

THE BABCOCK & WILCOX COMPANY
POWER GENERATION GROUP

J. H. JOHNSTON, DISTRICT SALES MANAGER

From *John Williamson - Fox*
J. T. JANIS, SERVICE MANAGER

BDS 603 5

Cust. SMUD

RANCHO SECO

File No.
or Ref.

Subj. MASTER SERVICES CONTRACT ADDITION

Date
MAY 25, 1978

This letter to cover one customer and one subject only

Attached is a proposed letter for establishment of Master Services Contract Task 68 for your signature and transmittal to SMUD. If you have any questions or require additional information, please advise.

JTJ/rh
Attachment

cc: GG Anderson
R. Berchin
RP Breeding
JM Burnett
HS Muir
BA Karrasch
GM Olds
JD Phinney
DG Culberson - SPR-330 File

3-11-330

7910040526

May 25, 1978

Mr. J. J. Mattimoe
Assistant General Manager & Chief Engineer
Sacramento Municipal Utility District
6201 S. Street
Sacramento, California

Subject: Master Services Contract
SMUD Contract 6367, B&W No. 595-7072
Task 68 - Analysis of March 20, 1978 Reactor Trip
and Rapid Cooldown Transient

References: 1. B&W letter, JT Janis to RJ Rodriguez, dated March 23, 1978
2. B&W letter, JT Janis to RP Oubre, dated April 5, 1978

Dear Mr. Mattimoe:

As a result of the March 20, 1978 Loss of NNY Power Incident and subsequent rapid cooldown transient, B&W was requested to perform an evaluation of the effects of the transient on the Reactor Coolant System (RCS). The following Task Proposal has been prepared to cover the analysis work performed to date and to obtain your approval for additional analyses to address NRC concerns and determine the long term effect on the life of RCP casings. This task is proposed under the terms and conditions of the subject Master Services Contract.

Task 68 - Analysis of the March 20, 1978 Reactor Trip and Rapid Cooldown Transient

Task 68-01 - Evaluation of Transient

Scope

- a. The Company will perform an evaluation of the effects of the transient conditions on the major components of the RCS and the nuclear fuel assemblies on an expedited basis and provide SMUD with a letter reporting its recommendations. This evaluation will be performed based on transient data provided by SMUD.
- b. The Company will perform a detailed QA review of the calculations and analyses performed, including verification of validity of the input data used. A detailed report will be provided along with any additional

Estimated Cost

Charges will be on a time and material basis at the appropriate Master Services Contract rates in effect at the time work is performed. Item (a) above is completed and item (b) is in the report preparation stage. Through April, 526 MH and 0.5 computer hours have been expended and it is estimated an additional 200 MH will be required to complete the task. Travel and living expenses will be billed at cost.

Schedule

Item (a) has been completed (Reference 1). Item (b) report should be ready for transmittal about July 1, 1978.

Task 68-02 - Follow-on Fracture Mechanics AnalysisScope

In task 68-01 (a) above, a quick and conservative fracture mechanics analysis of the reactor vessel beltline region and outlet nozzle region was performed. This was an ASME Section III type analysis using a postulated flaw size of 1/4 the vessel thickness. The factors-of-safety calculated for the beltline region and outlet nozzle region are 1.8 and 1.2 respectively.

Since the NRC staff expressed an interest in analysis of factors-of-safety for smaller flaw sizes, B&W proposes to perform additional fracture mechanics analysis for the reactor coolant pressure boundary areas most susceptible to flaw growth. These areas are the reactor vessel beltline region, reactor vessel outlet and inlet nozzle regions, and the lower head of the OTSG just below the tube sheet to shell junction. The analysis will be performed for a series of flaw sizes ranging from the minimum reportable flaw size defined by Section XI to a flaw depth of about 0.5 a/w. The final output of the analysis will be a plot of factors-of-safety versus flaw sizes and plots of factors-of-safety versus time during the transient for several flaw sizes. The application of these curves would be that once the desirable factors-of-safety are established, SMUD would need only to demonstrate that it's NDT technique would have detected the associated flaw size during the baseline inspection.

Estimated Cost

Charges will be on a time and materials basis at the appropriate Master Services Contract rates in effect at the time work is performed. It is estimated that approximately 310 MH will be required to complete this task. Travel and living expenses, if any, will be billed at cost.

