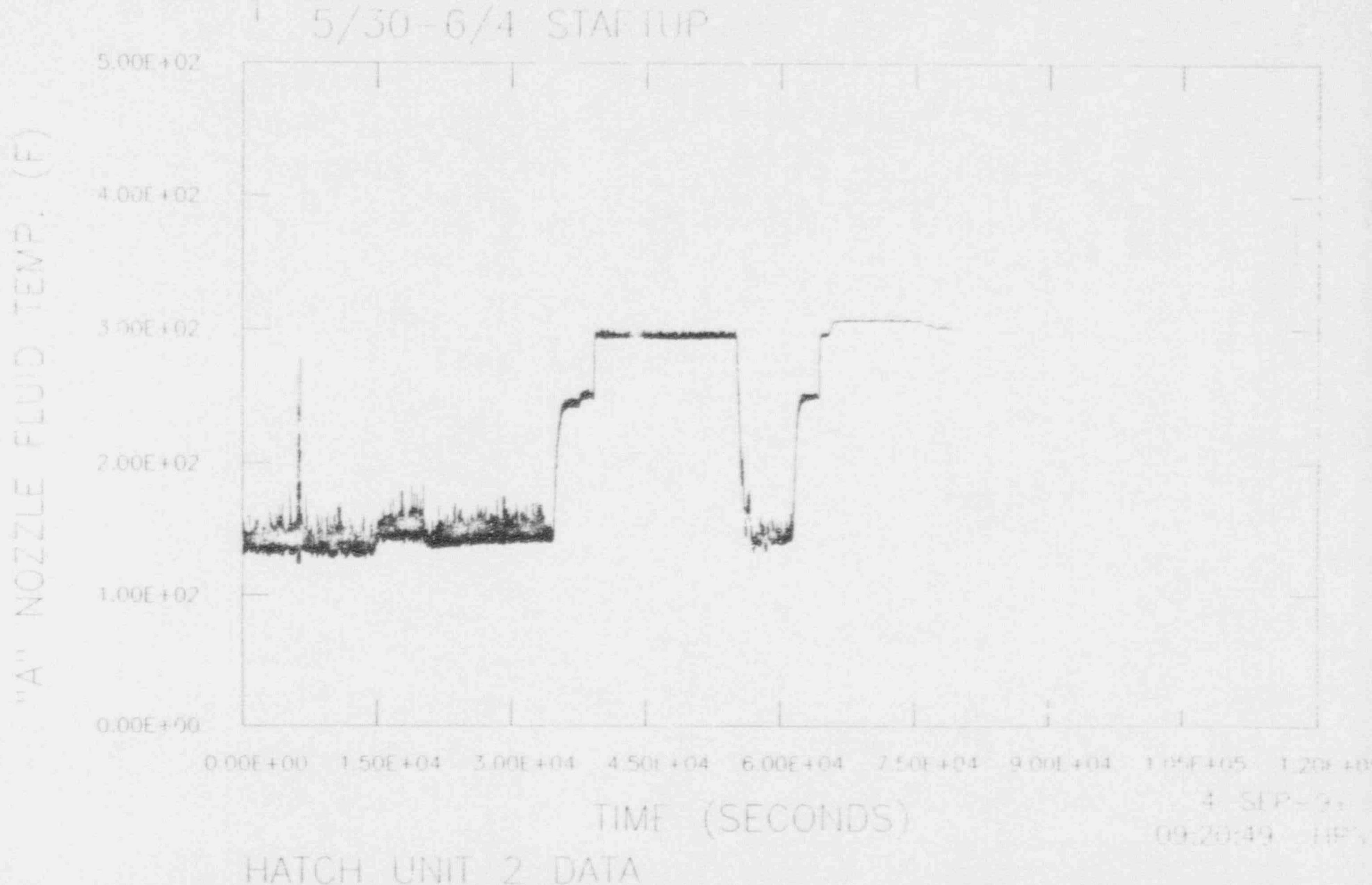

3/19-3/21 SHUTDOWN
"A" NOZZLE FLUID TEMP. (F)
5.00E+02
4.00E+02
3.00E+02
2.00E+02
1.00E+02
0.00E+00
0.00E+00
1.50E+04
3.00E+04
4.50E+04
6.00E+04
7.50E+04
9.00E+04
TIME (SECONDS)
HATCH UNIT 2 DATA

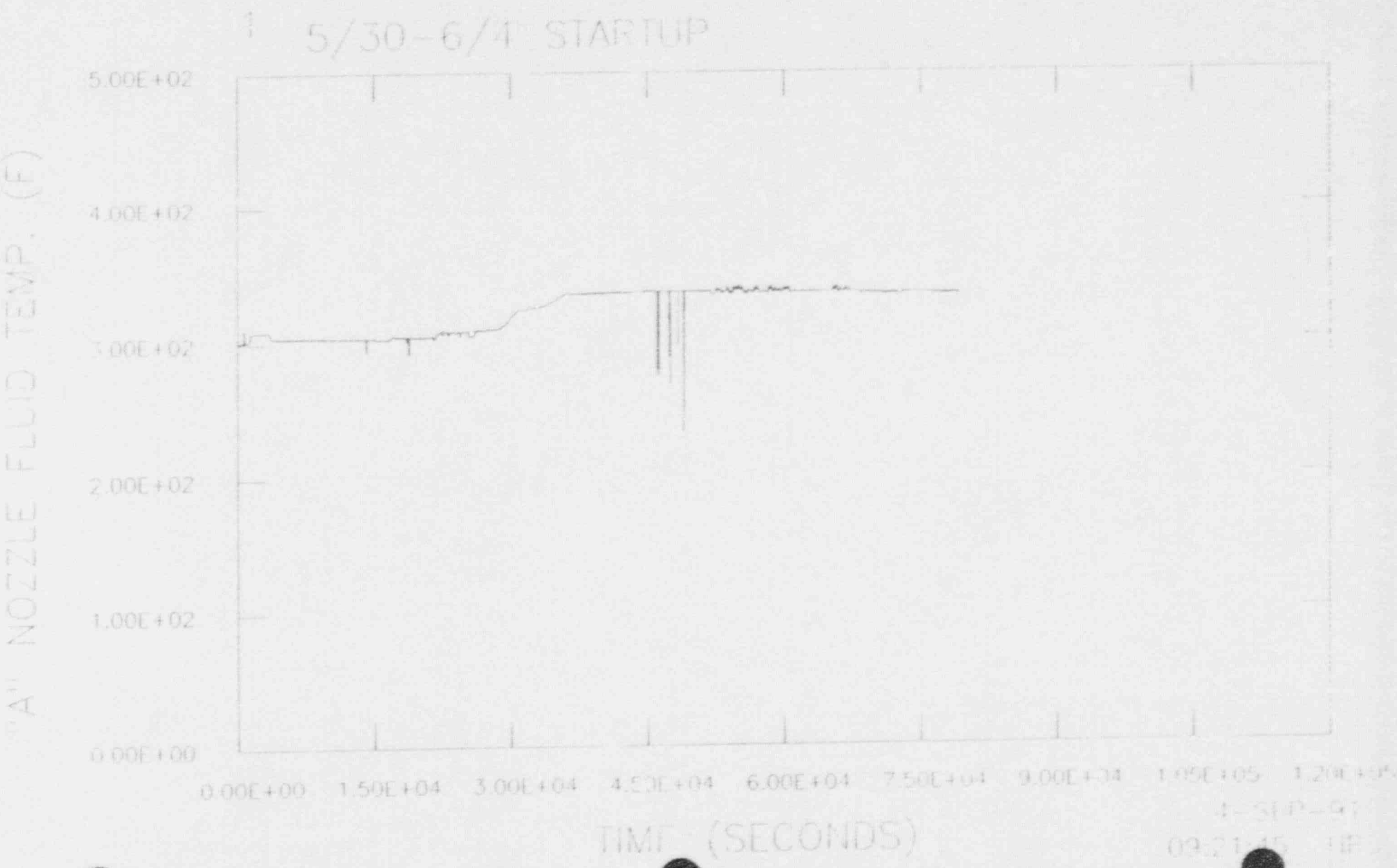

1
3/19-3/21 SHUTDOWN
'A' NOZZLE FLUID TEMP. (F)
5.00E+02
4.00E+02
3.00E+02
2.00E+02
1.00E+02
0.00E+00
0.00E+00
1.50E+04
3.00E+04
4.50E+04
6.00E+04
7.50E+04
9.00E+04
1.05E+05
TIME (SECONDS)
HATCH UNIT 2 DATA


1 5/30-6/4 STARTUP
"A" NOZZLE FLUID TEMP. (F)
5.00E+02
4.00E+02
3.00E+02
2.00E+02
1.00E+02
0.00E+00
0.00E+00
1.50E+04
3.00E+04
4.50E+04
6.00E+04
7.50E+04
9.00E+04
1.05E+05
TIME (SECONDS)
4 SEP 91
HRS
HATCH UNIT 2 DATA

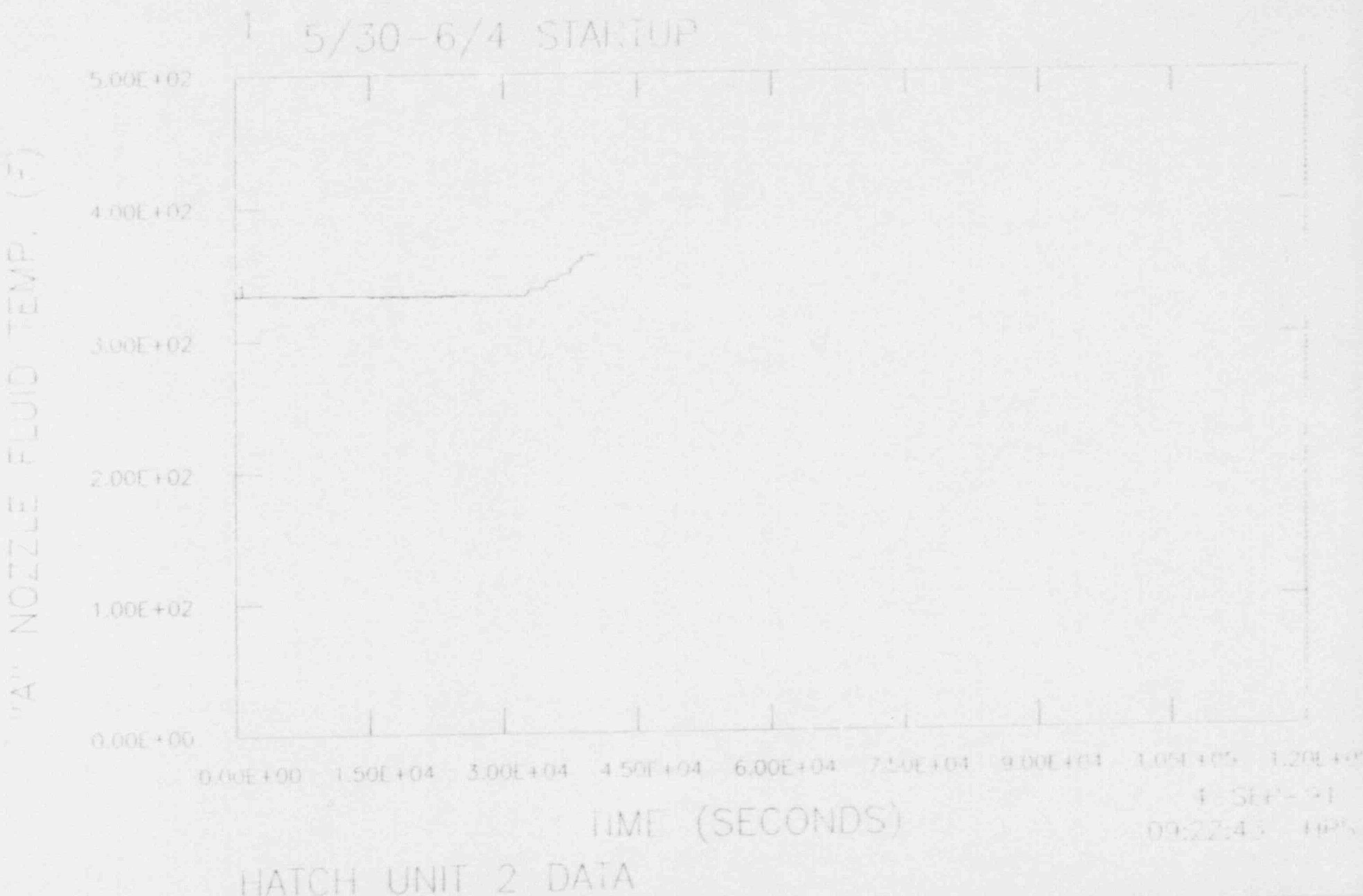

1
5/30--6/4 STARTUP
"A" NOZZLE FLUID TEMP. (F)
5.00E+02
4.00E+02
3.00E+02
2.00E+02
1.00E+02
0.00E+00
0.00E+00
1.50E+04
3.00E+04
4.50E+04
6.00E+04
7.50E+04
9.00E+04
1.05E+05
1.20E+05
TIME (SECONDS)
09:18:34 HRS
HATCH UNIT 2 DATA


5/30-6/4 STARTUP
"A" NOZZLE FLUID TEMP. (F)
5.00E+02
4.00E+02
3.00E+02
2.00E+02
1.00E+02
0.00E+00
0.00E+00
1.50E+04
3.00E+04
4.50E+04
6.00E+04
7.50E+04
9.00E+04
1.05E+05
TIME (SECONDS)
4 SEP 91
HATCH UNIT 2 DATA


5/30-6/4 STARTUP
"A" NOZZLE FLUID TEMP. (F)
5.00E+02
4.00E+02
3.00E+02
2.00E+02
1.00E+02
0.00E+00
0.00E+00
1.50E+04
3.00E+04
4.50E+04
6.00E+04
7.50E+04
9.00E+04
1.05E+05
1.20E+05
TIME (SECONDS)
HATCH UNIT 2 DATA



