GENERAL ATOMIC

GA-A14386
UC-77

FORT ST. VRAIN SURVEILLANCE AND TESTING PROGRAM

QUARTERLY PROGRESS REPORT FOR THE PERIOD ENDING MARCH 31, 1977

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PREVIOUS QUARTERLY REPORTS IN THIS SERIES

GA-A13209, for the period ending September 30, 1974

GA-A13282, for the period ending December 31, 1974

GA-A13454, for the period ending March 31, 1975

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GA-A13692, for the period ending September 30, 1975

GA-A13793, for the period ending December 31, 1975

GA-A13909, for the period ending March 31, 1976

GA-A14042, for the period ending June 30, 1976

GA-A14154, for the period ending September 30, 1976

GA-A14246, for the period ending December 31, 1976

ABSTRACT

This publication continues the quarterly report series on Fort St. Vrain (FSV) Surveillance and Testing. The program will perform post-startup tests on FSV plant components and systems to increase our knowledge of operating characteristics of large HTGRs. This report contains a summary of the findings made during an extended power run at 28% thermal power and 73 MW(e).

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FIGURE

TABLES

1. INTRODUCTION

The Fort St. Vrain (FSV) Surveillance and Testing Program is directed toward acquiring FSV operating experience for application to the design of large HTGRs. Four subtasks remain funded for FY-77, none of which were scheduled for completion by the end of the quarter. The plant rise to power, which was started on July 1, 1976, continues. During this quarter the plant operated for one month at approximately 28% thermal power and 73 MW(e), and was then shut down for maintenance of miscellaneous equipment.

2. ACCOMPLISHMENTS

2.1. SUBTASK A: STEAM GENERATOR PERFORMANCE AND CORROSION SURVEILLANCE

2.1.1. Instrumentation

The steam generator temperature scanner system continued to perform in a satisfactory manner as reported in the December 1976 quarterly report.*

During the February/March shutdown, the exit thermocouple probes on the steam tubes just above the ringheaders were replaced by welded thermocouples which are fastened directly to the tube wall. As reported in the December 1976 quarterly report, these welded thermocouples should eliminate the inaccuracies associated with the probes.

2.1.2. Thermal Performance

As reported in the December 1976 quarterly report, a comparison of predicted and measured performance (heat duty) at or above 27% power indicated a deviation from predicted heat duty in the main steam bundle ranging from +0.2% to −2.5%. Investigation has shown that the deviation was steam temperature dependent and that it was primarily the result of inadequate modeling of regenerative heating within the interspace region at low power levels. The interspace region is the area enclosed by the steam generator penetration within which the main steam, reheat steam and feedwater lines pass through the bottom head of the PCRV. It was necessary to make a correction in the calculation of predicted performance to account for the

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*"Fort St. Vrain Surveillance and Testing Program Quarterly Progress Report for the Period Ending December 31, 1976," ERDA Report GA-A14246, General Atomic Company, January 1977.

larger than expected regenerative heating effects between the main steam and the feedwater and cold reheat steam. Less significant changes were also made in the modeling of the heat losses in the steam downcomer tubes and heat gain in the bundle inlet leads. An analysis of a data point at 28% power after these changes had been made showed a deviation between predicted and measured performance of less than 0.5%. The temperature dependence of these performance anomalies precluded their discovery when main steam temperatures were at or below 800°F. The larger than expected losses have prevented the steam generator from making expected main steam temperature at low power. Also, the excessive losses have forced the SH1 steam temperatures to be higher than expected and thus the co-flow SH2 is thermally pinched (i.e., steam exit temperature nearly equal to helium temperature). This results in a non-effective SH2, at the low loads thus far encountered.

The model changes discussed above will have a small, but not insignificant, effect on predicted performance at 100% power. It is felt at this time that as the plant approaches full-power operation the actual (i.e., measured) performance of the steam generator will more nearly conform to design performance.

2.1.3. Feedwater Chemistry

Water samples were taken from the economizer inlet. The period over which new data are provided herein covers the period from December 21, 1976, to February 22, 1977. Data through December 20, 1976, were discussed in the December 1976 quarterly report. Plots of the feedwater data are included in Appendix A.

Except for minor perturbations, the reactor was above 25% power through the end of January. During this time period the water chemistry was generally within specified limits. The reactor was shut down after January, but the feedwater was monitored continuously and the limits shown on the plots were maintained throughout the system cleanup, except for short periods when problems arose and were corrected.

2.2. SUBTASK D: PCRV STRUCTURAL RESPONSE VERIFICATION

2.2.1. PCRV Sensor Data Collection and Reduction

During the quarter, PCRV sensor data between 27% and 29% reactor power levels and at reactor shutdown were collected. The reductions of sensor data were performed using computer programs.

The structural response of the PCRV was evaluated by comparing the tendon load changes and concrete strains with the anticipated values which were established based on the analysis of the initial proof pressure test results. The latest data were acceptable.

Concrete temperatures of less than 120°F were indicated by the embedded thermocouples, which is consistent with expected results at this power level. A comparison of the top head concrete temperatures with those taken during earlier startup tests at corresponding power levels showed average temperature drops of 5°F to 10°F, reflecting the lowering of inlet cooling water temperature. It is now predicted the concrete temperatures in the top head will be within the 150°F design limit at full power.

As part of the data updating effort, sensor data collected since PCRV construction via the data acquisition system have been collated on one magnetic tape in a manner that permits the use of computer plotting routines. Work was initiated to prepare computer plots of data from vibrating wire strain gages, Carlson strain gages, tendon load cells and thermocouples collected up to the end of 1976.

2.2.2. Time-Dependent Structural Analysis

Time-dependent structural response of the FSV PCRV had initially been estimated on the basis of a projected loading history. To properly account for the observed vessel behavior, the creep analysis was redone using the actual FSV loading history. The finite element model used in the creep analysis is similar to that reported in Appendix E of the FSV FSAR. Stress contours and time/response history plots from the analytical results were prepared for evaluation and validation with the sensor data that are being collected as part of this subtask. This evaluation is an ongoing part of the program.

2.3. SUBTASK G: TRANSIENT ANALYSIS PROGRAM (TAP) CODE VERIFICATION

2.3.1. Relevant Plant Status Items

During January the plant was operated at nearly a constant 28% thermal power. Minor adjustments in helium flow, feedwater flow and core power were made from time to time and several parameter changes were imposed during a gain test which was performed January 18 and 19. On January 20 through January 22, the main controllers were partially tuned and placed in automatic for a short time. A loss of auxiliary boiler steam on January 25 caused a loss of steam-driven boiler feedpumps, a shutdown of the "C" circulator and a temporary reduction of power to below 5%.

2.3.2. Data Acquisition System for Model Verification

During this period most of the desired channels from the Data Acquisition System for Model Verification (DASMV) were reading acceptable values. Minor errors were noted on certain parameters which are being evaluated. In each case, alternate instrumentation is available to provide necessary data to continue the model verification program.