Schedule

It is estimated that the span time to complete this analysis will be eight (8) weeks from date of task authorization.

Task 68-03 - Effect of Transient on Accumulated Usage Factor of Reactor Coolant PumpScope

The effect of the large temperature and pressure transients on the pump casing and cover is best evaluated by the responsible engineer who performed the stress analysis on these pumps. It is believed that these transients could have caused some cyclic type stresses which will increase the lifetime usage factor slightly. B&W will contact with the engineering firm who performed the original stress analysis to perform this additional fatigue evaluation.

Estimated Cost

The cost quotation received from the engineering firm is \$8,000. This cost will be billed under the Procurement terms of the Master Services Contract.

Schedule

It is estimated that the span time to complete this analysis will be six (6) weeks.

Work on Task 68-01 was initiated on the verbal authorization of Mr. R. P. Oubre'. Work on Tasks 68-02 and 68-03 will be initiated upon authorization of this task letter.

This proposal is valid until June 30, 1978 except that the Company shall have the right to withdraw or amend this proposal at any time before formal acceptance by the purchaser and subsequent acceptance in writing by an authorized representative of the Company.

If you have any questions or require additional information, please contact me or Joel Janis in Lynchburg. You may authorize this task by signing in the space provided below and returning the copy to us.

Very truly yours,

J. H. Johnston

END:JH/nh

cc: RP Oubre' RJ Rodriguez
 DG Raasch

APPROVED: _____

DATE: _____

FRACTURE MECHANICS EVALUATION OF THE REACTOR VESSEL OF THE SMUD UNIT AT PARCHO
SECO FOR THE UNANTICIPATED TRANSIENT OF 3/7/78

A fracture mechanics analysis of the reactor vessel has been performed using the conservative approach outlined in the ASME Code, Section III, Appendix G. Specific details of the analytical approach are documented in B&W Topical Report BAW-10046A, Rev. 1.

Analyses were performed on the two most critical areas of the reactor vessel - the beltline region and the outlet nozzle region. At the lowest temperature during the transient the material is still in the upper shelf region (ductile behavior). Due to the low level of radiation, degradation to beltline region materials is not significant enough to produce a shift in the transition reference temperature. Consequently, the beltline region materials are also at the upper shelf toughness region.

Factors-of-safety for the beltline region and outlet nozzles have been calculated as follows:

$$F. \text{ of } S. = \frac{K_{IR}}{K_{Im} + K_{It}}$$

where K_{IR} is the reference stress intensity factor

K_{Im} is the stress intensity factor due to pressure

K_{It} is the stress intensity factor due to the thermal gradient through the thickness

The factor-of-safety for the beltline region is 1.8.
The factor-of-safety for the outlet nozzles is 1.2.

Detailed calculations are provided on sheet 3 of 3.

It should be recognized that this is a very conservative analysis. A postulated flaw size of 1/4t is assumed for the beltline region and the 3.0" nozzle corner flaw is assumed for the outlet nozzle. The material fracture toughness (reference stress intensity value) used is 200 ksi/in. Finally, the calculation of the stresses associated with the transient is conservative.

Charles E. Harris

Charles E. Harris, PE
March 23, 1978

Henrik S. Palme

Henrik S. Palme
March 23, 1978

CALCULATIONS:BASIC RELATIONSHIP $2K_{IM} + K_{IT} \leq K_{IR}$ BELTLINE REGION

$$(2K_{IM} + K_{IT})_{MAX} = 174.366 \text{ Ksi}\sqrt{\text{in}} \quad (\text{M13CCAT})^{**}$$

LQ91 3/23/78

$$K_{IM} = P M_m \frac{r_i^2 + r_o^2}{r_o^2 - r_i^2} \quad (\text{BFW-10046A, Rev. 1})$$