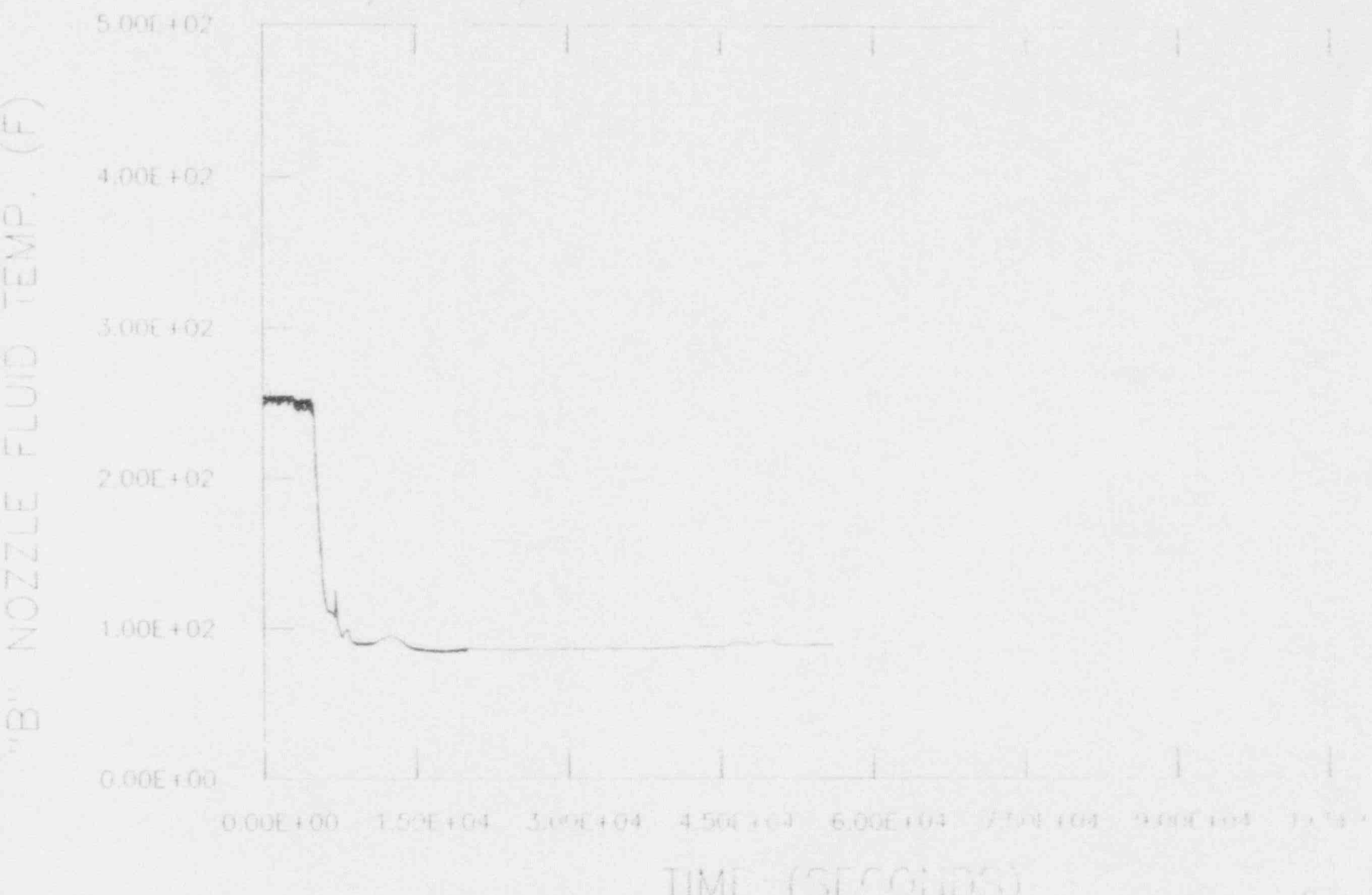
HATCH UNIT 2 DATA



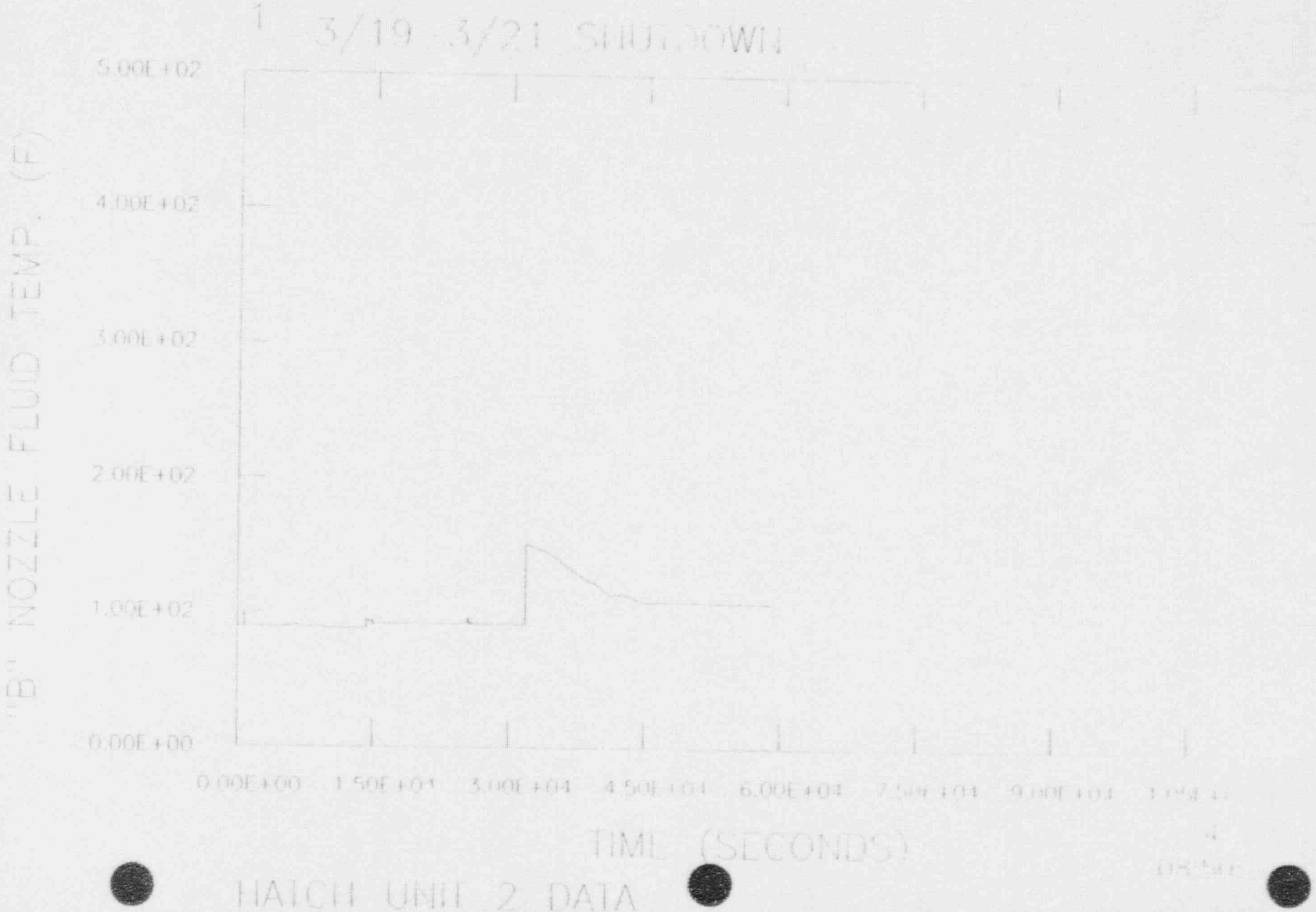
1
5/30-6/4 STARTUP
"A" NOZZLE FLUID TEMP. (F)
5.00E+02
4.00E+02
3.00E+02
2.00E+02
1.00E+02
0.00E+00
0.00E+00
1.50E+04
3.00E+04
4.50E+04
6.00E+04
7.50E+04
9.00E+04
1.05E+05
1.20E+05
TIME (SECONDS)
HATCH UNIT 2 DATA

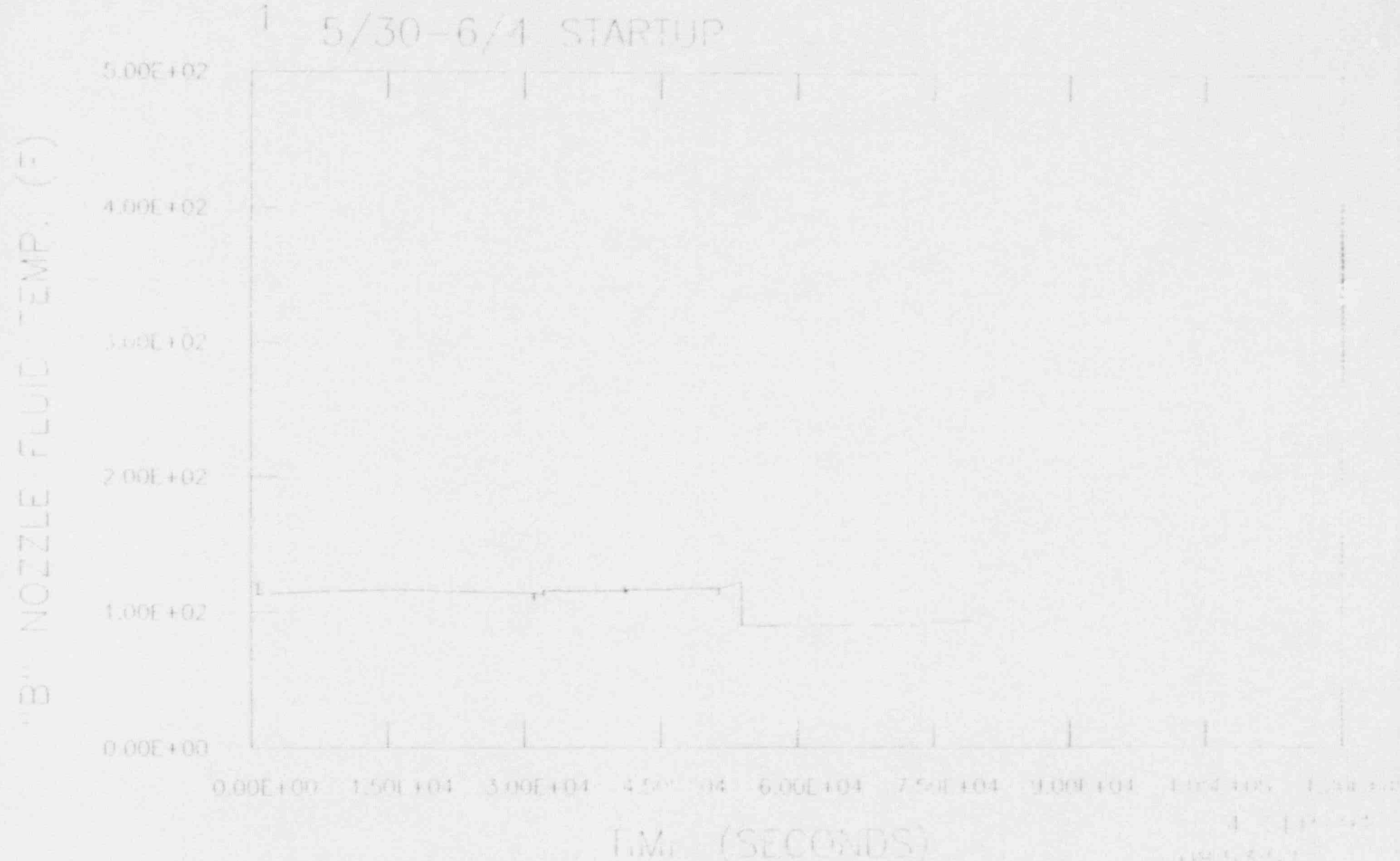
A.2 Loop B Mixed Fluid Temperature Plots

3/19 - 3/21 SHUTDOWN

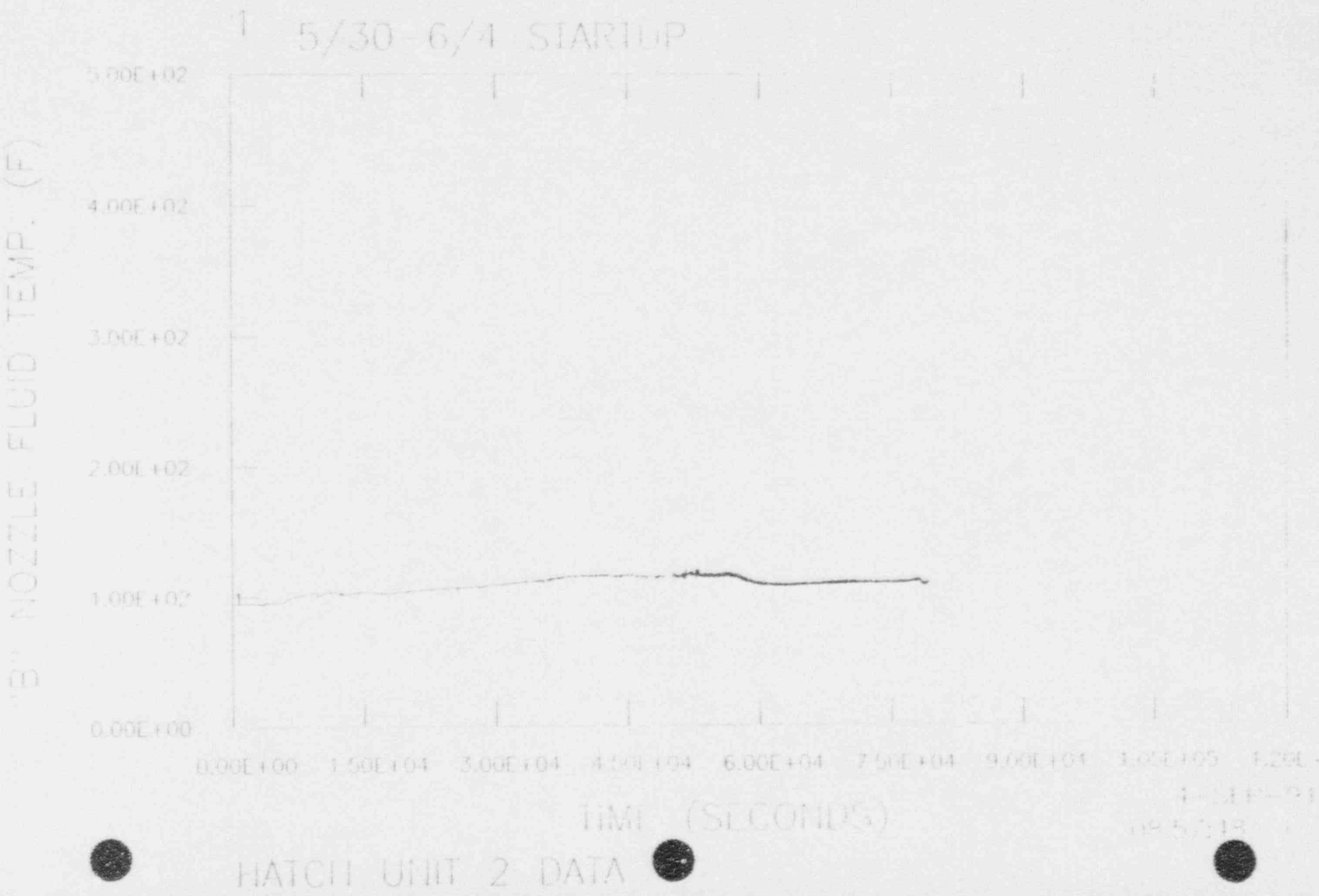


HATCH UNIT 2 DATA

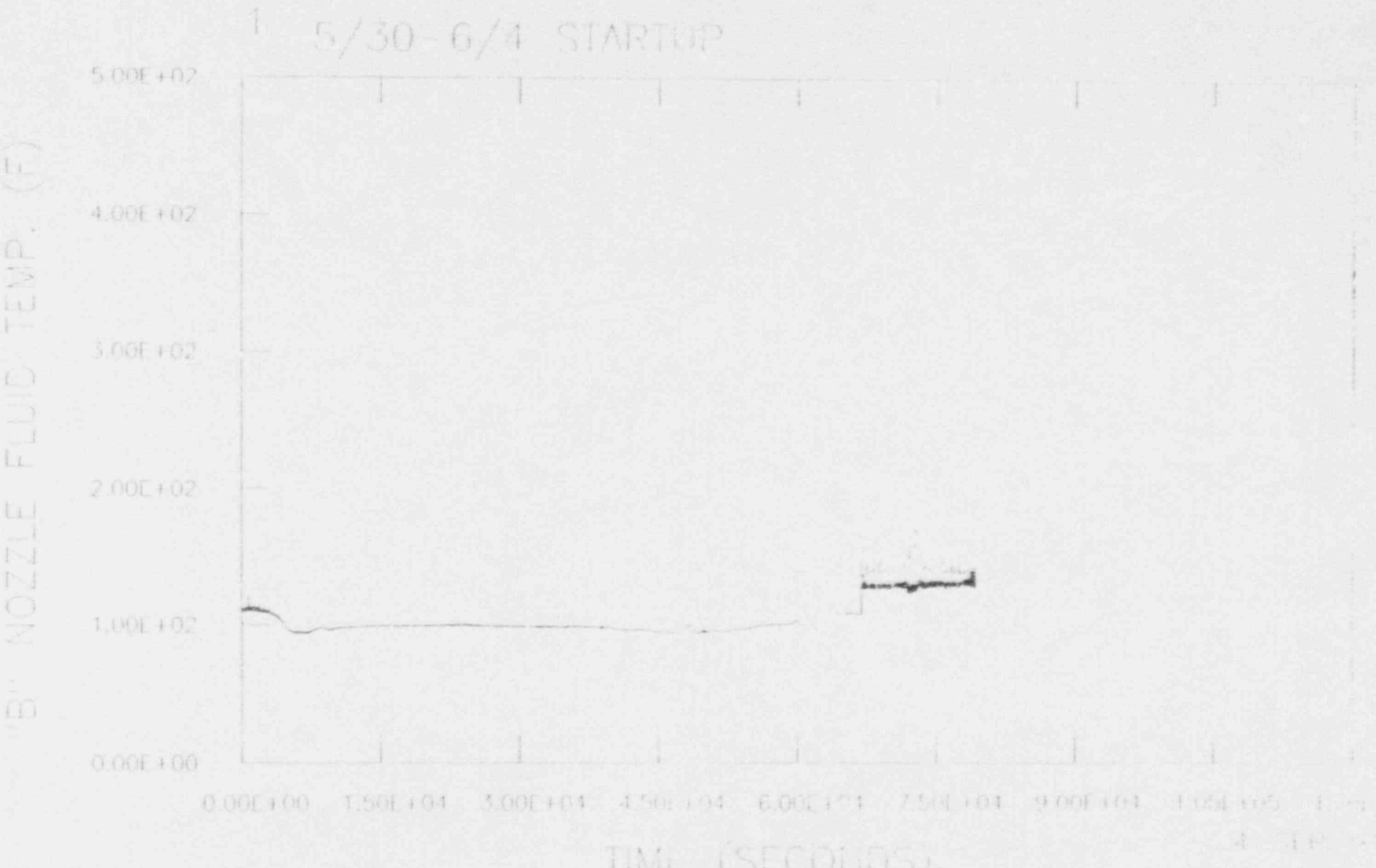

1 3/19 3/21 SHUTDOWN
"B" NOZZLE FLUID TEMP. (F)
5.00E+02
4.00E+02
3.00E+02
2.00E+02
1.00E+02
0.00E+00
0.00E+00
1.50E+04
3.00E+04
4.50E+04
6.00E+04
TIME (SECONDS)
HATCH UNIT 2 DATA