An investigation into the accuracy of the temperature measurements from the DASMV was conducted. When the data were compared to instrument data sheet values, minor errors were noted in many of the temperatures. This error was attributed to the use of a linear fit (instead of the non-linear thermocouple curve) for the temperature conversion on the DASMV. The system was modified to use the thermocouple curve. The approximate errors at the 28% power level that will be corrected as a result of this change are as follows:

	Parameter	Correction with Non-Linear Curve (°F)
TT1174	Circulator 1C Helium Temperature	- 3.5
TT1178	Circulator 1B Helium Temperature	- 3.5
TT2205	Loop 1 Feedwater Temperature	-17
TT2206	Loop 2 Feedwater Temperature	-17
TT5220	HP Turbine Inlet Temperature	-13
TT5207	Loop 1 Main Steam Desuperheater Temp	- 3
TT5208	Loop 2 Main Steam Desuperheater Temp	- 3
TT2227-6	SG B-1-6 Trim Control Temperature	- 0.6
TT2228-6	SG B-2-6 Trim Control Temperature	- 0.6
TT5216	Circulator Inlet Steam Temperature	- 4

2.3.3. Performance Comparisons

During this period no planned transients were imposed on the plant, but several steady-state performance comparisons were made. Initial steady-state comparisons at about 28% thermal power indicated a difference in steam generator performance. This difference was later attributed to convective and radiative regenerative heat losses in the steam generator. Equations were added in TAP to represent these heat losses and the plant performance was repredicted and compared to measured data. Table 1 shows the final comparison. (This effect on steam generator performance was discussed above for Subtask A.)

TABLE 1

28.5% POWER MEASURED VS PREDICTED PERFORMANCE

	Measured Data 1557 Hr Jan 18	TAP, Match of Measured Data
Core Power, %	28.5	28.6
Helium Flow, %	39.3	38.7
Circulator Speed, rpm	3860	3700
Feedwater Flow, lb/hr	561,600	561,600
Main Steam, Temperature, °F	925	934
Reheat Steam Temperature, °F	971	963
Feedwater Temperature, °F	283	282
Cold Reheat Temperature, °F	543	539
Core Inlet Temperature, °F	642	651
Steam Generator Inlet Temperature, °F	1115	1118
Reactor Pressure, psia	648	656
Attemperator Flow (Add to Feedwater), lb/hr	0	0

To provide a better comparison of plant vs TAP performance the effects of parameter changes on overall plant performance were determined. The major parameters of core power, circulator speed, and feedwater flow were held constant with the operating control system in manual mode and parameters such as feedwater temperature, cold reheat attemperation, PCRV pressure, etc. were changed individually. Table 2 presents the resulting gain (influence coefficient) table with the changes from the base case shown in parentheses. Performance gains were also obtained for changes in the major parameters. Data from the plant gain study which was performed on January 18 and 19 will be used for comparison with TAP predicted gains.

TABLE 2

FSV - TAP CODE GAIN TABLE - MANUAL OPERATION
INCLUDING STEAM GENERATOR LOSSES

Parameter	Case (1) Equilibrium at 925/971	Case (2) (1) with Circ. Speed + 300 rpm	Case (3) (1)with Circ. Speed - 300 rpm	Case(4) (1) with Power + 0.005 (0.5%)	Case(5) (1) with FW Flow + 20K lb/hr	Case(6) (1) with FW Temp + 25°F
Core Power, %	27.59	27.59 (0)[a]	27.59 (0)	28.02 (+.43)	27.61 (+.02)	27.59 (0)
Helium Flow, %	38.8	42.1 (+3.3)	35.5 (-3.3)	38.8 (0)	38.8 (0)	38.8 (0)
Circulator Speed, rpm	3700	4000 (+300)	3400 (-300)	3700 (0)	3700 (0)	3700 (0)
Feedwater Flow, lb/hr	561,600	561,600 (0)	561,600 (0)	561,600 (0)	581,760 (+20,160)	561,600 (0)
Main Steam Temp, °F	934	936 (+2)	930 (-4)	963 (+29)	865 (-69)	971 (+37)
Reheat Steam Temp, °F	963	960 (-3)	968 (+5)	996 (+33)	901 (-62)	1002 (+39)
Feedwater (FW) Temp, °F	282	282 (0)	282 (0)	282 (0)	282 (0)	307 (+25)
Cold Reheat Steam (CRS) Temp, °F	539	538 (-1)	537 (-2)	577 (+38)	446 (-93)	587 (+48)
Core Inlet Temp, °F	651	678 (+27)	622 (+29)	677 (+26)	598 (-53)	689 (+38)
Core Outlet Temp, °F	1132	1121 (-11)	1148 (+16)	1166 (+34)	1079 (-53)	1170 (+39)
Steam Generator He Inlet Temp, °F	1118	1108 (-10)	1131 (+13)	1151 (+33)	1065 (-53)	1156 (+38)
Reactor Pressure, psia	656	670 (+14)	642 (-14)	672 (+16)	627 (-29)	678 (+22)
Attemperator Flow, lb/hr	0	0 (0)	0 (0)	0 (0)	0 (0)	0 (0)

TABLE 2 (continued)
FSV - TAP CODE GAIN TABLE - MANUAL OPERATION
INCLUDING STEAM GENERATOR LOSSES

Parameter	Case (7) (1) with FW Temp - 25°F	Case (8) (1) with Attemp. Flow + 5%	Case (9) (1) with CRS Temp + 40°F	Case (10) (1)with PCRV Pressure = 600 psia	Case (11) (1) with RHT Flow - 45K lb/m
Core Power, %	27.59 (0)	27.61 (+.02)	27.59 (0)	27.59 (0)	27.59 (0)
Helium Flow, %	38.8 (0)	38.9 (+.1)	38.8 (0)	36.0 (-2.8)	38.8 (0)
Circulator Speed, rpm	3700 (0)	3700 (0)	3700 (0)	3700 (0)	3700 (0)
Feedwater Flow, lb/hr	561,600 (0)	561,600 (0)	561,600 (0)	561,600 (0)	561,600 (0)
Main Steam Temp, °F	894 (-40)	868 (-66)	949 +15)	937 (+3)	951 (+17)
Reheat Steam Temp, °F	925 (-38)	885 (-78)	979 (+16)	968 (+5)	982 (+19)
Feedwater Temp, °F	258 (-24)	282 (0)	282 (0)	282 (0)	280 (-2)
Cold Reheat Steam Temp, °F	488 (+51)	369 (-170)	584 (+45)	544 (+5)	555 (+16)
Core Inlet Temp, °F	615 (-36)	605 (-46)	658 (+7)	625 (-26)	665 (+14)
Core Outlet Temp, °F	1096 (-36)	1085 (-47)	1139 (+7)	1143 (+11)	1146 (+14)
Steam Generator He Inlet Temp, °F	1081 (-37)	1071 (-47)	1125 (+7)	1127 (+9)	1132 (+14)
Reactor Pressure, psia	636 (-20)	630 (-26)	661 (+5)	600 (-56)	665 (+9)
Attemperator Flow, lb/hr	0 (0)	28,080 (+28,080)	0 (0)	0 (0)	0 (0)

[(a)] Parentheses denotes change from case (1) condition.

2.4. SUBTASK I: VALVE PERFORMANCE INSPECTION

2.4.1. Introduction

This report deals with a study to determine the temperature distribution and to verify the compatibility of auxiliary equipment (switches, elastomers, hydraulic actuators, etc.) used on large high-temperature steam valves located in the secondary coolant system of the Fort St. Vrain HTGR. Data accumulated to date are the result of steam temperatures of approximately 900°F developed when the plant was operating at approximately 28% power. Testing will continue until design steam temperatures have been attained.