Removing the F. of S. of 2

time = 40 mins.
temp $T_{IT} = 468^\circ\text{F}$

$$P = 2100 \text{ Psi}$$

$$r_i = 85.5 \text{ in.}$$

$$r_o = 94.0 \text{ in.}$$

$$M_m = 2.9 \quad (\text{ASME Code Appendix G, Fig. G-2214.1})$$

$$K_{IM} = 64.45 \text{ Ksi}\sqrt{\text{in}}$$

$$K_{IT} = 174.366 - 2(64.45) = 45.47 \text{ Ksi}\sqrt{\text{in}}$$

$$F. \text{ of } S. = \frac{200 \text{ Ksi}\sqrt{\text{in}}}{64.45 + 45.47} = 1.82$$

$$F. \text{ of } S. = 1.82$$

NOZZLE REGION

$$(2K_{IM} + K_{IT})_{MAX} = 247.286 \text{ Ksi}\sqrt{\text{in}} \quad (\text{M14CCAT})^{**}$$

LQ91 3/23/78

time = 75 mins
temp $T_{IT} = 392^\circ\text{F}$

$$K_{IM} = P F (\%a) \frac{r_i^2 + r_o^2}{r_o^2 - r_i^2} \sqrt{f_a} \quad (\text{BFW-10046A, Rev. 1})$$

Removing F. of S. of 2

$$P = 2100 \text{ Psi}$$

$$r_i = 84.1875 \text{ in.}$$

$$r_o = 96.3125 \text{ in.}$$

$$F(\%a) = 1.74 \quad (\text{WRC Bulletin 175, Fig. A5-1})$$

$$a = 3.0 \text{ in.} \quad (\text{Appendix G, Section III, ASME Code})$$

$$K_{IM} = 83.9 \text{ Ksi}\sqrt{\text{in}}$$

$$K_{IT} = 247.286 - 2(83.9) = 79.49 \text{ Ksi}\sqrt{\text{in}}$$

$$F. \text{ of } S. = \frac{200 \text{ Ksi}\sqrt{\text{in}}}{83.9 + 79.49} = 1.22$$

$$F. \text{ of } S. = 1.22$$

Charles E. Harris, Jr.
March 23, 1978

Computer runs M13CCAT and M14CCAT are based on a computer code developed to perform the BFW-10046A, Rev. 1 calculations. It should be noted that the thermal stress intensities are calculated based on a uniform temperature through the vessel thickness that may not be the same as for this transient.