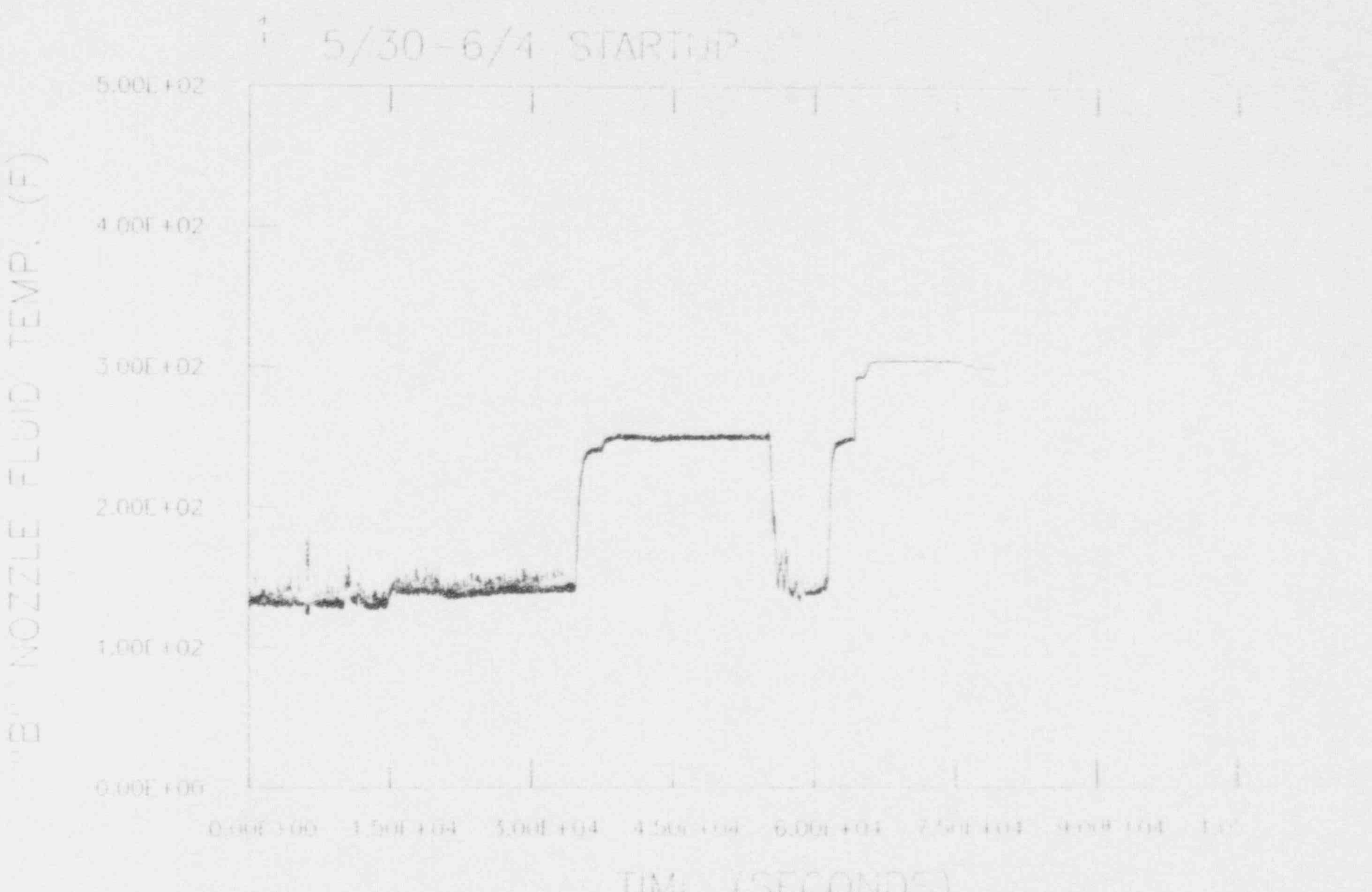
HATCH UNIT 2 DATA



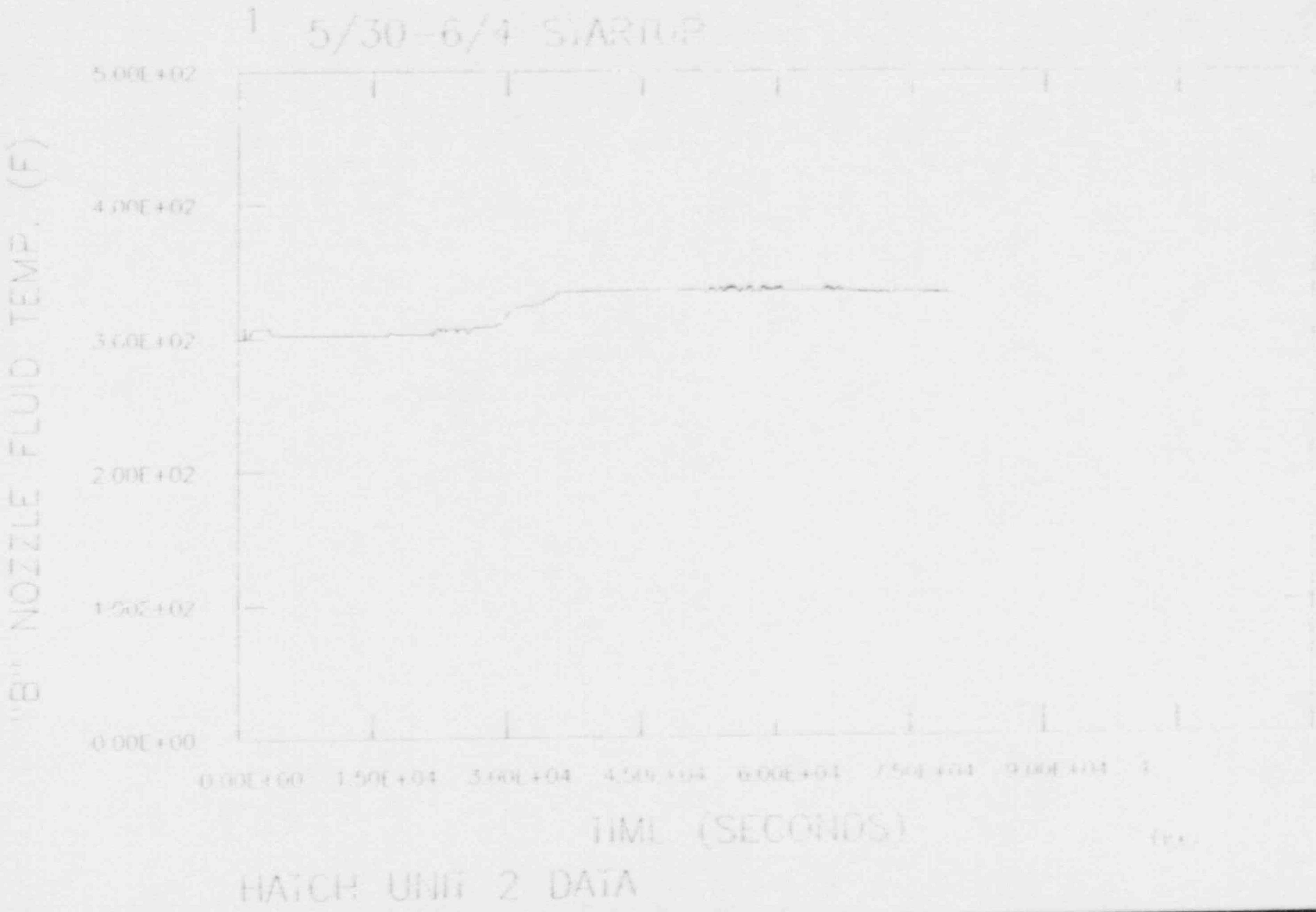
1 5/30-6/4 STARTUP
8" NOZZLE FLUID TEMP. (F)
5.00E+02
4.00E+02
3.00E+02
2.00E+02
1.00E+02
0.00E+00
0.00E+00
1.50E+04
3.00E+04
6.00E+04
7.50E+04
9.00E+04
TIME (SECONDS)
HATCH UNIT 2 DATA



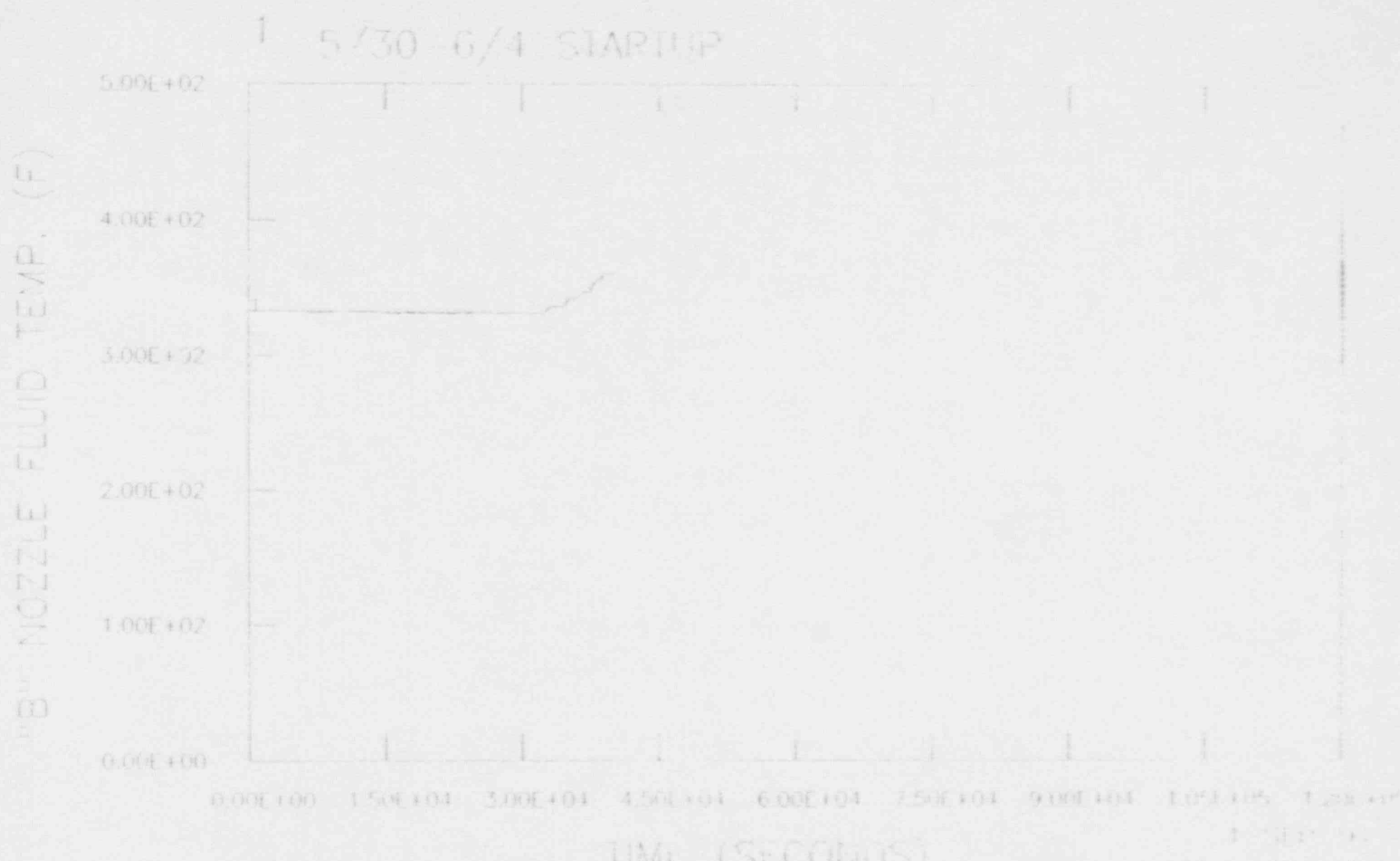
HATCH UNIT 2 DATA



HATCH UNIT 2 DATA

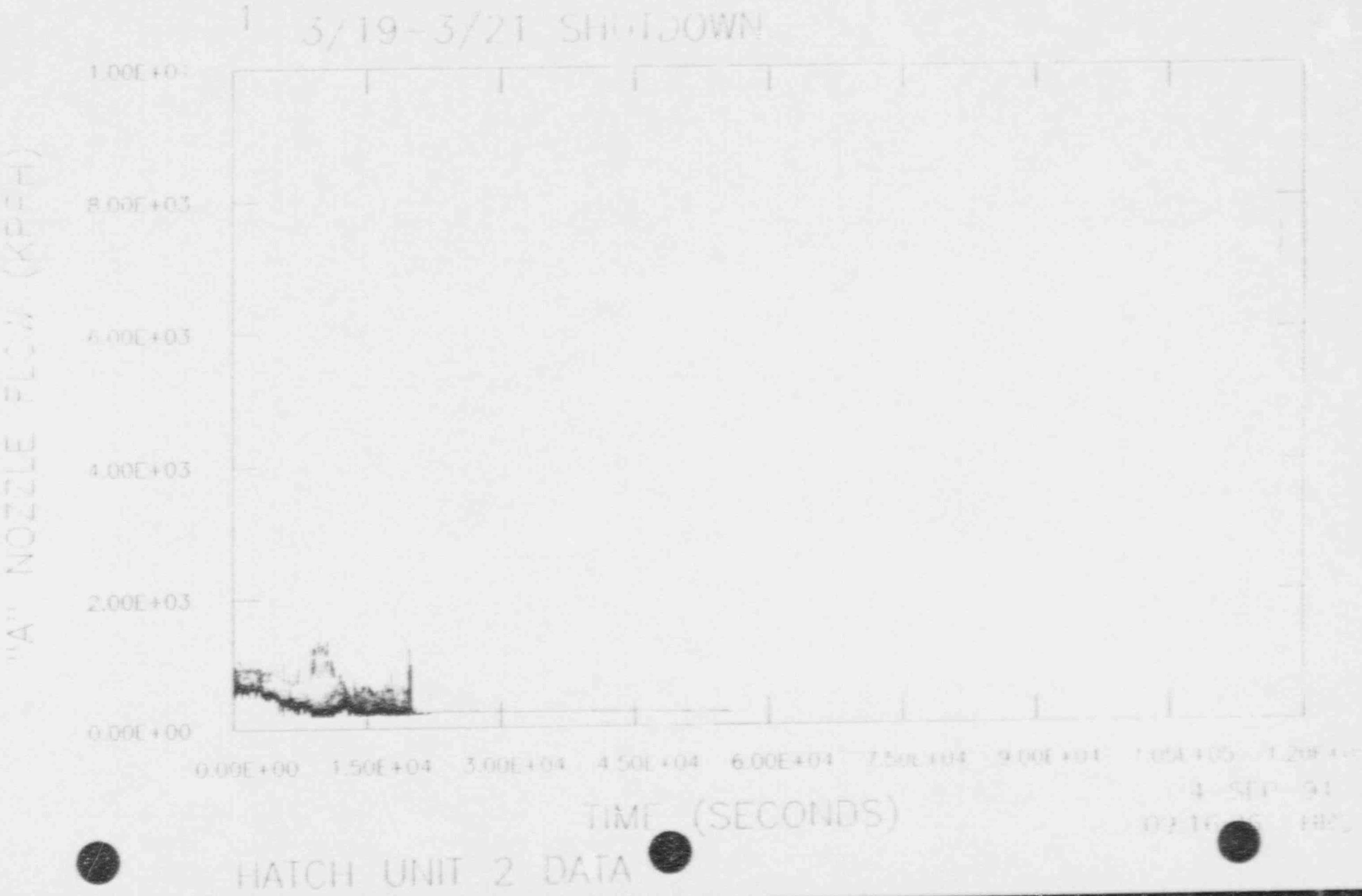


1
5/30-6/4 STARTUP
"B" NOZZLE FLUID TEMP. (F)
5.00E+02
4.00E+02
3.00E+02
2.00E+02
1.00E+02
0.00E+00
0.00E+00
1.50E+04
3.00E+04
4.50E+04
6.00E+04
7.50E+04
9.00E+04
TIME (SECONDS)
HATCH UNIT 2 DATA

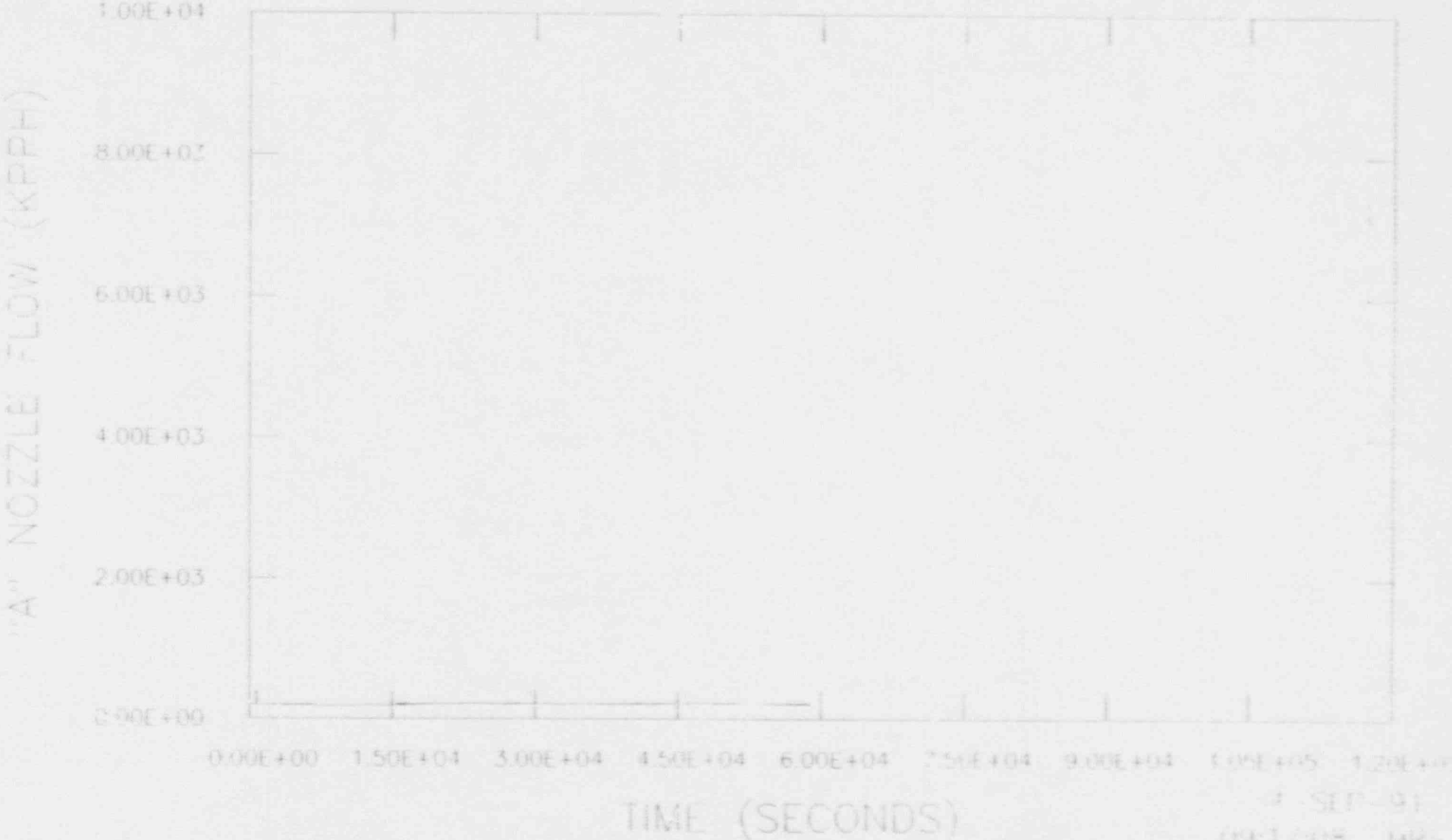


HATCH UNIT 2 DATA

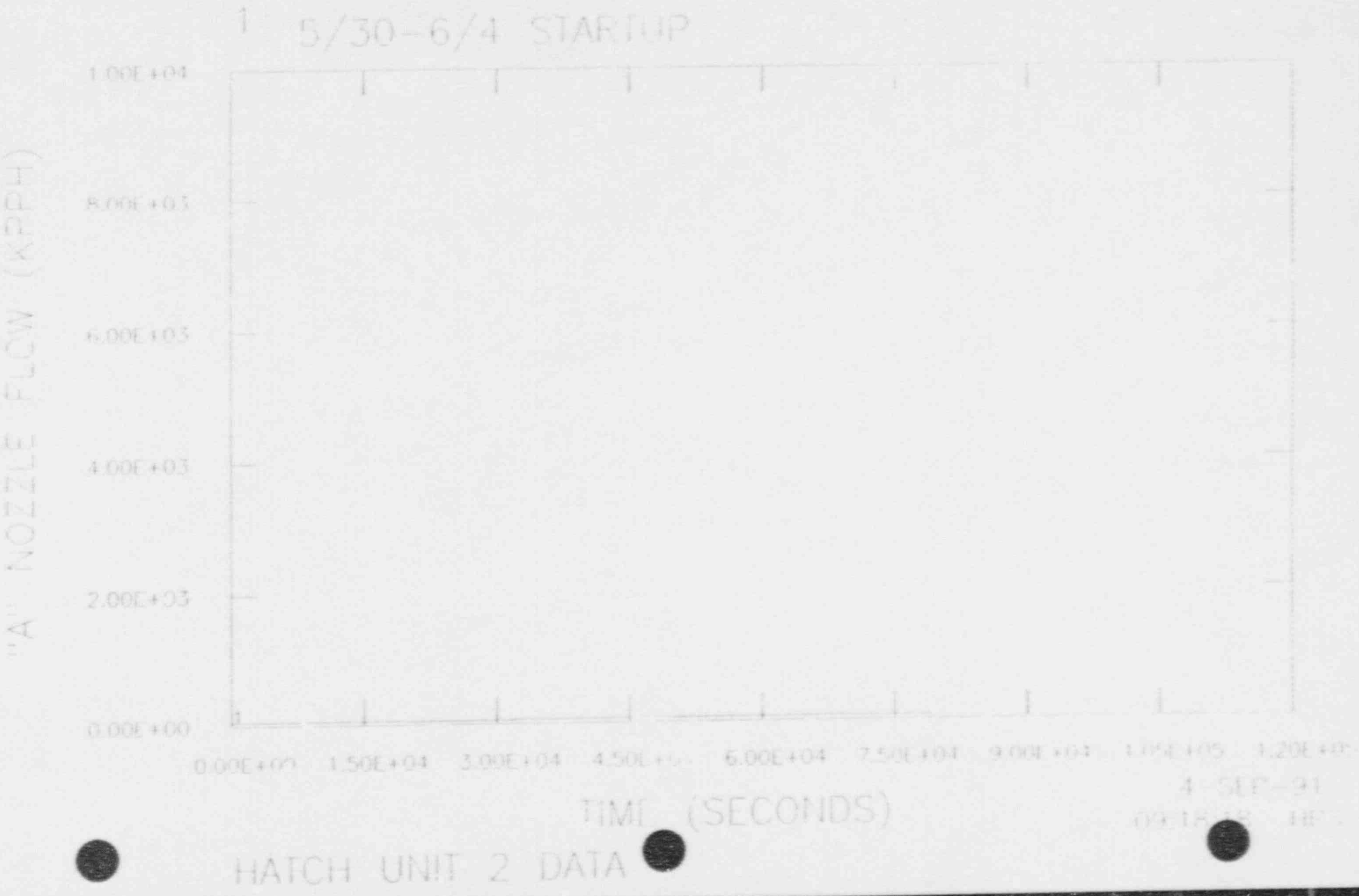
A.3 Loop A Mixed Fluid Flow Plots


1
3/19-3/21 SHUTDOWN
"A" NOZZLE FLOW (KPPH)
1.00E+04
8.00E+03
6.00E+03
4.00E+03
2.00E+03
0.00E+00
0.00E+00
1.50E+04
3.00E+04
4.50E+04
6.00E+04
7.50E+04
9.00E+04
1.05E+05
1.20E+05
TIME (SECONDS)
4 SEP 91
HATCH UNIT 2 DATA