2.4.2. Purpose

The objectives of this investigation are:

1. To determine valve actuator temperature resulting from heat being transferred from the steam valves' pressure boundary parts to evaluate the following:

    a. Hydraulic oil compatibility.
    b. Compatibility of elastomer used in seals.
    c. Compatibility of electrical insulation.

2. To determine position switch compatibility due to the temperature of the environment.

3. To determine the temperature of exposed metal surfaces of the valve structure to evaluate other potential hazards.

4. To determine heat loss from the valves' exposed structure.

2.4.3. Discussion

Many questions arose regarding the temperature existing on the exposed surfaces of high-temperature valves and the compatibility of the valves' appurtenances. These data were not immediately available from the manufacturer and the geometry of the valves made analysis difficult.

Six valves were instrumented to gather temperature data. These included: the main steam isolation valves (HV-2223 and HV-2224), the hot reheat steam isolation valves (HV-2253 and HV-2254), and the main steam bypass startup steam isolation valves (HV-2292 and HV-2293). Figure 1 indicates the placement of the thermocouples on each valve and Table 3 is a summary of data taken to date.

The main steam and hot reheat steam valves are normally open during the period the plant is generating electricity and the bypass valves are open only during the transitional periods of startup and shutdown to a temperature of 800°F.

Since FSV has not operated at the maximum main steam or reheat steam temperature (1000°F and 1025°F, respectively), the data are incomplete. However, very definite trends have been established and it is significant to review these data with respect to the objectives.

Thermocouple #7, attached in each case to the valve yokes' upper structure to which the hydraulic cylinder is attached, varies from 100°F to 135°F. These values are well within the operational capabilities of the Gulf Harmony 53 hydraulic fluid, the cylinder rod end seals, other seals and the amount of thermal energy transferred to the hydraulic fluid.

Thermocouple #6 is affixed to a circular disc mounted on the valve stem which picks up the limit switches' actuation arm. The temperature of this disc is indicative of the heat being transferred up the valve's stem. The temperature of the discs did not exceed 172°F.

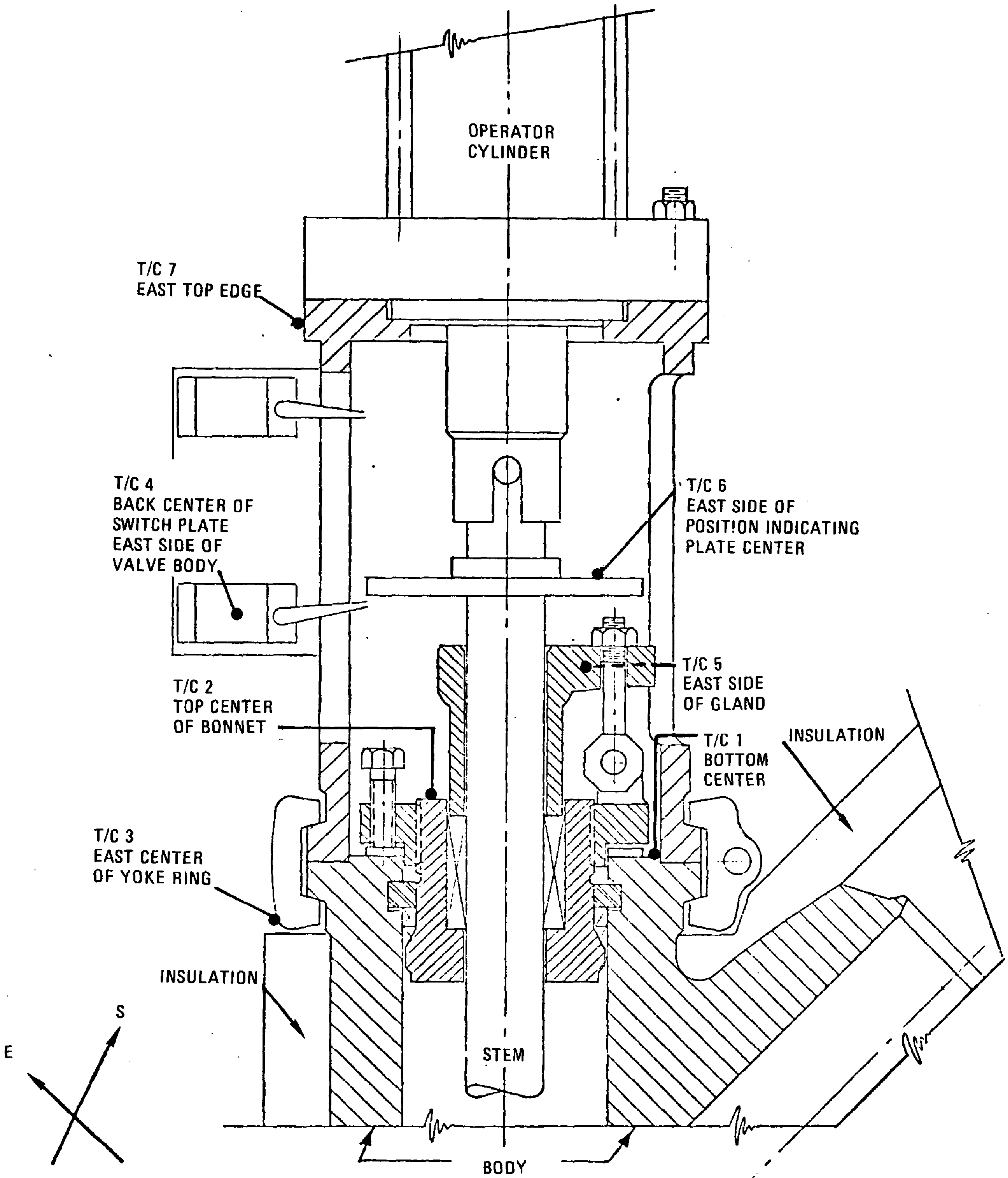


Fig. 1. Main steam isolation valve HV-2223 (typical installation)

TABLE 3
VALVE SURFACE TEMPERATURE STUDY[a]

Valve Description	Tag No.	Temperature (°F)							Process Fluid Temp. (°F)
		TC 1	TC 2	TC 3	TC 4	TC 5	TC 6	TC 7	
Main steam isolation Rockwell - 16" 4402	HV-2223	545	555	530	160	380	150	113	890
(WCB) JMMY	HV-2224	550	585	438	132	285	150	100	910
Hot reheat steam isolation Rockwell 22" X 20" X 22"	HV-2253	545	585	487	165	162	172	106	910
7502 (WC9) JMMY	HV-2254	544	575	487	187	465	160	122	910
Main steam startup bypass	HV-2292	---	---	445	150	280	136	132	775
isolation, Rockwell		305	210	302	125	---	---	112	890*
8" X 6" X 8", 4414 (WC9) JMMY	HV-2293	393	370	435	145	230	105	135	635

[a] Unless marked by * the valve was opened when temperatures were measured.

Thermocouple #5 is attached to each valve's packing gland follower. The observed temperatures with the main steam and reheat steam valves (HV-2223, HV-2224, HV-2253 and HV-2254) operating at approximately 910°F varied from 162°F to 465°F. Additional data will be of interest to see if the spread converges.