COLD LEG RC TEMPERATURE, F

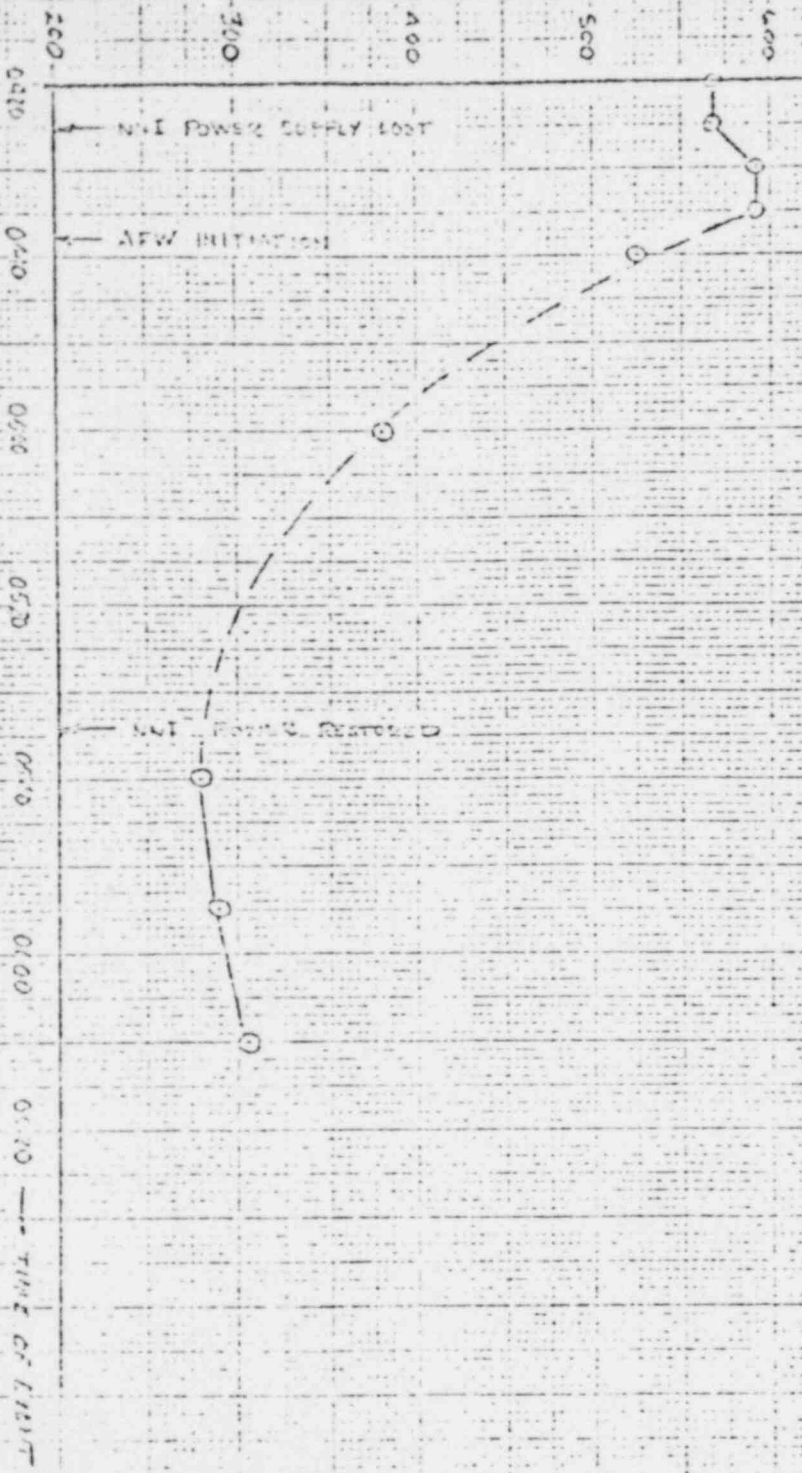


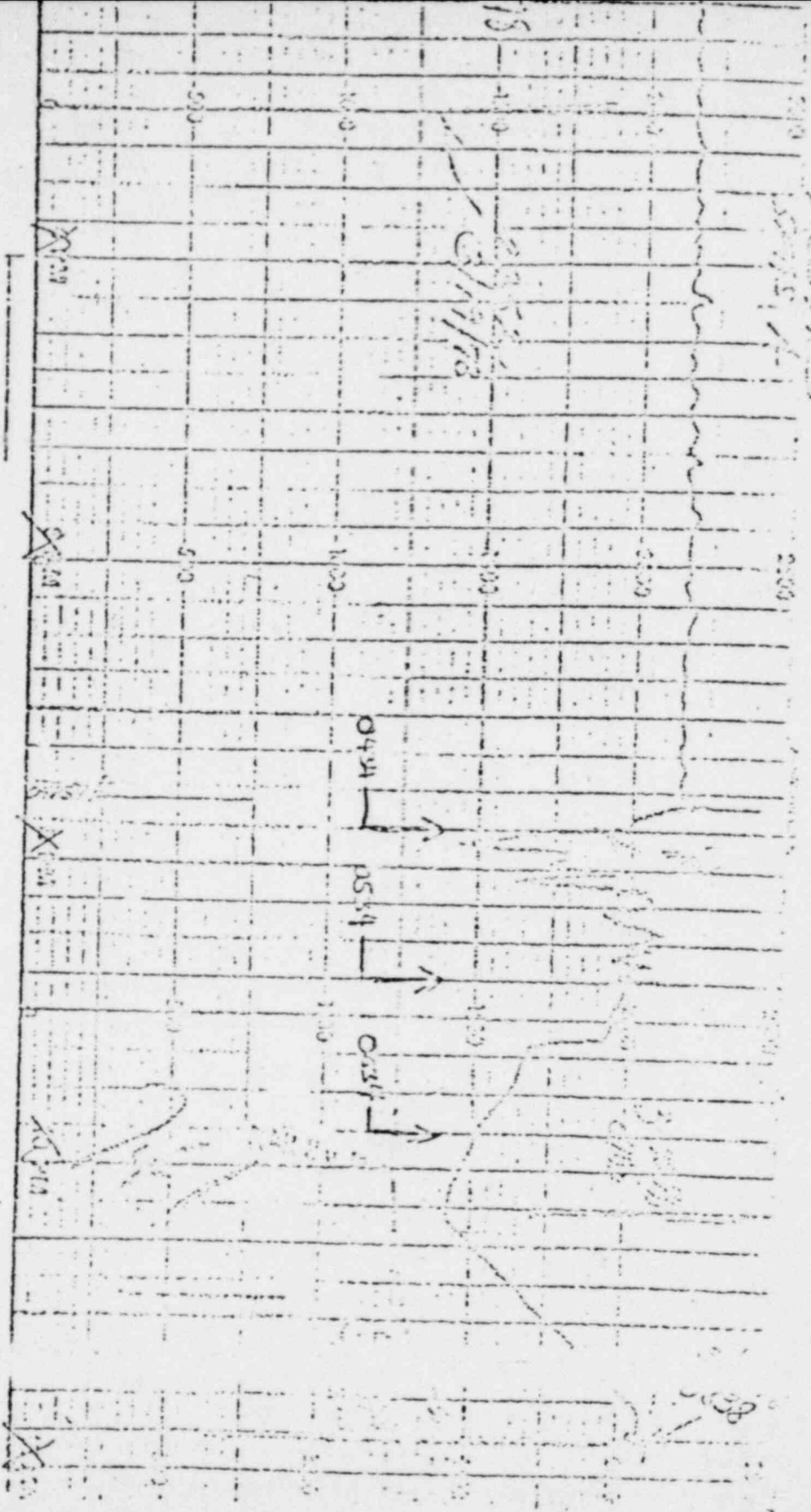
FIGURE 1
 SNUD RAPID COOLDOWN
 DC TEMPERATURE

10-10-68
 46-1000

46-1000

46-1000

0425



CC: SA KARRASCH
JM Burnett
CW Bruny/MW
JR Burris

THE BABCOCK & WILCOX COMPANY
POWER GENERATION GROUP

TO J. T. JANIS, NUCLEAR SERVICE, NPGD

From P. A. SHERBURNE/L. H. BOHN, STEAM GENERATOR UNIT

BDS 603.5

Cust. SMUD

File No.
or Ref.

Subj. SMUD RAPID COOLDOWN--EFFECTS ON OTSG INTEGRITY

Date
MARCH 23, 1978

1.0 RECOMMENDATION

Our preliminary assessment of the potential damage to the SMUD steam generator during the 3/20/78 rapid cooldown transient is complete. Based on that assessment, we feel that the steam generators have not been adversely affected and that the SMUD plant can return to power operation with the following qualifications:

- (1) Plant operation should be limited to 75% power pending (a) confirmation of transient data supplied by SMUD to date, (b) QA of shell temperature calculations made by B&W, and (c) review and approval of OTSG tube load calculations made by C. W. Bruny (MW).
- (2) Confirmation from SMUD that the Loose Parts Monitoring System is operational, especially in the area of the upper tubesheet.

In addition, SMUD should be informed of B&W's intention to perform an inspection of the steam generators during the next outage.

2.0 DISCUSSION