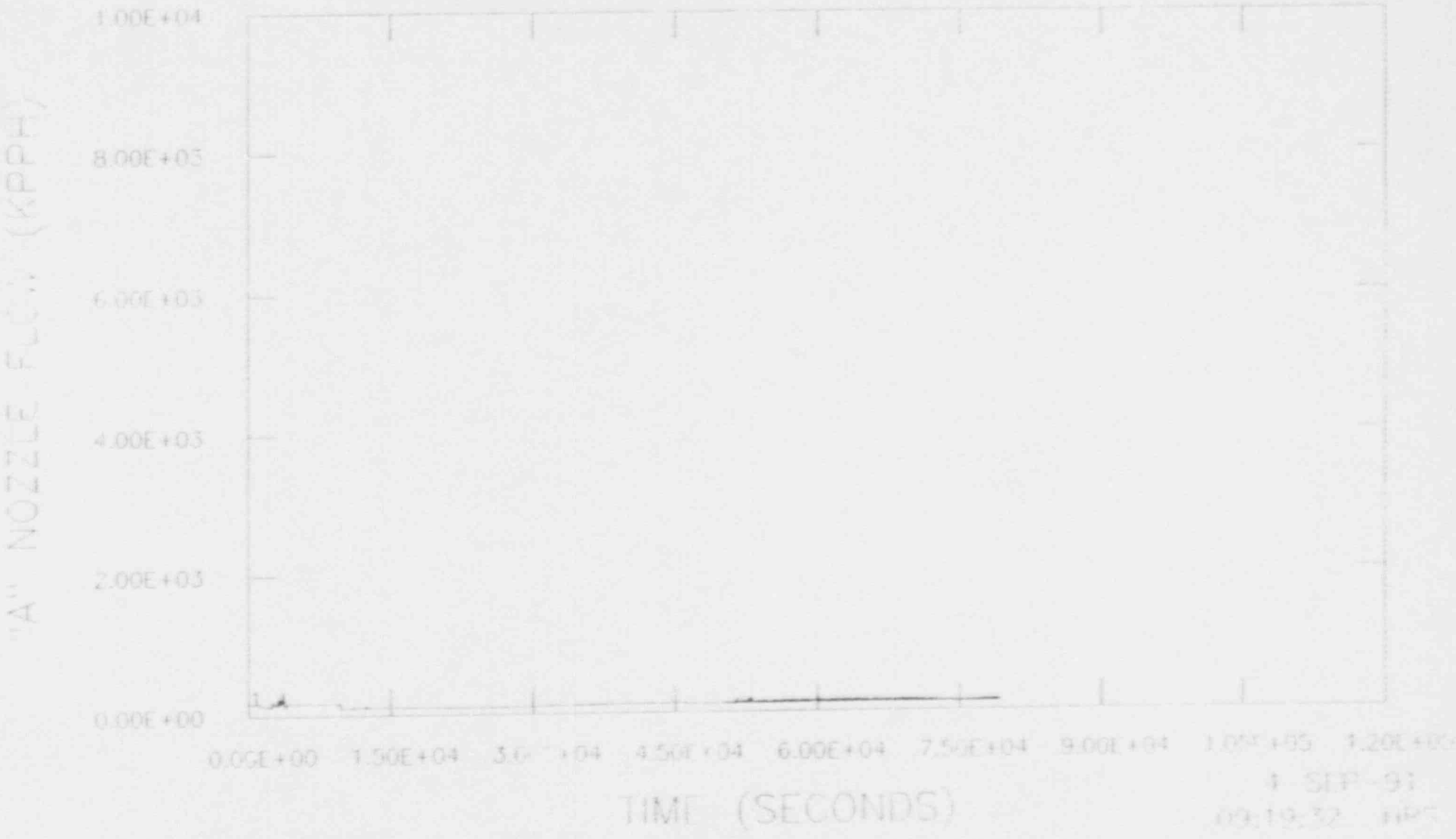
1 3/19–3/21 SHUTDOWN



HATCH UNIT 2 DATA

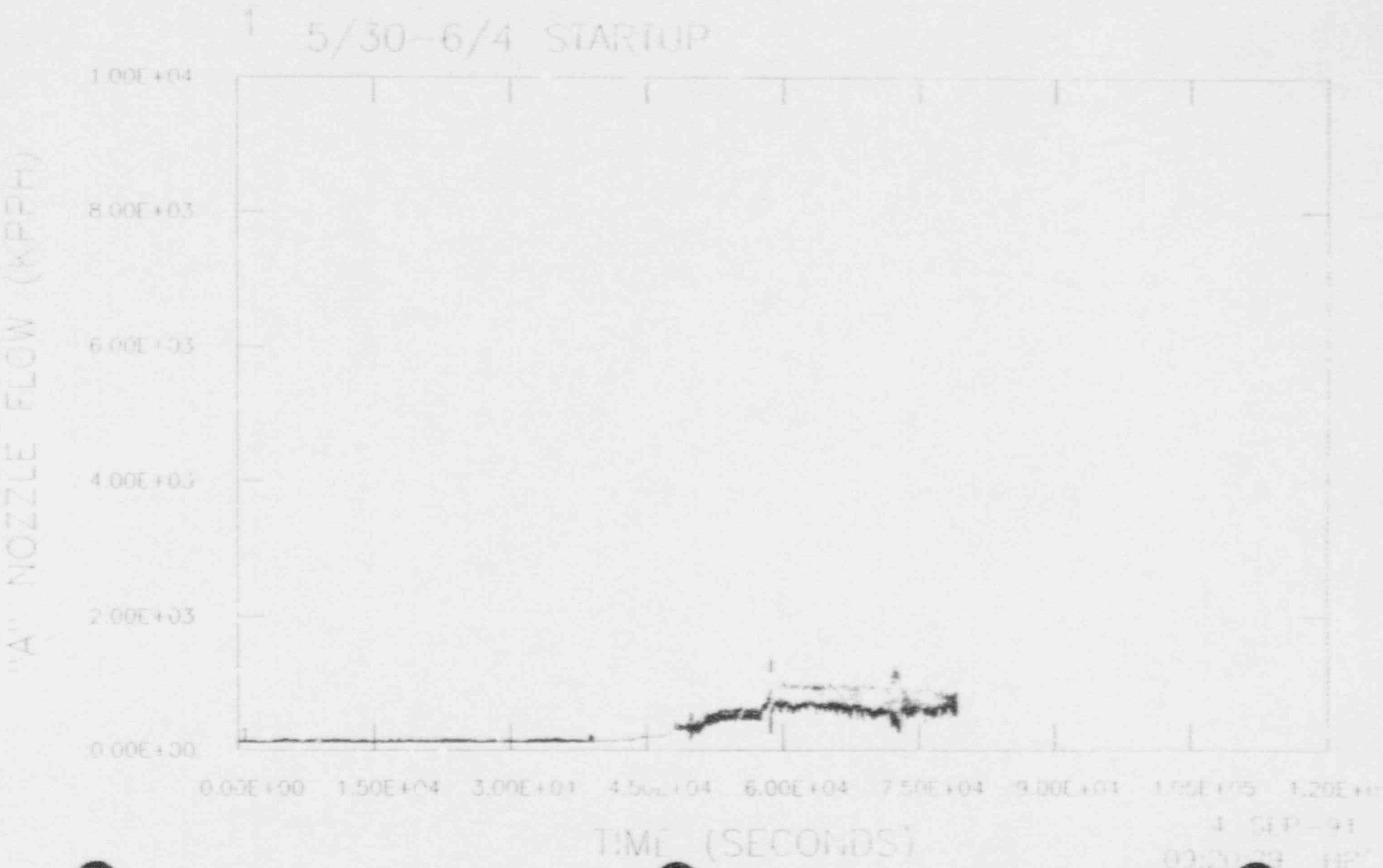

1 5/30-6/4 STARTUP
"A" NOZZLE FLOW (KPPH)
1.00E+04
8.00E+03
6.00E+03
4.00E+03
2.00E+03
0.00E+00
0.00E+00
1.50E+04
3.00E+04
6.00E+04
7.50E+04
1.20E+05
TIME (SECONDS)
4-SEP-91
HATCH UNIT 2 DATA