Thermocouple #4 is attached under the lower limit position indicating switch. The maximum temperatures recorded to date are in the range of 125°F to 187°F. These switches will be observed as the rise to power program continues since their upper design temperature is 200°F.

Thermocouple #3 is attached to the valve's yoke ring clamp immediately next to the insulation of the valve body. This clamp is not a portion of the valve's pressure boundary but it does join the yoke structurally to the valve body. Temperatures recorded at this location for the main steam and reheat steam valves at 910°F are 302°F to 530°F. The maximum temperatures recorded on the bypass valves (HV-2292 and HV-2293) which remain open until approximately 800°F is reached were 280°F and 230°F, respectively.

Thermocouples #1 and #2 are located on the valve body inside the yoke and on the valve bonnet, respectively. These locations were chosen to determine the hottest uninsulated metal surfaces of the valves. On the main steam and reheat steam valves (HV-2223, HV-2224, HV-2253 and HV-2254), thermocouples No. 2 registered 555°F to 585°F. The same thermocouples on the bypass valves (HV-2292 and HV-2293) did not exceed 393°F.

2.4.4. Summary

Data taken to date indicate that the valve surface temperatures are within anticipated limits.

The actuators' hydraulic fluid, seals, and directional control valves all have operational temperature capabilities beyond those required to survive temperatures indicated by thermocouple #7.

The remainder of the surface temperatures are within anticipated limits and do not create a foreseeable hazard.

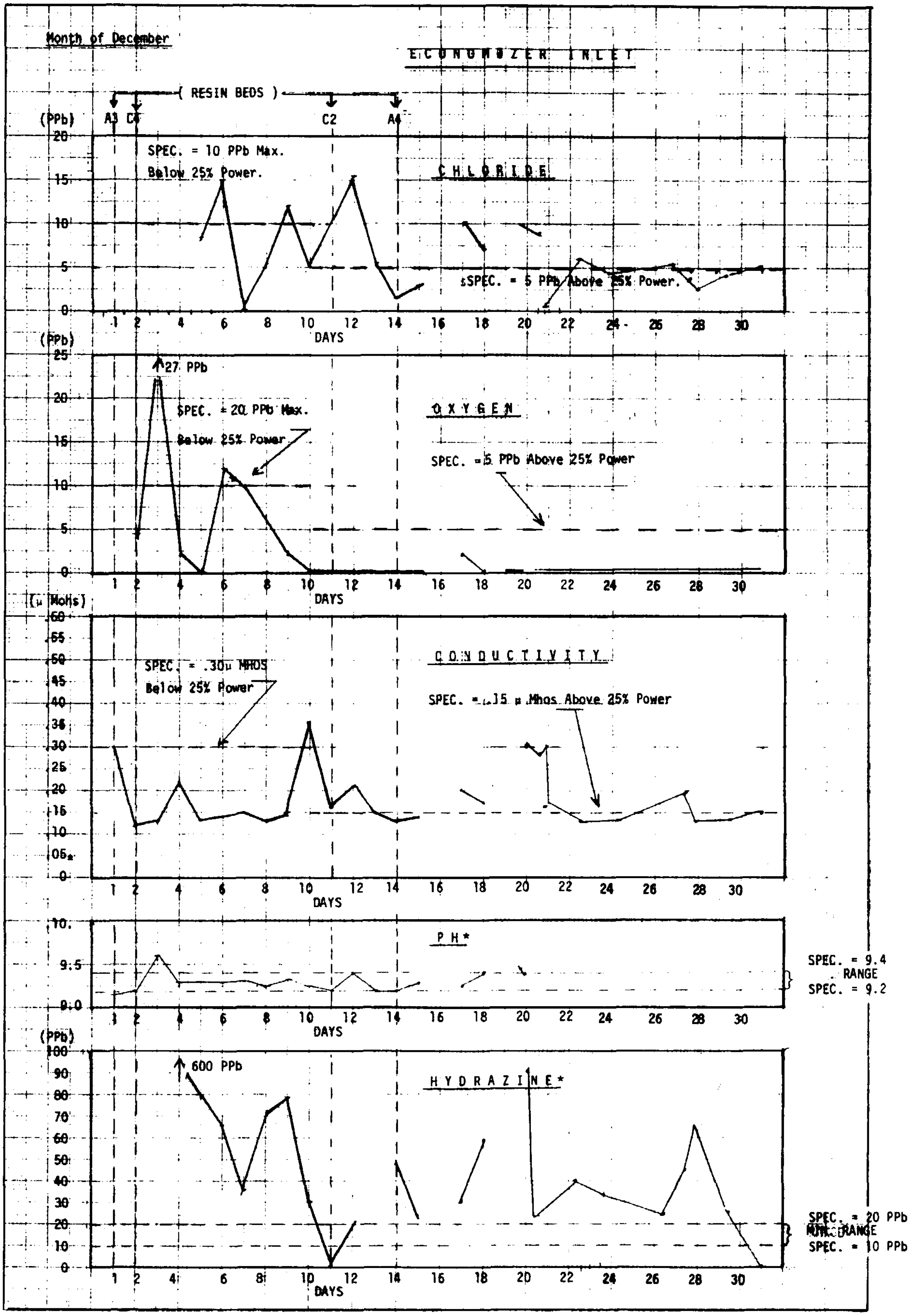
Valve heat loss calculations will not be accomplished until design main and reheat steam temperatures have been achieved.

The temperature of the limit switch (thermocouple #4) and its mounting plate will be closely monitored to determine the effect of higher power operation.