Figures 1-3 give the histories of RC cold leg temperature, steam pressure, and OTSG water level as reconstructed from data supplied by SMUD. The major concern with the steam generators during this transient is whether the tubes have plastically deformed due to the large axial tensile loads imposed on them. If this is the case, the tube load following return to power operation will be more compressive and tube natural frequencies will be lowered. The consequence of lesser tube frequencies is reduced safety margin for fluid elastic vibration (Connors' mechanism) of the peripheral tubes (upper span).

To assess whether or not the tube yield strength was exceeded during this transient, an estimate of average shell temperature vs. time was made. A plot of the average shell temperature as calculated is shown in Figure 4. From this calculation, the maximum tube-to-shell temperature differential (ΔT) is 170°F at 0515 hours. (Note that average tube temperature is equal to RC cold leg temperature.) This information was transmitted verbally to C. W. Bruny at Mt. Vernon who subsequently determined that 170°F tube-to-shell ΔT does not result in a tube stress greater than yield. Based on this information and on our judgement that the calculated average shell temperature is conservative, we concluded that the tube stress did not exceed yield and that the tubes were not plastically deformed. Due to the preliminary nature of this assessment, however, we advised that plant operation be restricted to 75% power pending QA of the calculations.

Additional effort is required by Mt. Vernon personnel to assess the effect of the rapid cooldown on the OTSG cumulative usage factor. This effect will be determined and forwarded when complete.