1 5/30-6/4 STARTUP

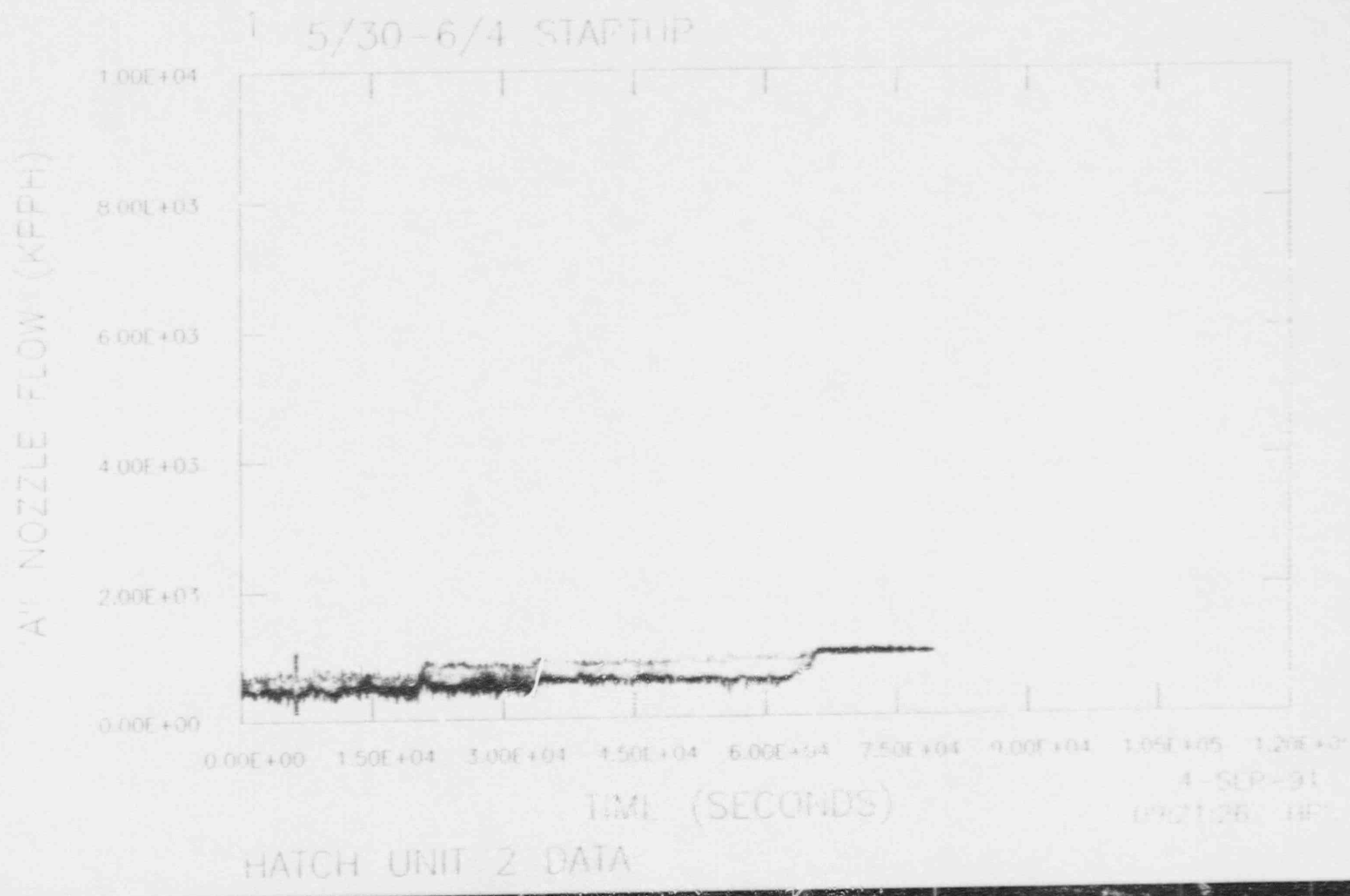


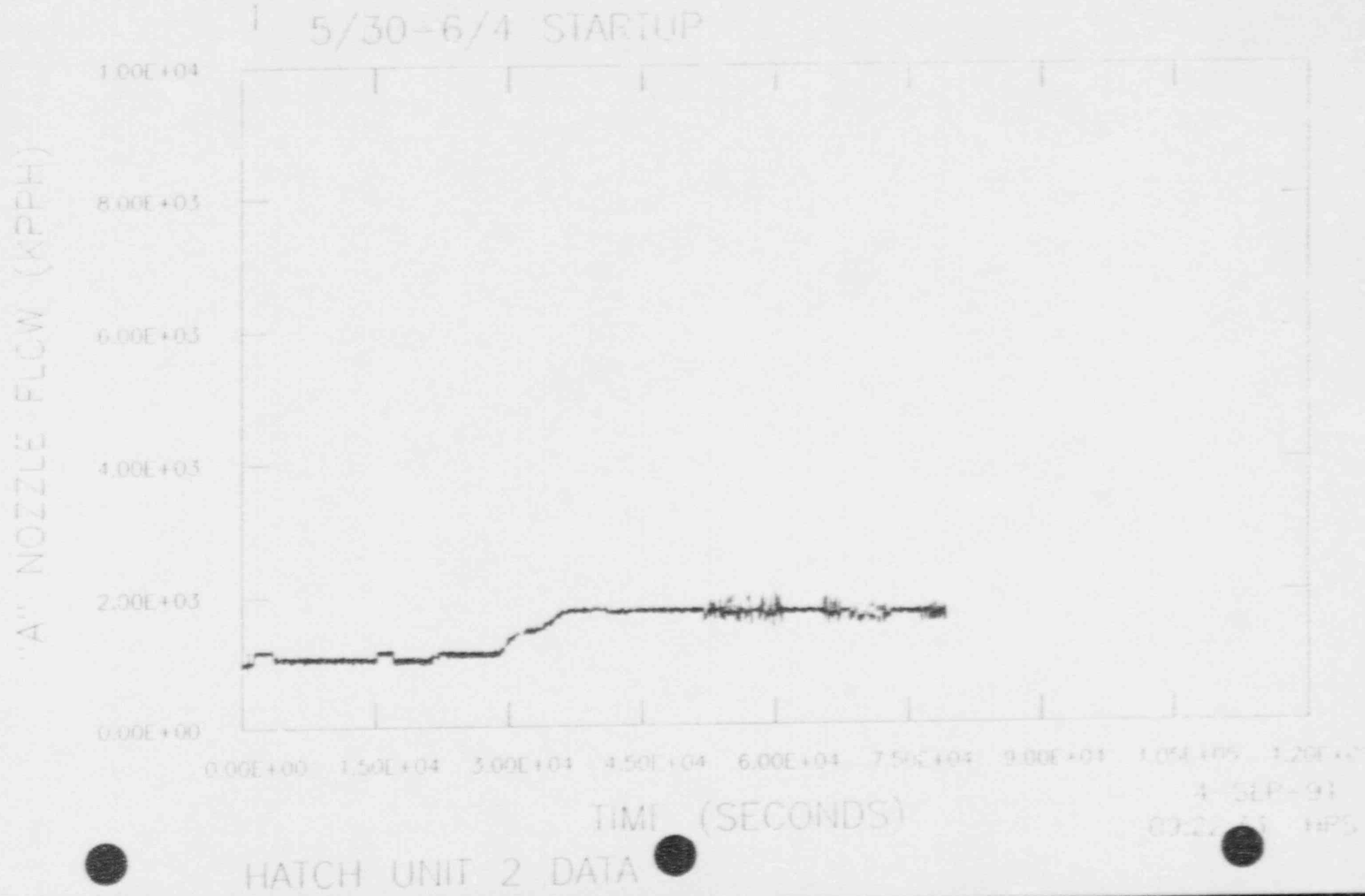
1-SEP-91

HATCH UNIT 2 DATA

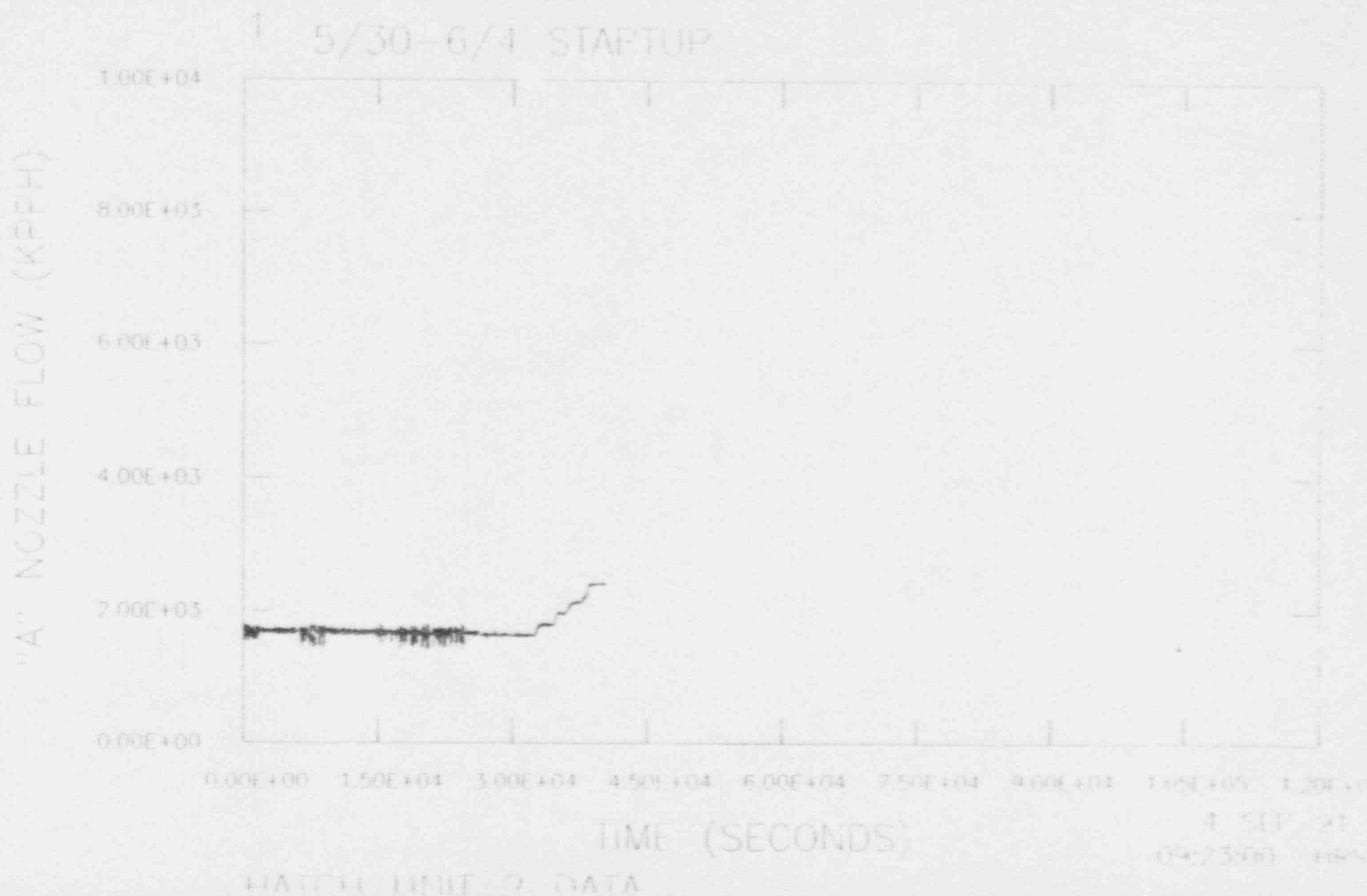


HATCH UNIT 2 DATA


1 5/30-6/4 STARTUP
A" NOZZLE FLOW (KPPH)
1.00E+04
8.00E+03
6.00E+03
4.00E+03
2.00E+03
0.00E+00
0.00E+00
1.50E+04
3.00E+04
4.50E+04
6.00E+04
7.50E+04
9.00E+04
1.05E+05
TIME (SECONDS)
4-SEP-91
HATCH UNIT 2 DATA

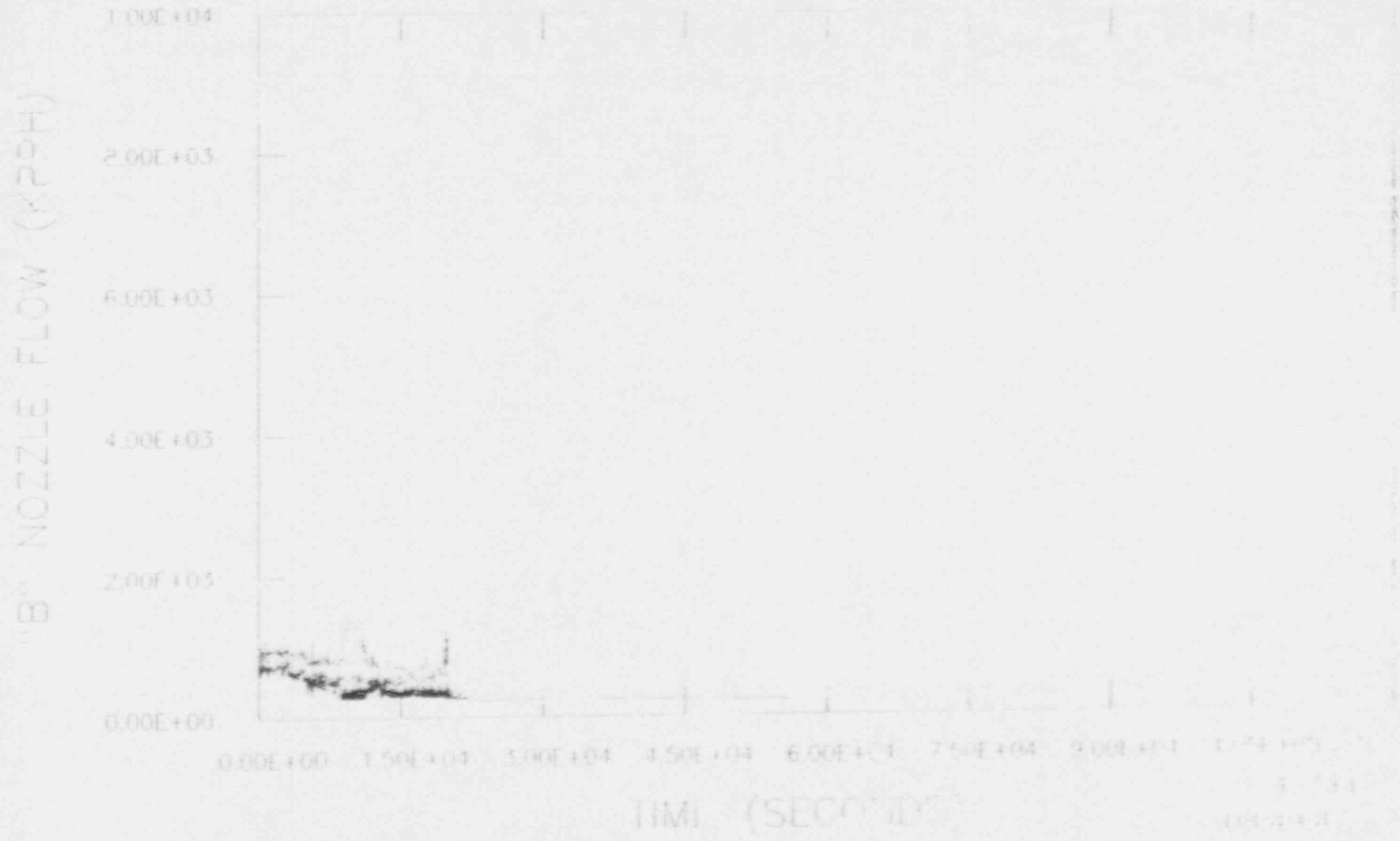


1
5/30-6/4 STARTUP
"A" NOZZLE FLOW (KPPH)
1.00E+04
8.00E+03
6.00E+03
4.00E+03
2.00E+03
0.00E+00
0.00E+00
1.50E+04
3.00E+04
4.50E+04
6.00E+04
7.50E+04
9.00E+04
1.05E+05
TIME (SECONDS)
4-SEP-91
HATCH UNIT 2 DATA

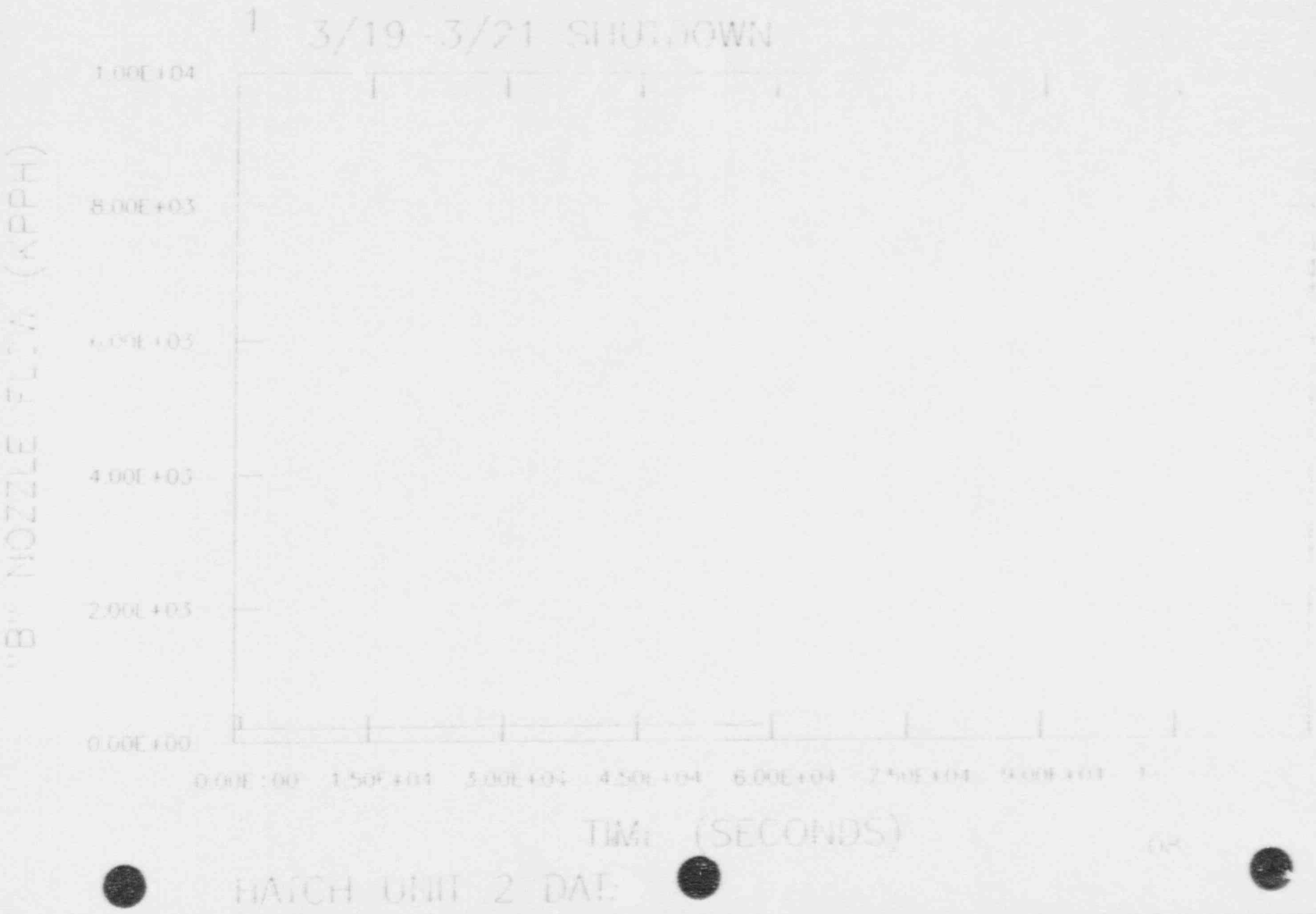

1
5/30-6/4 STARTUP
"A" NOZZLE FLOW (KPPH)
1.00E+04
8.00E+03
6.00E+03
4.00E+03
2.00E+03
0.00E+00
0.00E+00
1.50E+04
3.00E+04
4.50E+04
6.00E+04
7.50E+04
9.00E+04
1.05E+05
TIME (SECONDS)
HATCH UNIT 2 DATA