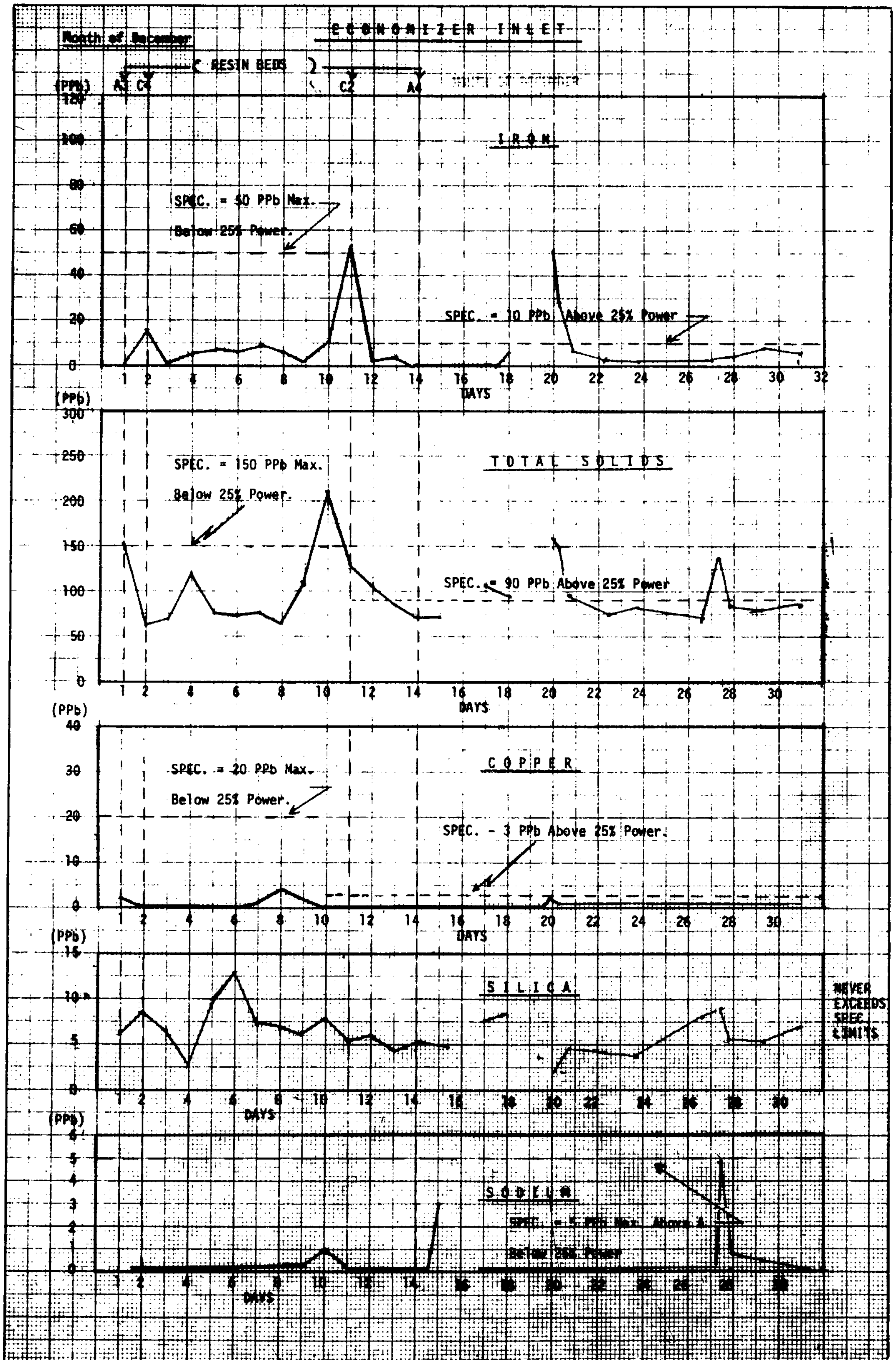
APPENDIX A

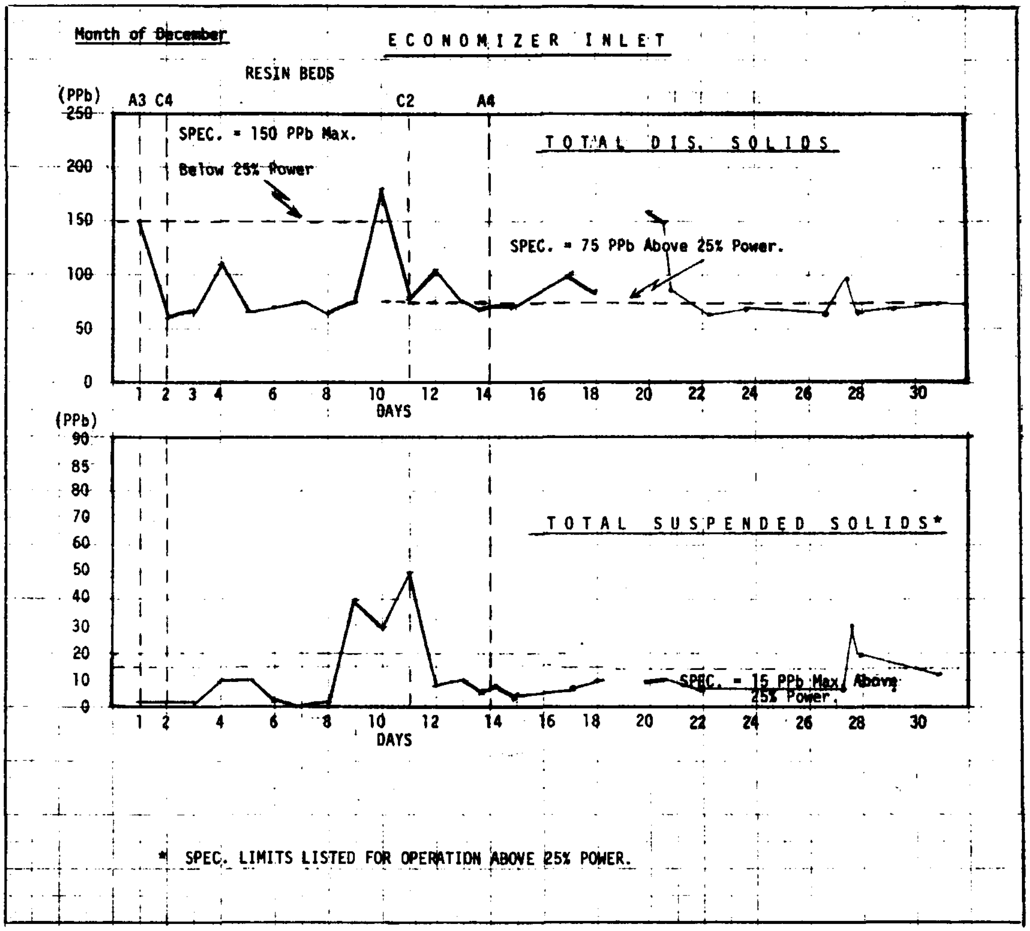
STEAM GENERATOR PERFORMANCE DATA

FEEDWATER CHEMISTRY DATA

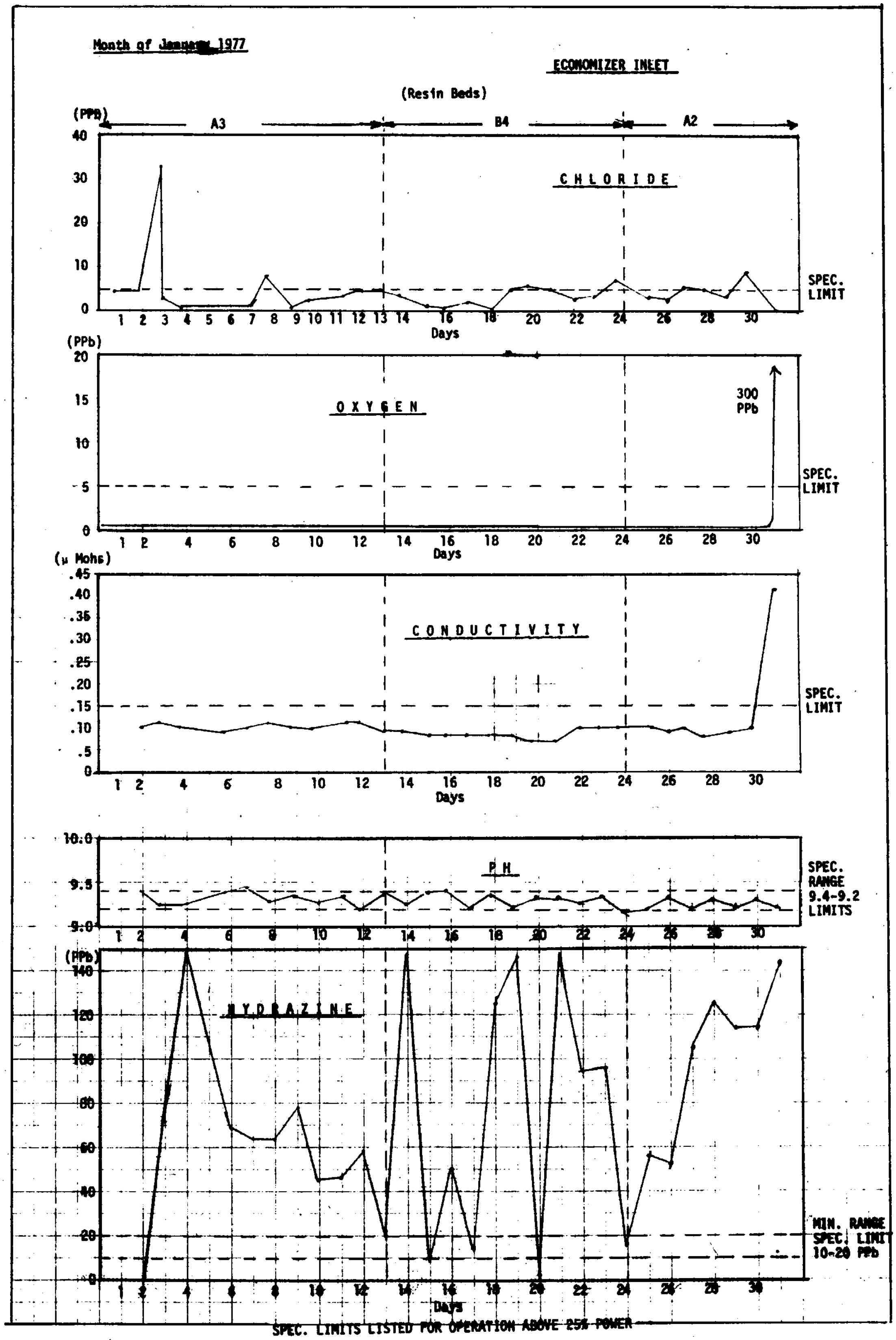


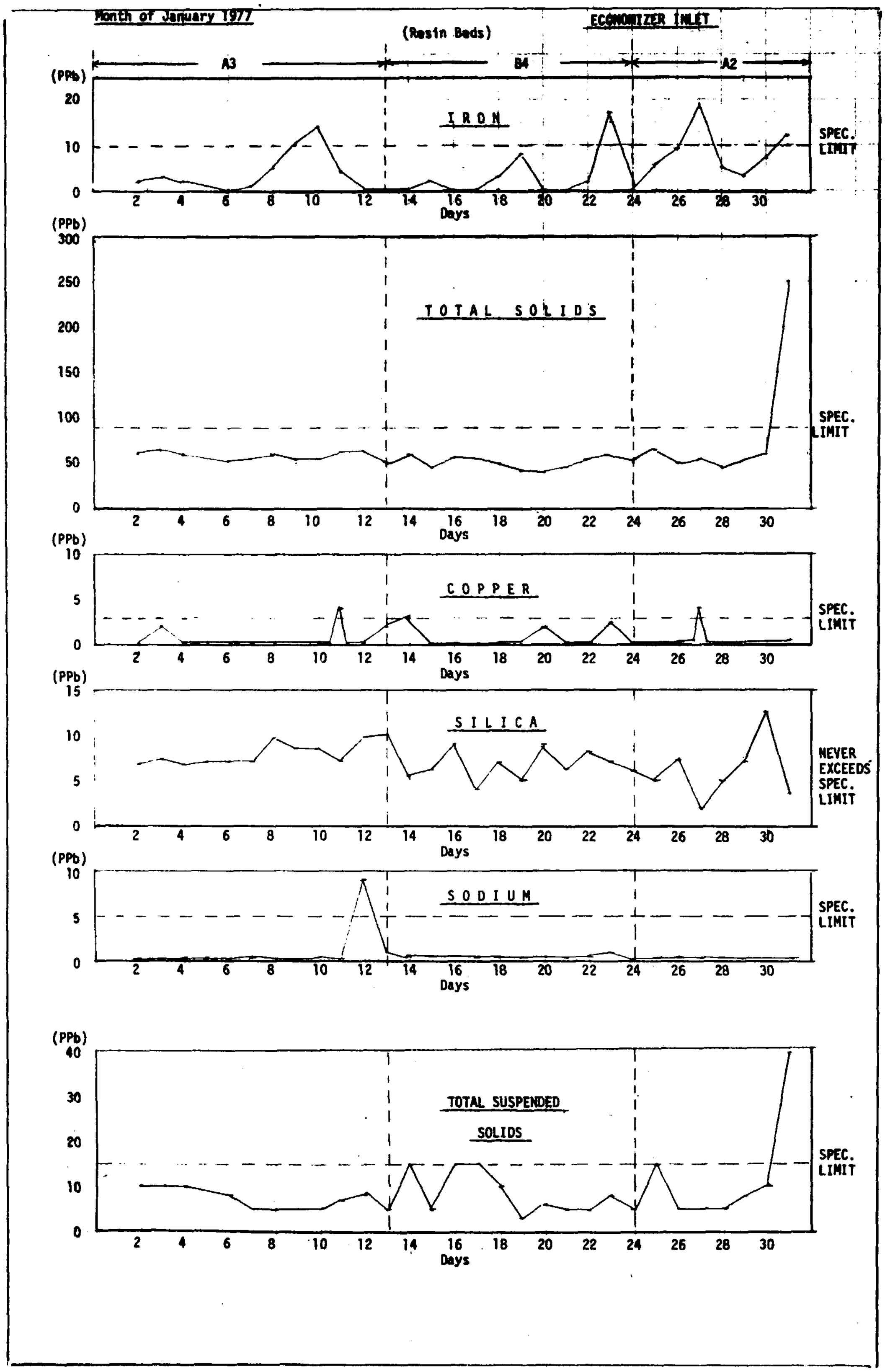
* SPEC. LIMIT LISTED FOR OPERATION ABOVE 25% POWER.