A.4 Loop B Mixed Fluid Flow Plots

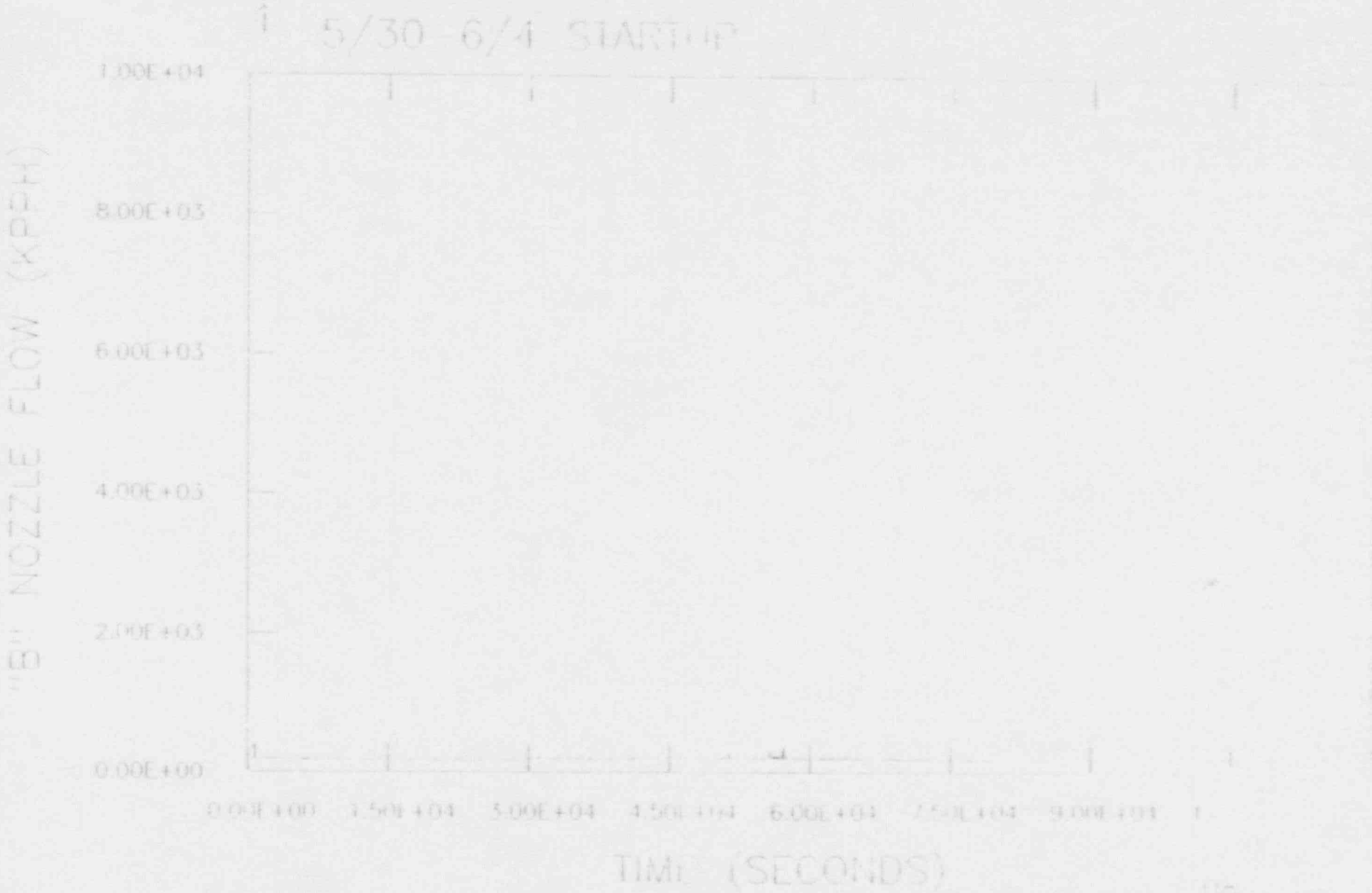
3/19 3/21 SHUTDOWN



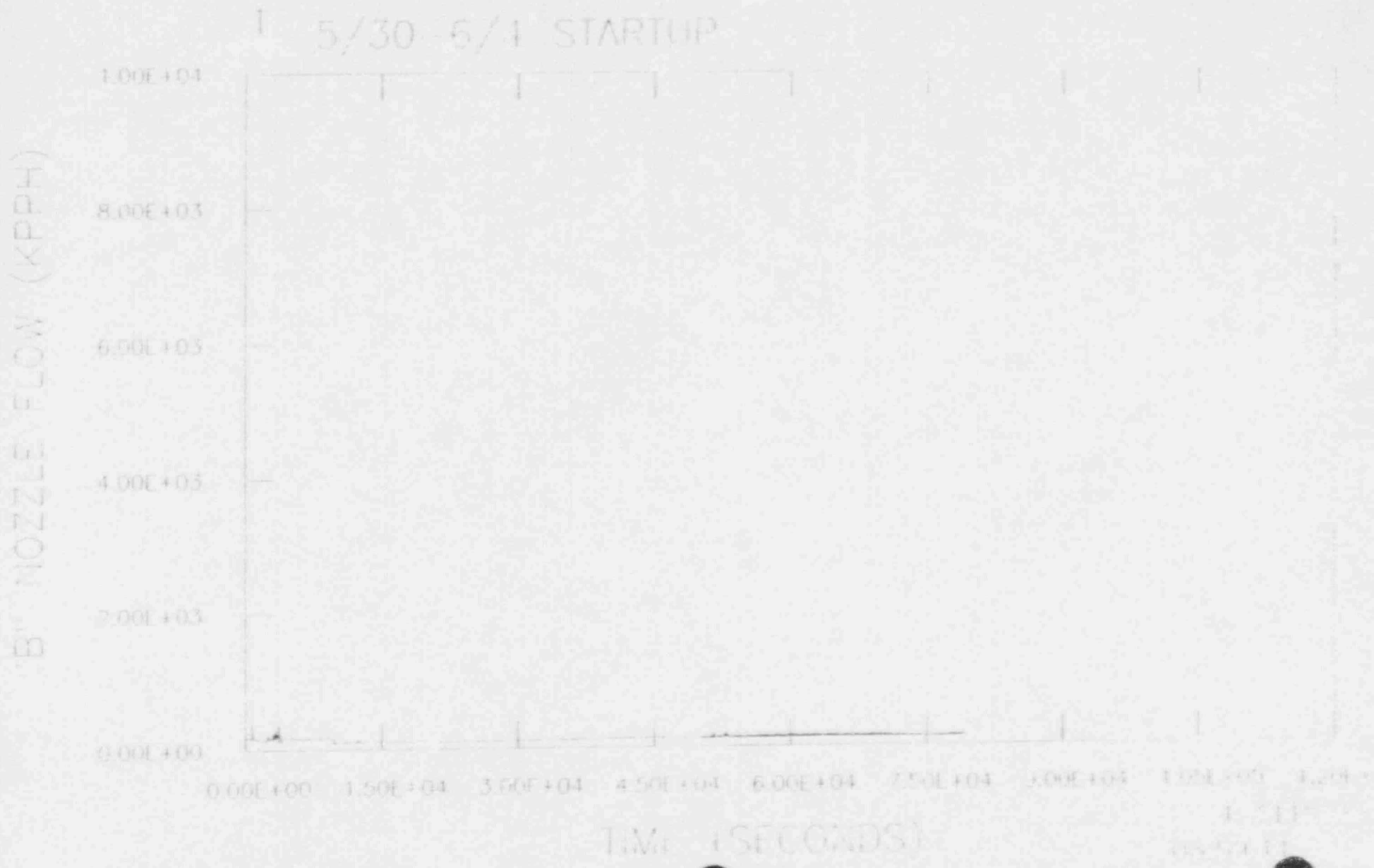
HATCH UNIT 2 DATA



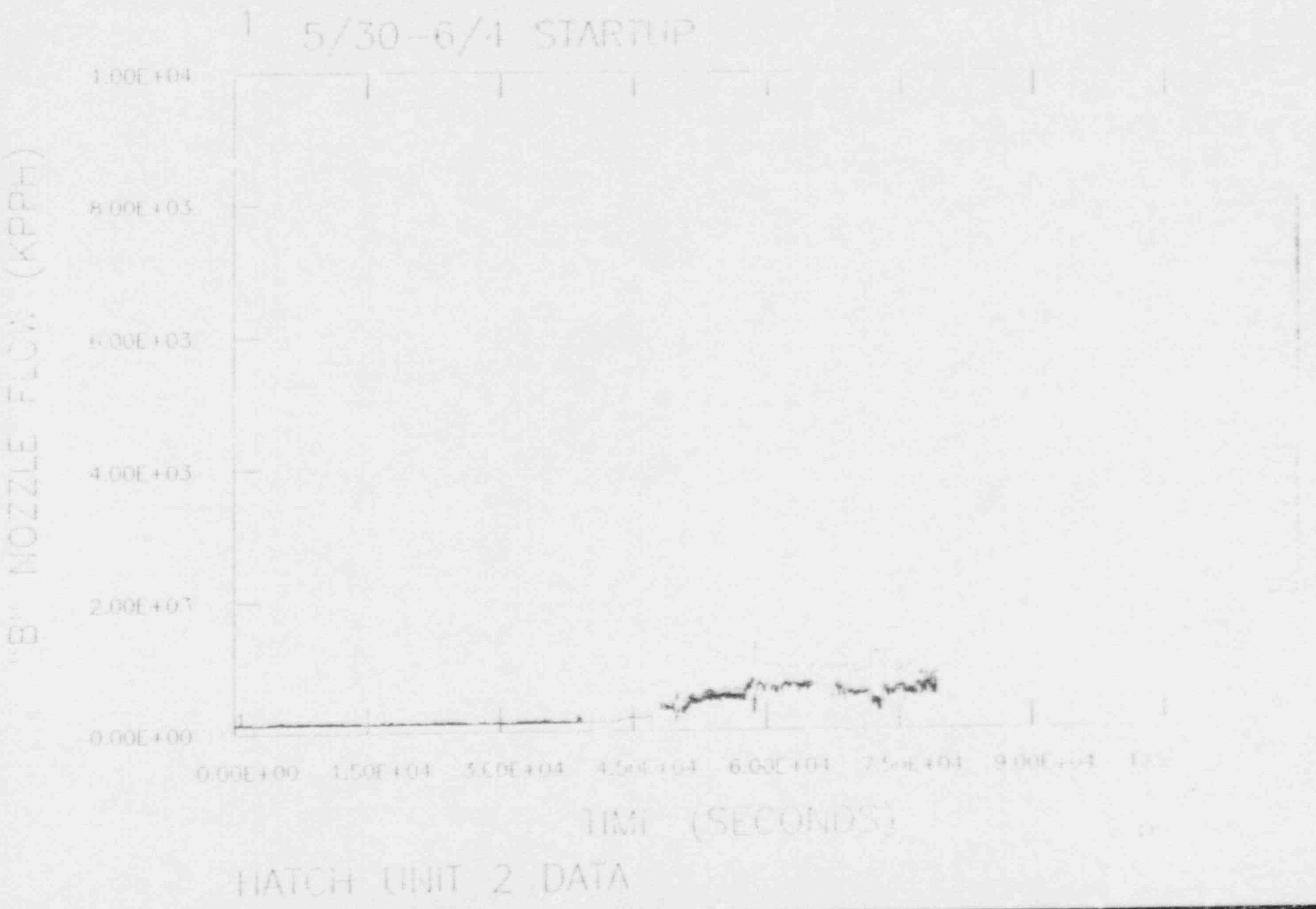
3/19 -3/21 SHUTDOWN
"B" NOZZLE FLOW (KPPH)
1.00E+04
8.00E+03
4.00E+03
2.00E+03
0.00E+00
0.00E+00
1.50E+04
3.00E+04
4.50E+04
6.00E+04
9.00E+04
TIME (SECONDS)
HATCH UNIT 2 DATA

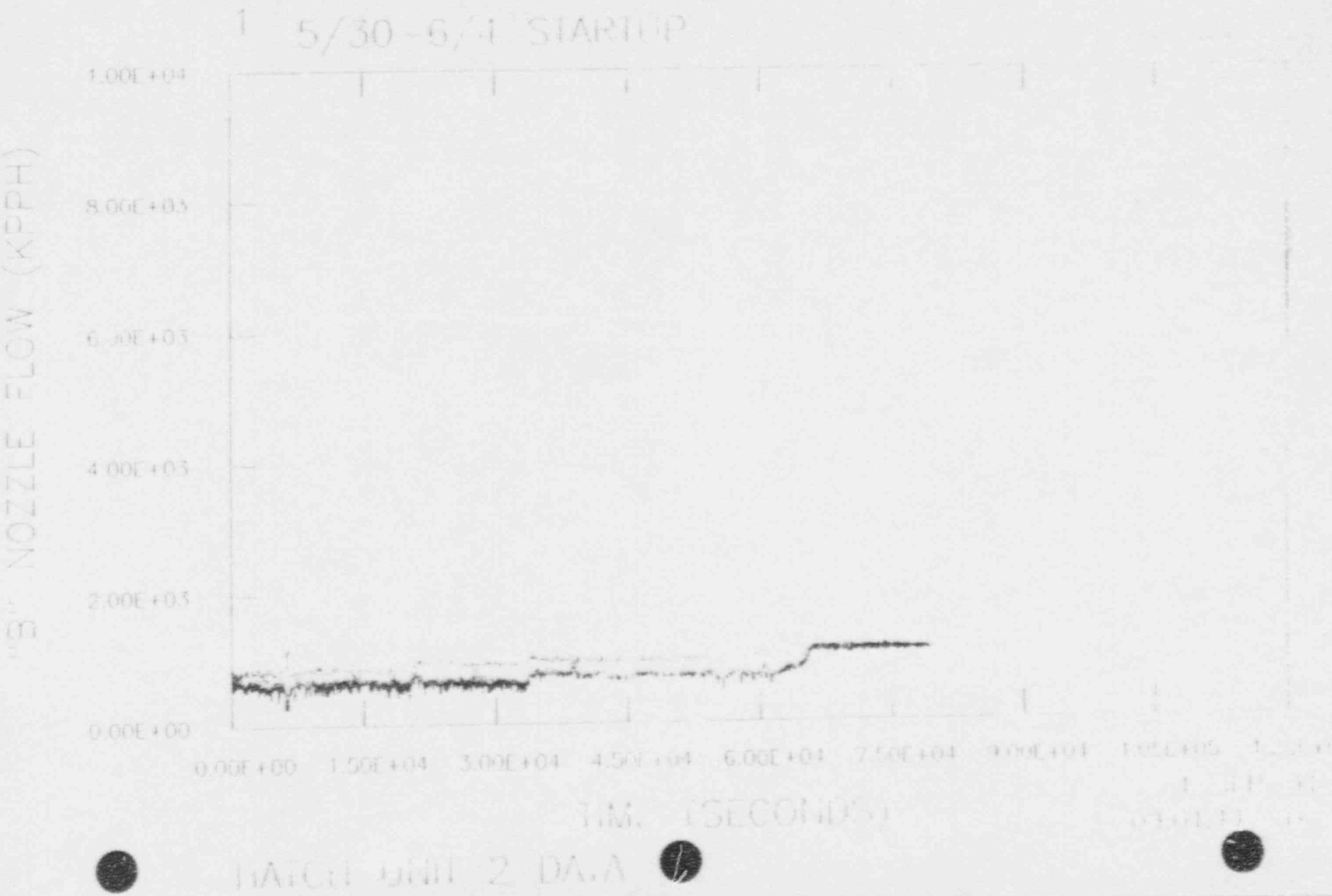


HATCH UNIT 2 DATA

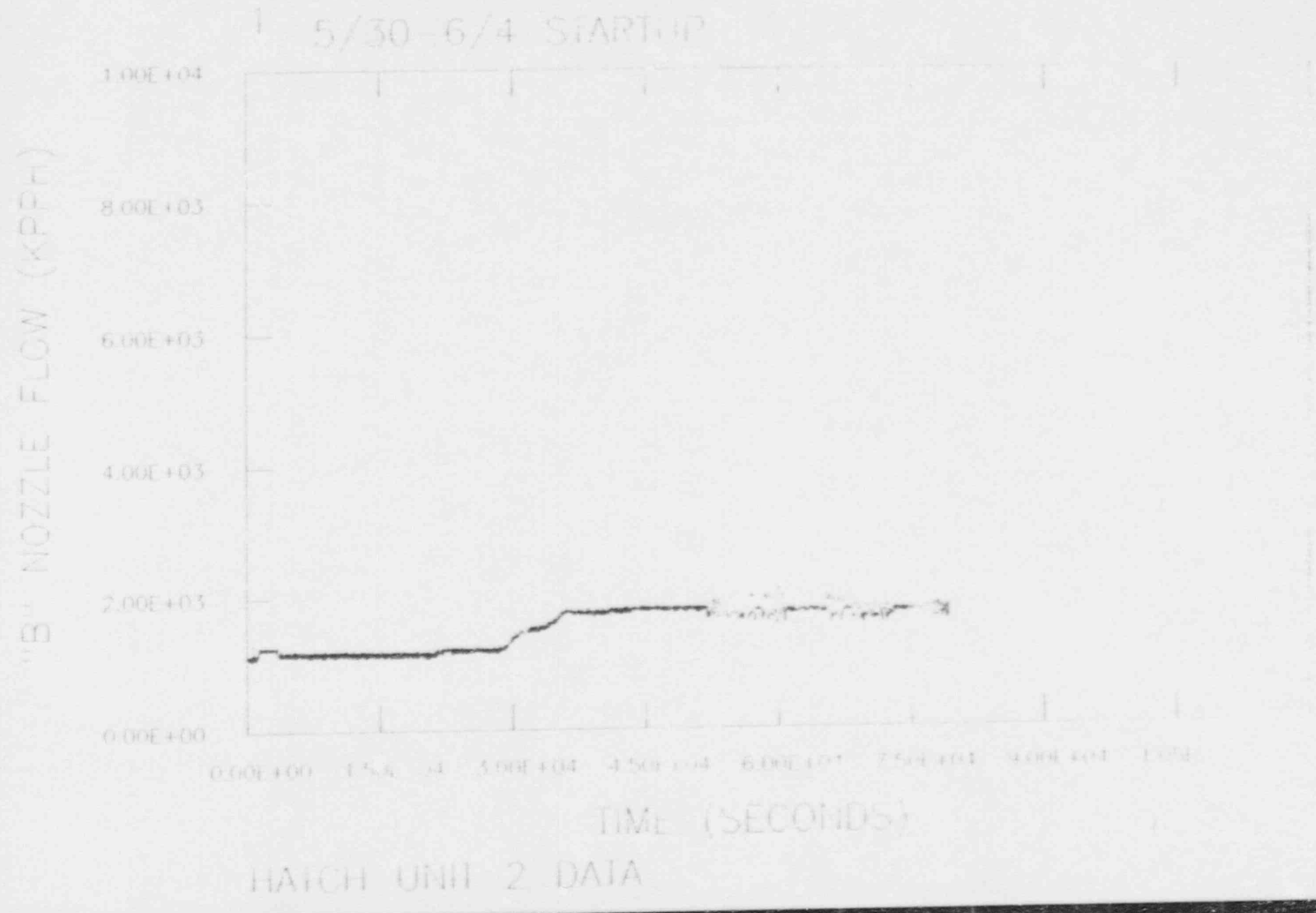


HATCH UNIT 2 DATA

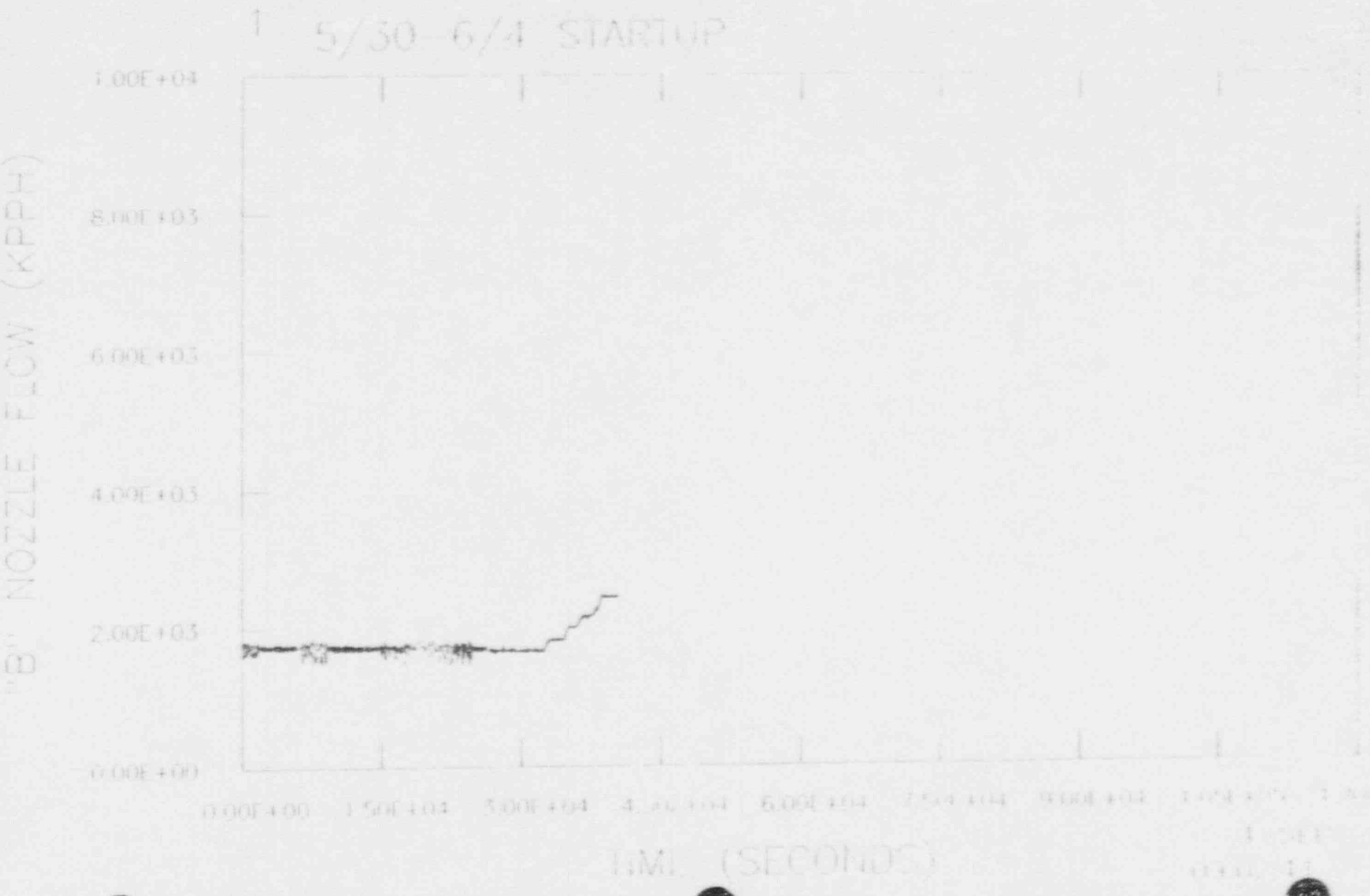

1
5/30–6/4 STARTUP
"B" NOZZLE FLOW (KPPH)
1.00E+04
8.00E+03
6.00E+03
4.00E+03
2.00E+03
0.00E+00
0.00E+00
1.50E+04
3.00E+04
4.50E+04
6.00E+04
7.50E+04
9.00E+04
TIME (SECONDS)
HATCH UNIT 2 DATA



1 5/30-6/4 STARTUP
"B" NOZZLE FLOW (KPPH)
1.00E+04
8.00E+03
6.00E+03
4.00E+03
2.00E+03
0.00E+00
0.00E+00
1.50E+04
3.00E+04
4.50E+04
6.00E+04
7.50E+04
9.00E+04
1.05E+05
TIM. (SECONDS)
HATCH UNIT 2 DATA

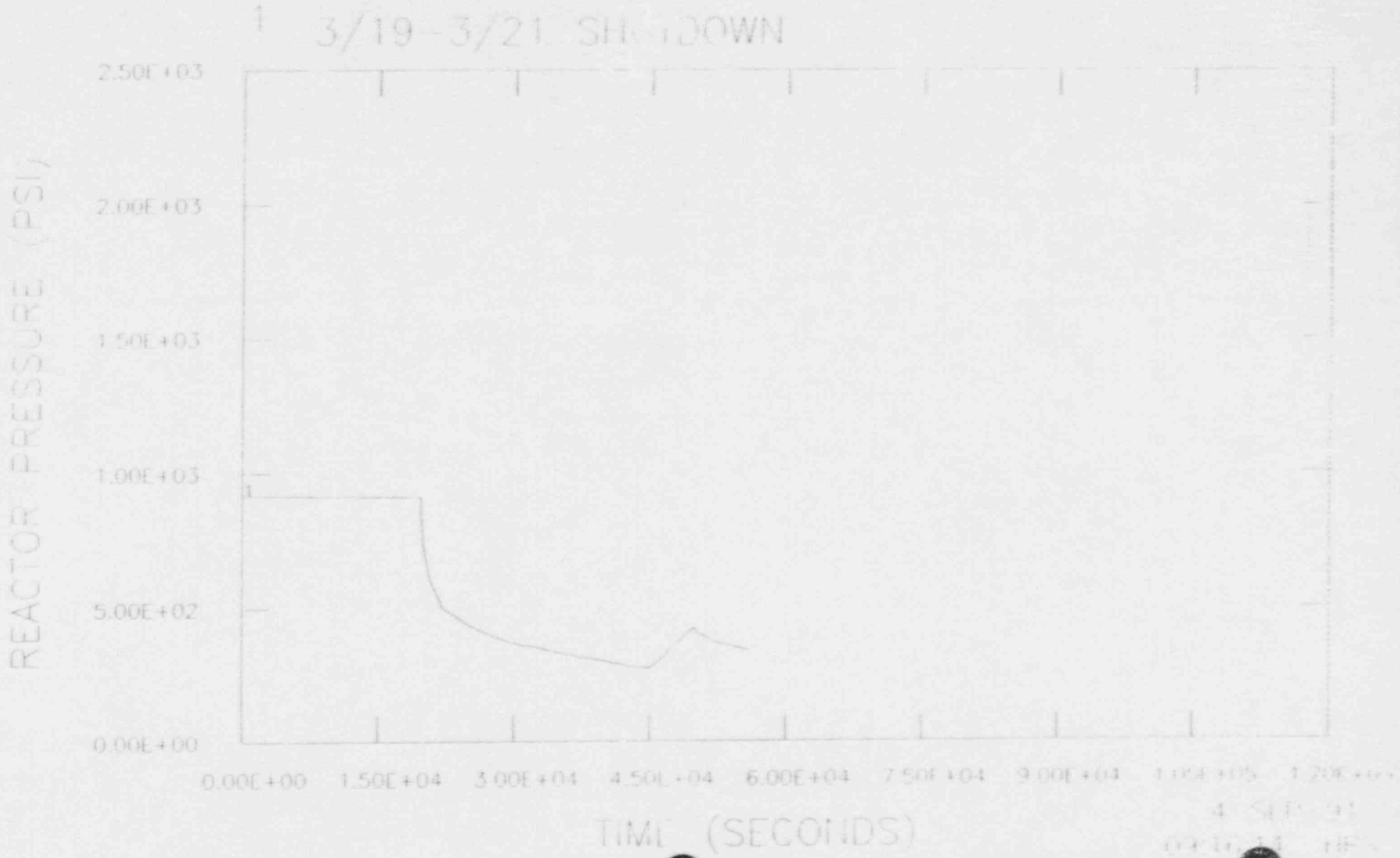


5/30-6/4 STARTUP
1.00E+04
8.00E+03
6.00E+03
4.00E+03
2.00E+03
0.00E+00
"B" NOZZLE FLOW (KPPH)
0.00E+00
3.00E+04
6.00E+04
TIME (SECONDS)
HATCH UNIT 2 DATA

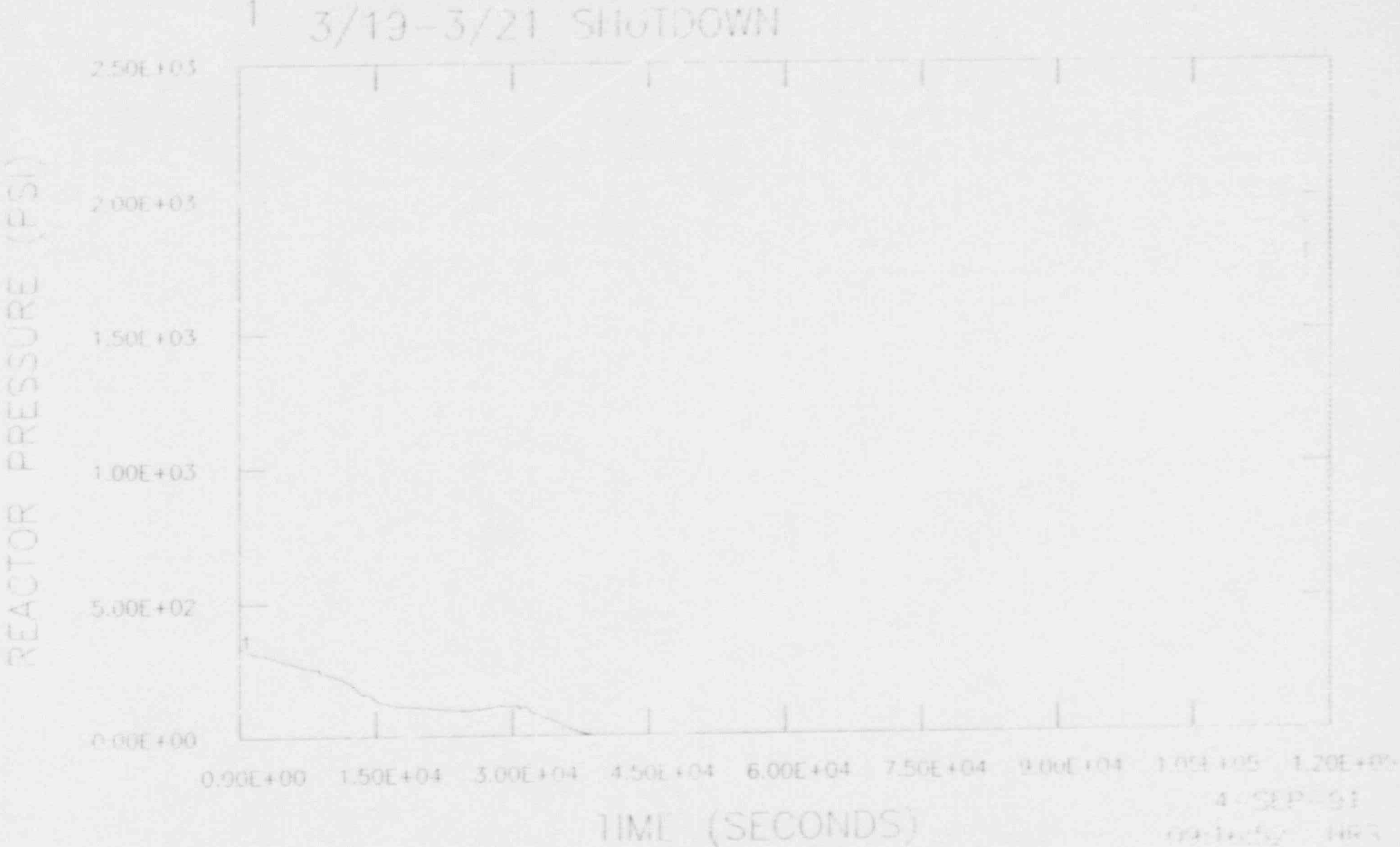


HATCH UNIT 2 DATA

A.5 Reactor Pressure Plots



HATCH UNIT 2 DATA



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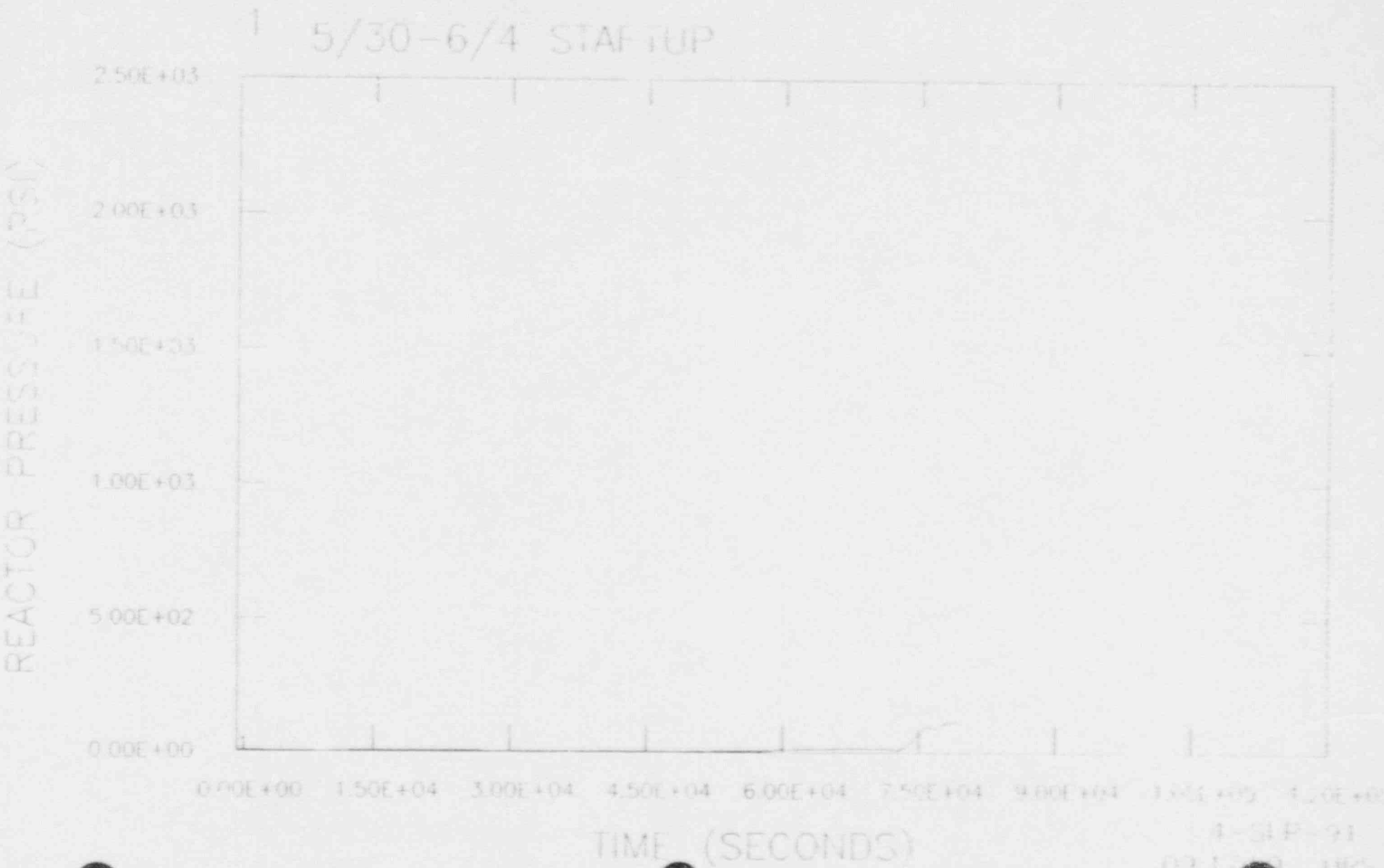

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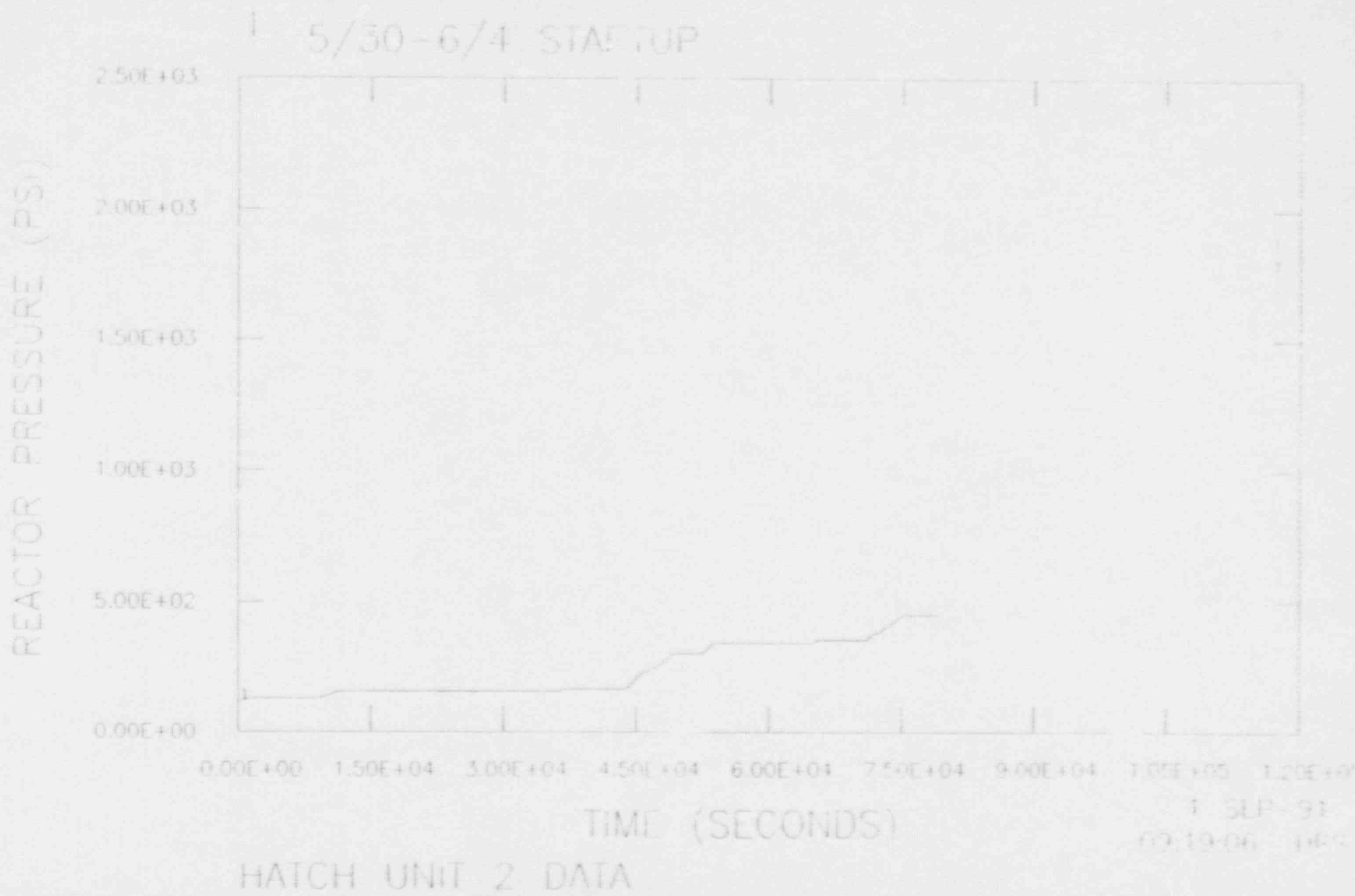

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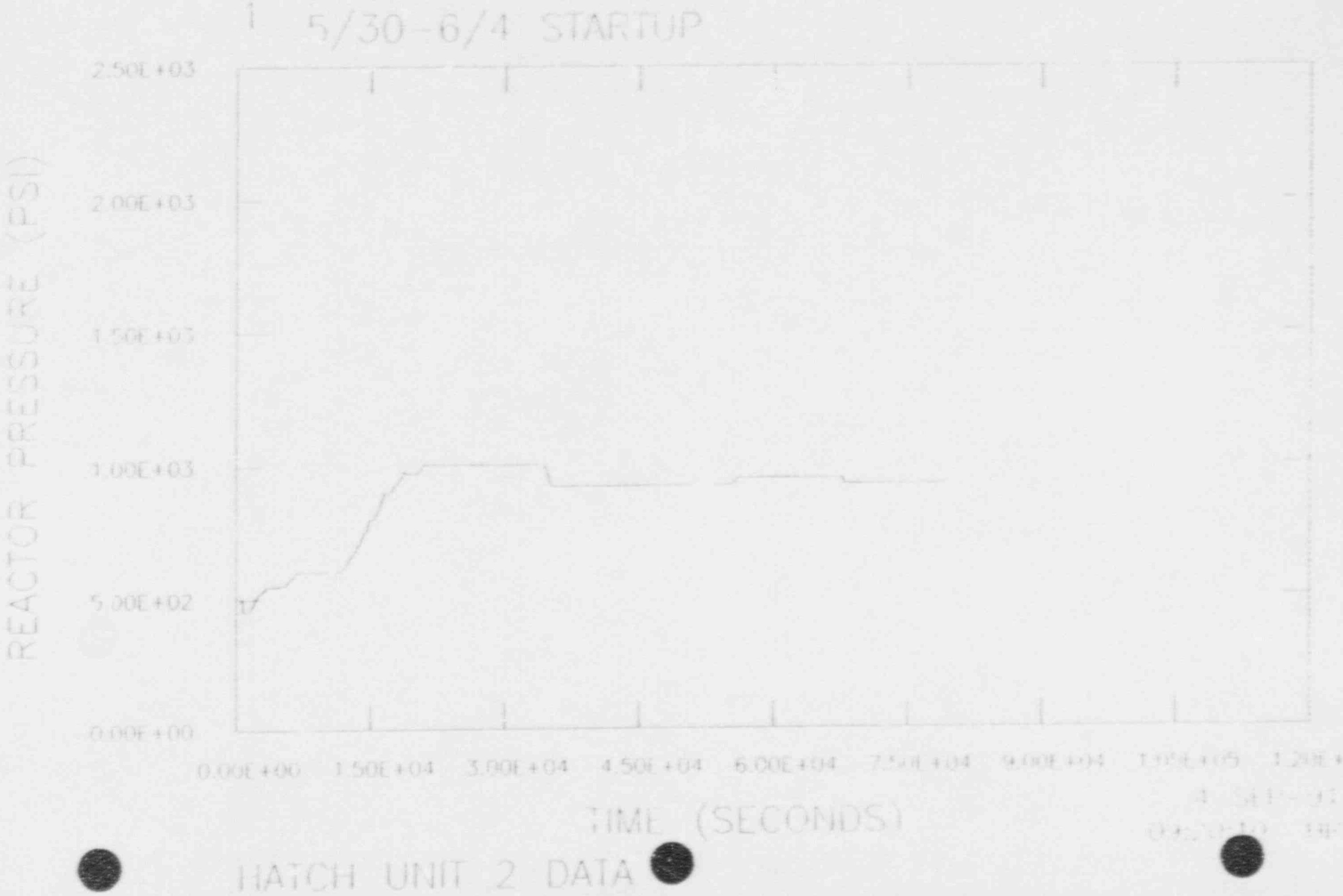
1.50E+03
1.00E+03
5.00E+02
0.00E+00
0.00E+00
1.50E+04
3.00E+04
4.50E+04
6.00E+04
7.50E+04
9.00E+04
1.05E+05
TIME (SECONDS)
HATCH UNIT 2 DATA