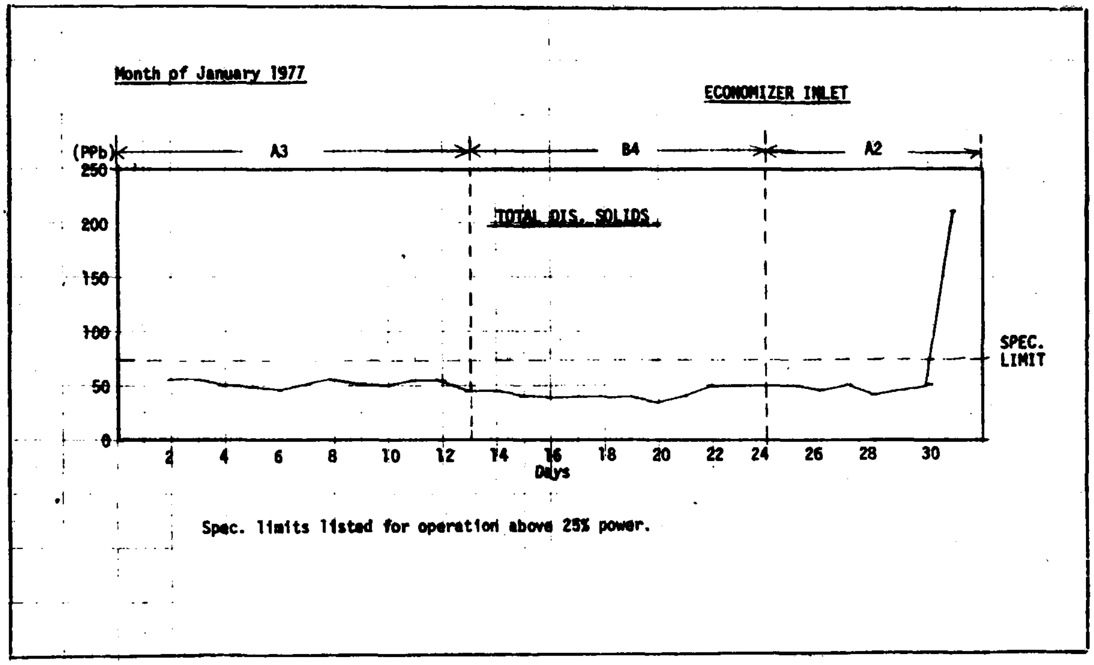

Month of December
E C O N O M I Z E R I N L E T
RESIN BEDS
A1 C4 C2 A4
I R O N
SPEC. = 50 PPb Max.
Below 25% Power.
SPEC. = 10 PPb Above 25% Power
T O T A L S O L I D S
SPEC. = 150 PPb Max.
Below 25% Power.
SPEC. = 90 PPb Above 25% Power
C O P P E R
SPEC. = 20 PPb Max.
Below 25% Power.
SPEC. - 3 PPb Above 25% Power.
S I L I C A
NEVER EXCEEDS SPEC. LIMITS
(PPb)
DAYS



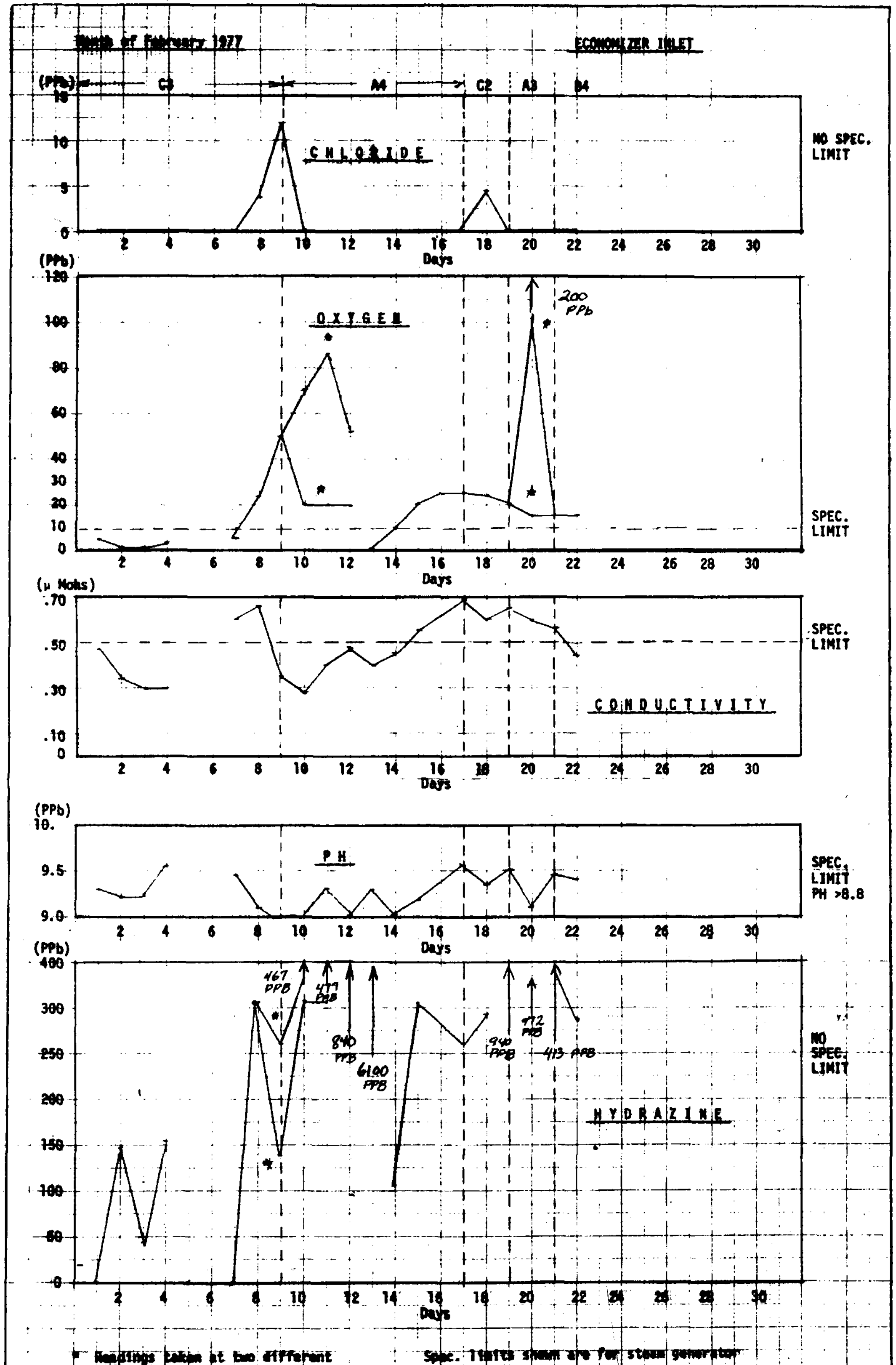
Month of December
ECONOMIZER INLET
RESIN BEDS
(PPb)
A3 C4
C2
A4
SPEC. = 150 PPb Max.
Below 25% Power
TOTAL DIS. SOLIDS
SPEC. = 75 PPb Above 25% Power.
DAYS
(PPb)
TOTAL SUSPENDED SOLIDS*
SPEC. = 15 PPb Max. Above 25% Power.
DAYS
* SPEC. LIMITS LISTED FOR OPERATION ABOVE 25% POWER.


Month of January 1977
ECONOMIZER INLET
(Resin Beds)
A3
B4
A2
(PPB)
CHLORIDE
SPEC. LIMIT
Days
(PPb)
OXYGEN
300 PPb
SPEC. LIMIT
Days
(μ Mohs)
CONDUCTIVITY
SPEC. LIMIT
Days
PH
SPEC. RANGE 9.4-9.2 LIMITS
(PPb)
HYDRAZINE
MIN. RANGE SPEC. LIMIT 10-20 PPb
DAYS
SPEC. LIMITS LISTED FOR OPERATION ABOVE 25% POWER


Month of January 1977
ECONOMIZER INLET
(Resin Beds)
A3
B4
A2
IRON
SPEC. LIMIT
TOTAL SOLIDS
SPEC. LIMIT
COPPER
SPEC. LIMIT
SILICA
NEVER EXCEEDS SPEC. LIMIT
SODIUM
SPEC. LIMIT
TOTAL SUSPENDED SOLIDS
SPEC. LIMIT
(PPb)
Days

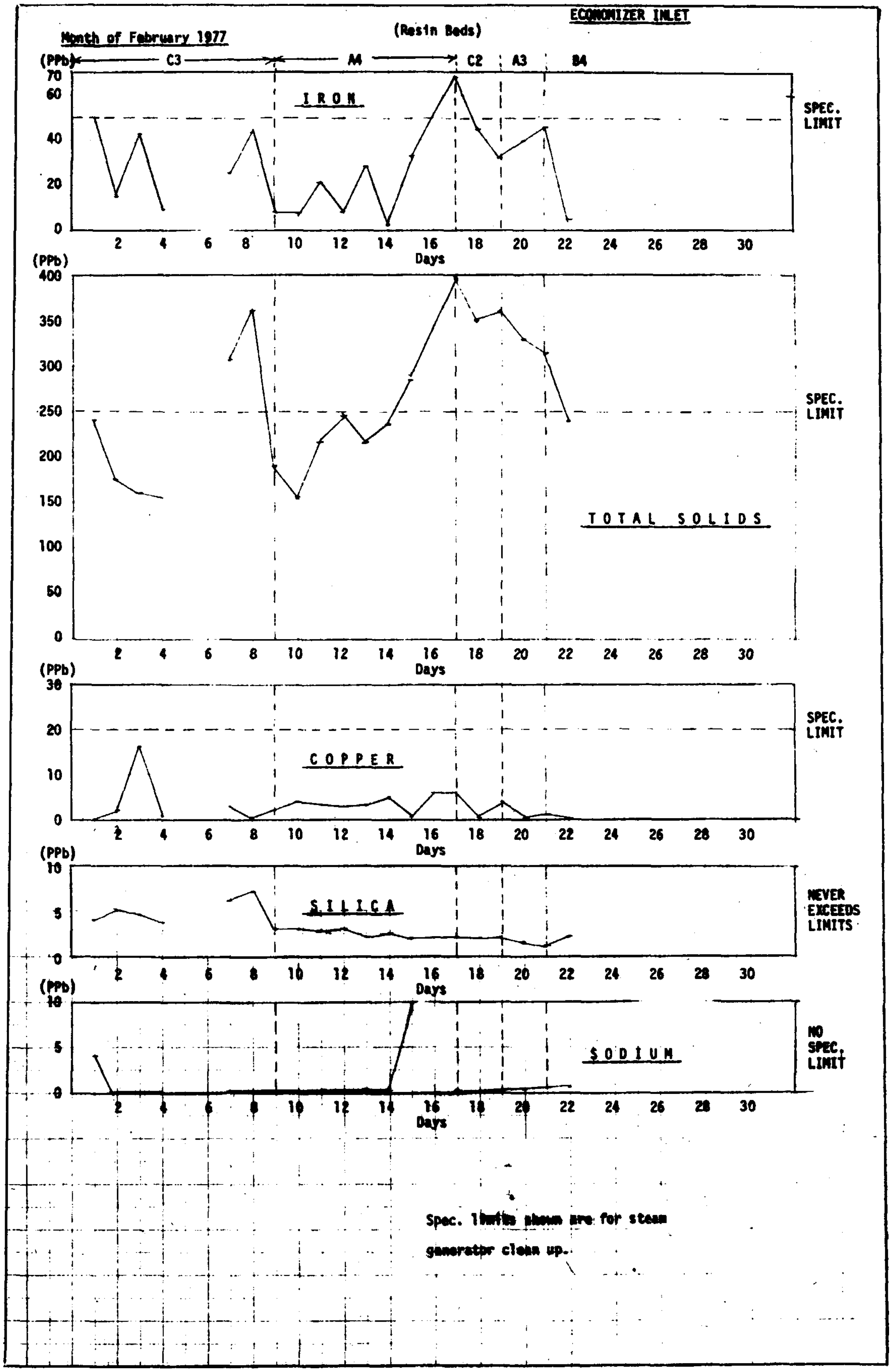


Spec. limits listed for operation above 25% power.



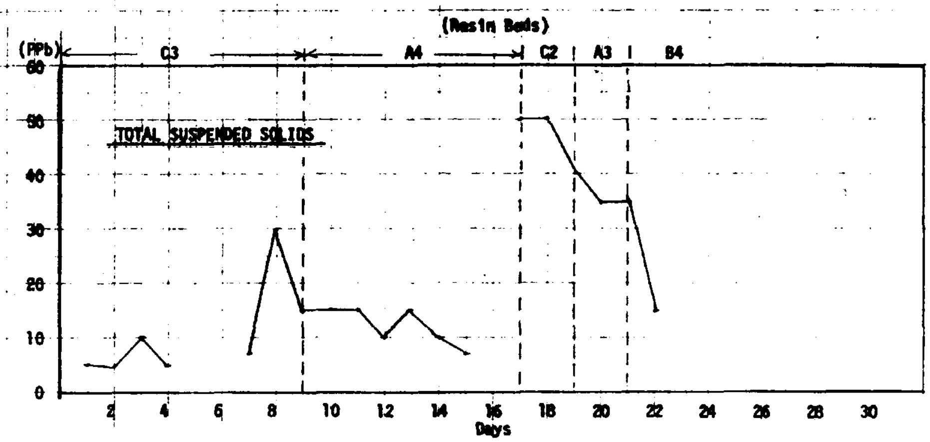
* Readings taken at two different locations.

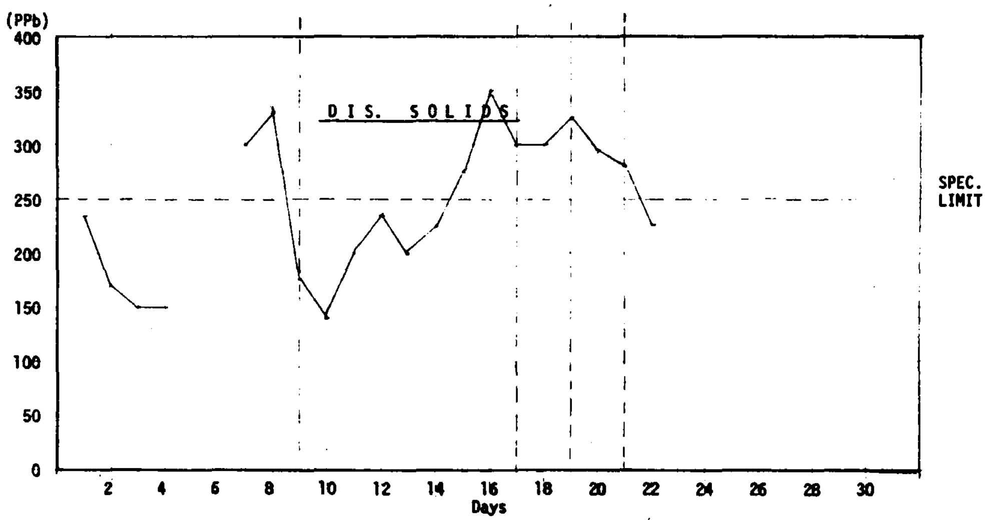
Spec. limits shown are for steam generator clean up.


ECONOMIZER INLET
Month of February 1977
(Resin Beds)
C3
A4
C2
A3
B4
IRON
SPEC. LIMIT
(PPb)
Days
TOTAL SOLIDS
SPEC. LIMIT
COPPER
SPEC. LIMIT
SILICA
NEVER EXCEEDS LIMITS
SODIUM
NO SPEC. LIMIT
Spec. limits shown are for steam generator clean up.

ECONOMIZER INLET

Month of February 1977





Spec. limits shown are for steam generator clean up.