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REACTOR PRESSURE (PSI)
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0.00E+00
0.00E+00
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3.00E+04
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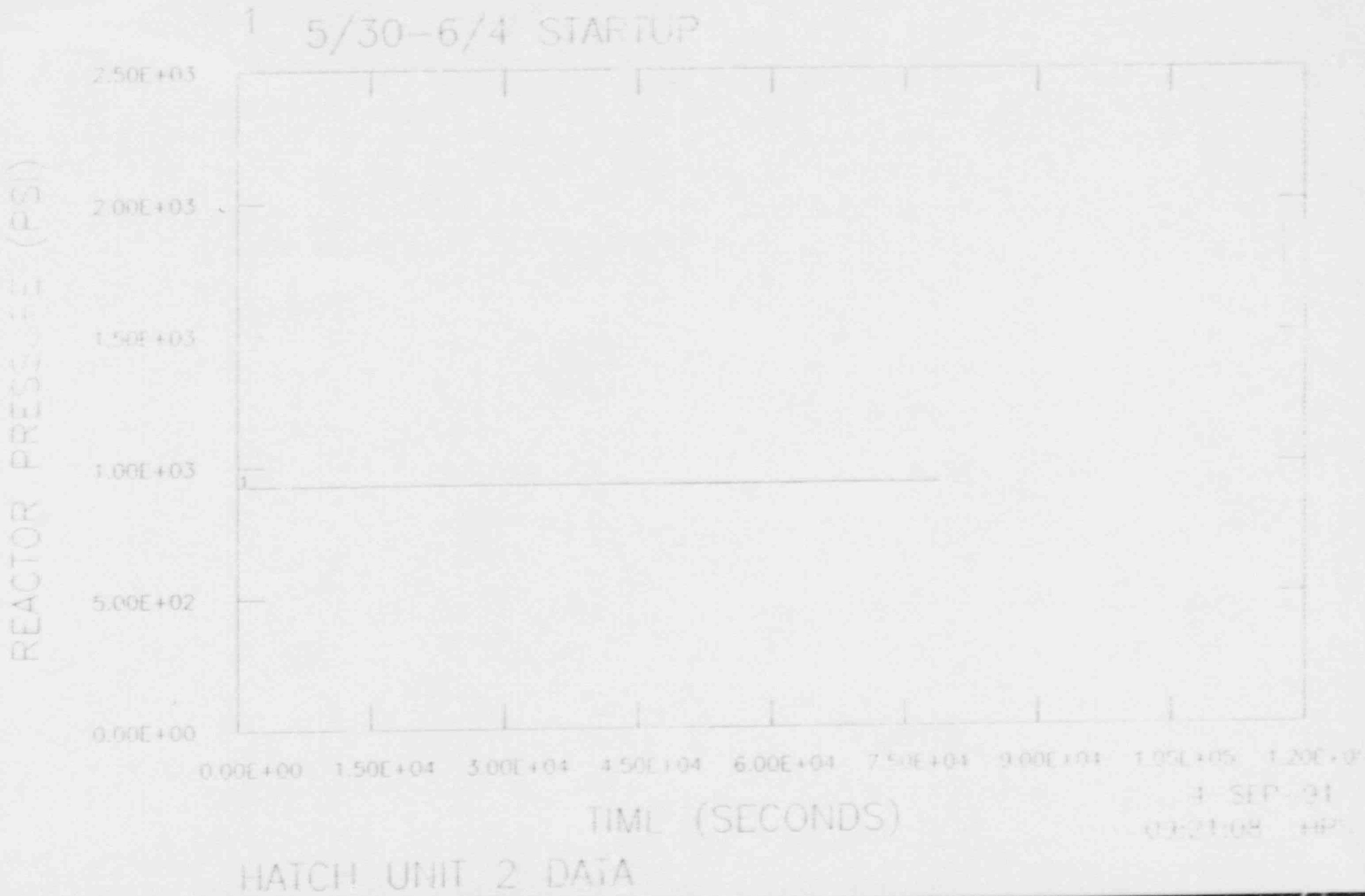
6.00E+04
9.00E+04
TIME (SECONDS)
HATCH UNIT 2 DATA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REACTOR PRESSURE (PSI)
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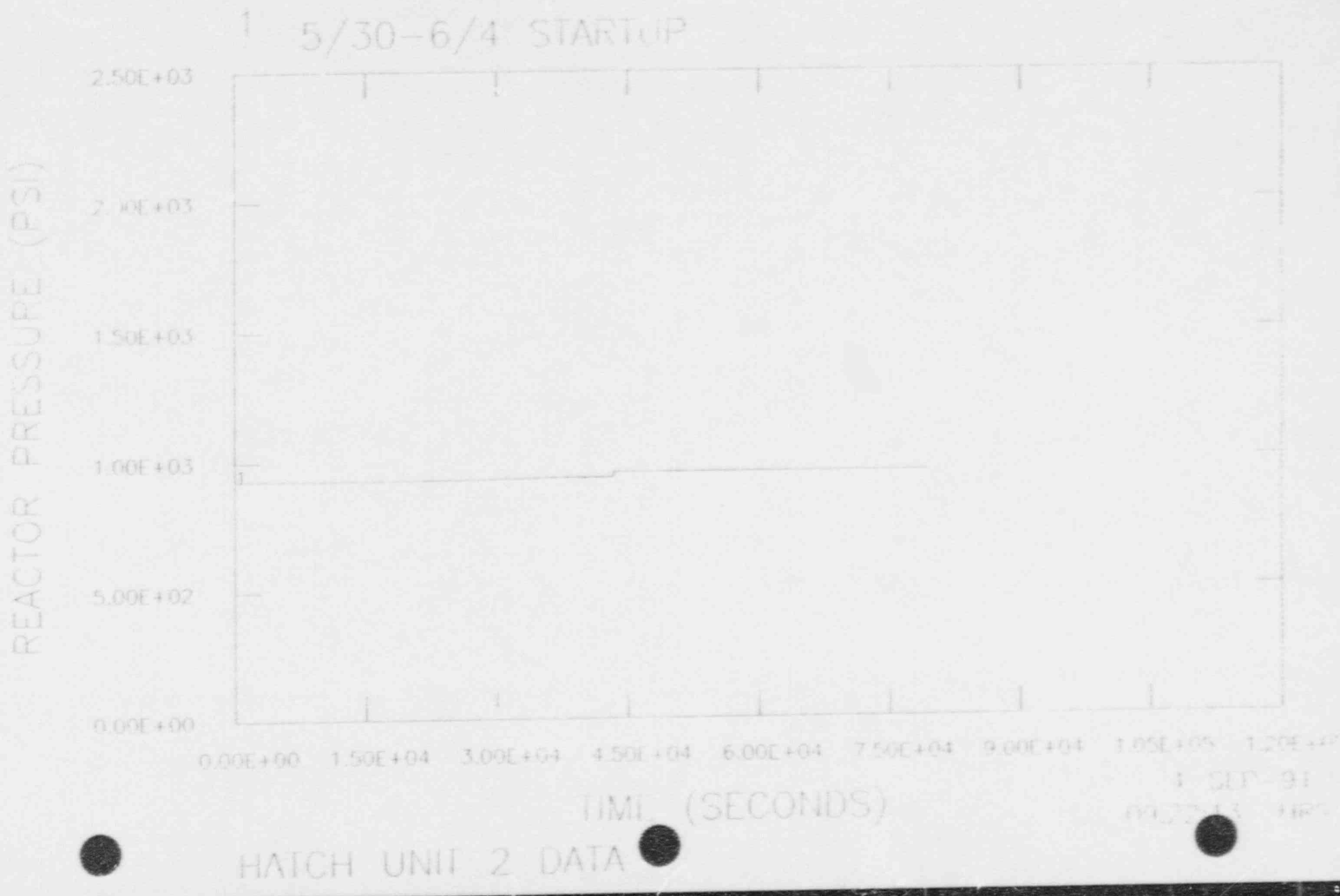
0.00E+00
0.00E+00
1.50E+04
3.00E+04
4.50E+04
6.00E+04
7.50E+04
9.00E+04
1.05E+05
TIME (SECONDS)
4 SEP 91
HATCH UNIT 2 DATA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5/30-6/4 STARTUP
REACTOR PRESSURE (PSI)
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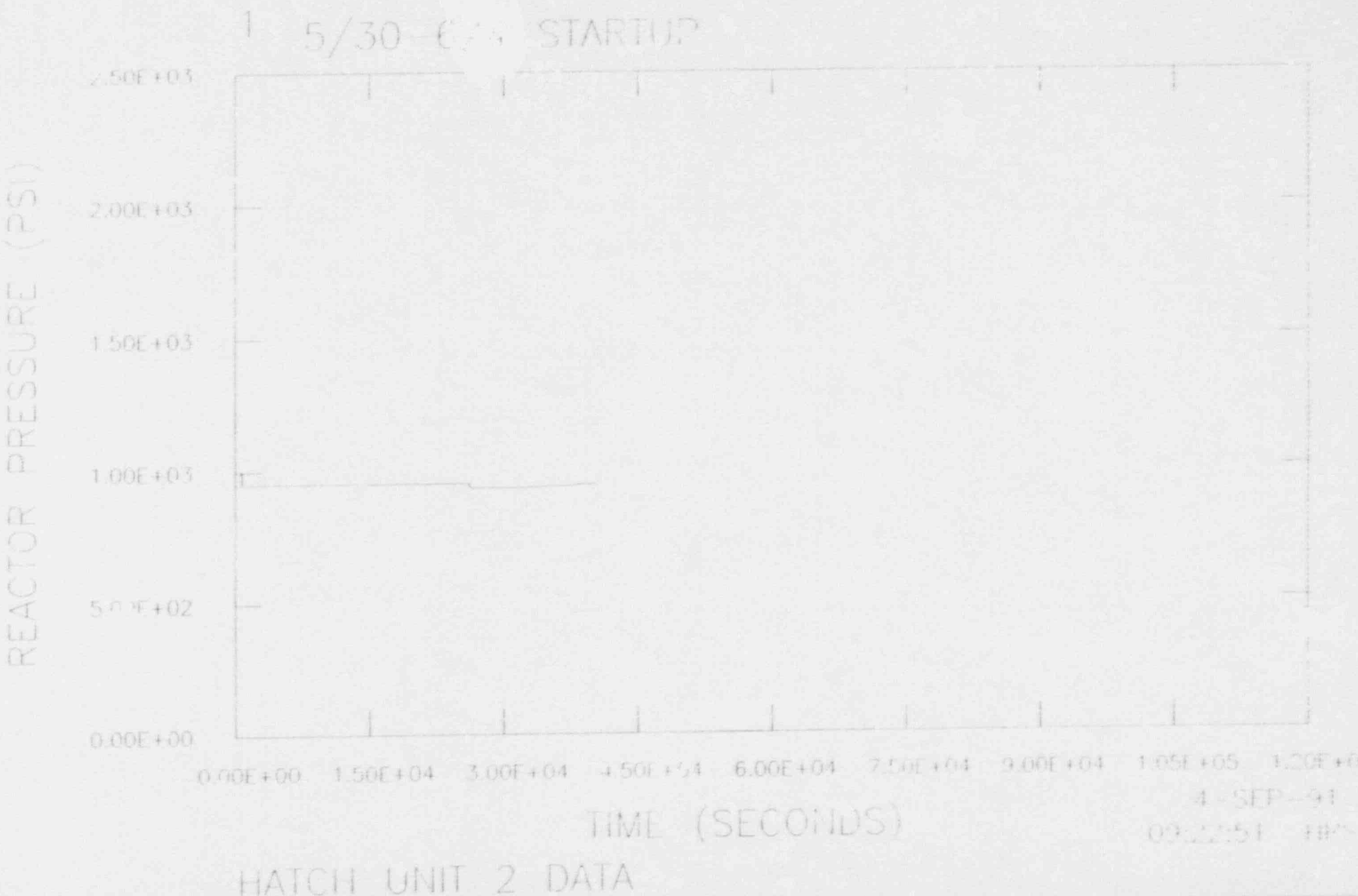
2.00E+03
1.50E+03
1.00E+03
5.00E+02
0.00E+00
0.00E+00
1.50E+04
3.00E+04
4.50E+04
6.00E+04
7.50E+04
9.00E+04
1.05E+05
TIME (SECONDS)
HATCH UNIT 2 DATA


1
5/30–6/ STARTUP
REACTOR PRESSURE (PSI)
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1.50E+03
1.00E+03
5.00E+02
0.00E+00
0.00E+00
1.50E+04
3.00E+04
4.50E+04
6.00E+04
7.50E+04
9.00E+04
1.05E+05
1.20E+06
TIME (SECONDS)
4-SEP-91
HATCH UNIT 2 DATA

APPENDIX B
COMPARISON OF STRESS INTENSITY FACTOR SOLUTIONS

The boundary integral equation/influence function magnification factors provided in Figure 1 were originally applied to the feedwater nozzle region several years ago when incidences of nozzle corner cracking were reported. Reference B-1 documents the comparison of experimental test results with those obtained from the use of the magnification factors shown in Figure 1. Those results showed good agreement between the two, and confirmed that continued use of the magnification factors was appropriate.

To further demonstrate the reasonableness of the Figure 1 magnification factors, stress intensity factor predictions were compared to those obtained from the detailed three dimensional (3-D) study reported by Gilman and Rashid (Reference B-2). Gilman and Rashid performed detailed 3-D finite element analysis to predict stress intensity factors for a nozzle geometry. These predictions were substantiated by comparison with available experimental data. Therefore, a comparison of the predictions made using the magnification factors given in Figure 1 to those of Gilman and Rashid is useful in demonstrating the validity of the predictions made in this report.

Figure B-1 shows the stress profile obtained by Gilman and Rashid from their 3-D finite element analysis. This profile was used along with the magnification factors of Figure 1 to predict stress intensity factors for comparison with the Gilman and Rashid results. This comparison is made in Figure B-2. It is seen that the agreement between the two predictions is very good, which provides further justification for the use of these factors.

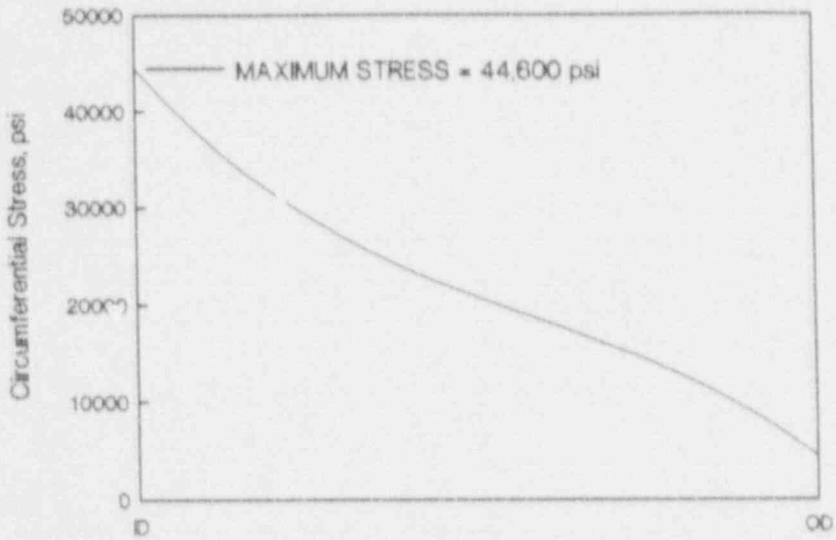

FIGURE B-1: Gilman & Rashid 3-D Stress Profile

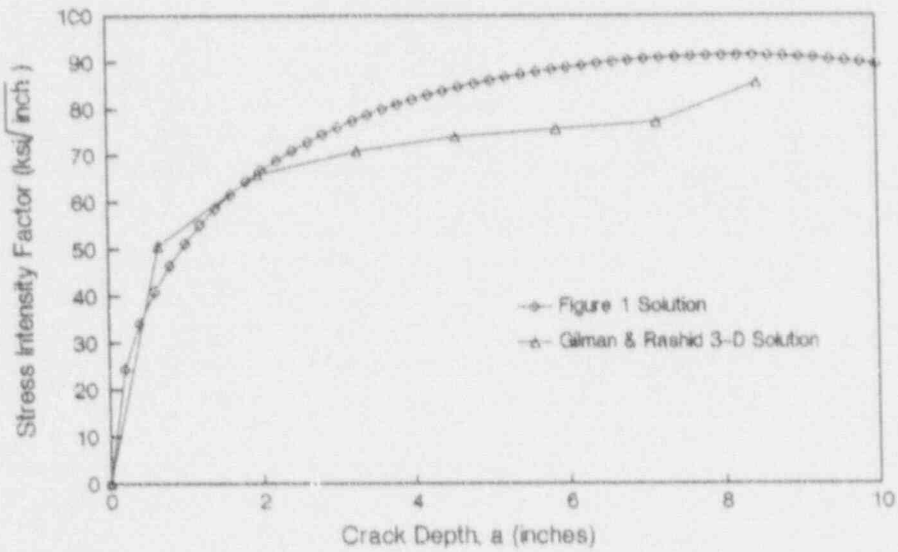

FIGURE B-2: Comparison of Stress Intensity Factor Solutions

REFERENCES:

[B-1] S.A. Delvin and P.C. Ricardella, "Fracture Mechanics Analysis of JAERI Model Pressure Vessel Test," 78-PVP-91, American Society of Mechanical Engineers, April 5, 1978.

[B-2] J.D. Gilman and Y.R. Rashid, "Three-dimensional Analysis of Reactor Pressure Vessel Nozzles," Transactions of the 1st International Conference for Structural Mechanics in Reactor Technology (SMIRT), Volume 4, Part G, September 1971.