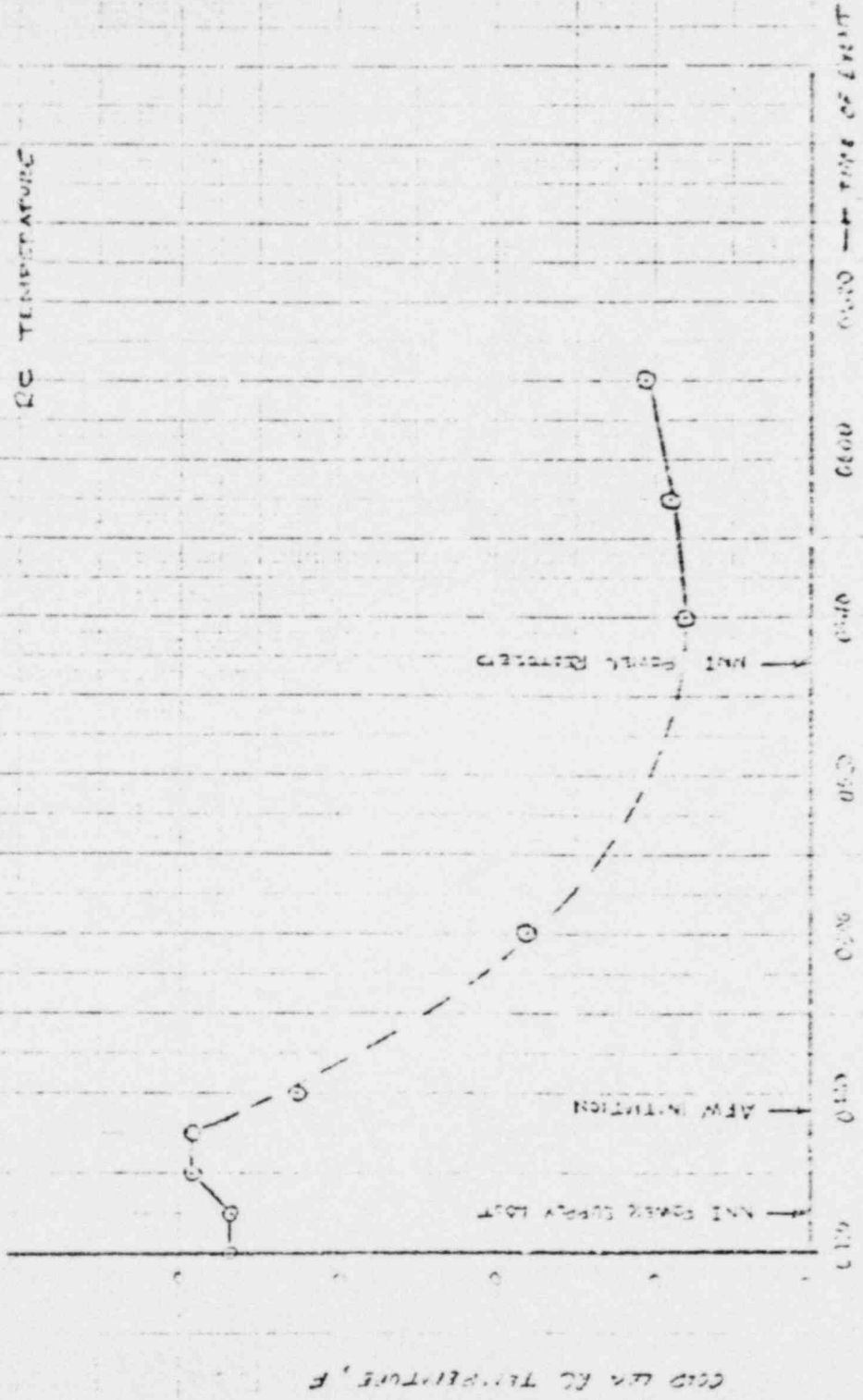
3.0 CLOSURE

It is our understanding that the above concerns and restrictions were verbally transmitted to SNUD by B. A. Karrasch on Wednesday afternoon, March 22, 1978.

PAS for LHB

PAS/LHB/ef

FIGURE 1
SMUD RAPID COOLDOWN 3/20/78
DC TEMPERATURE



DC TEMPERATURE, F

NMI POWER SUPPLY LOST

AFW INITIATION

NMI POWER RESUMED

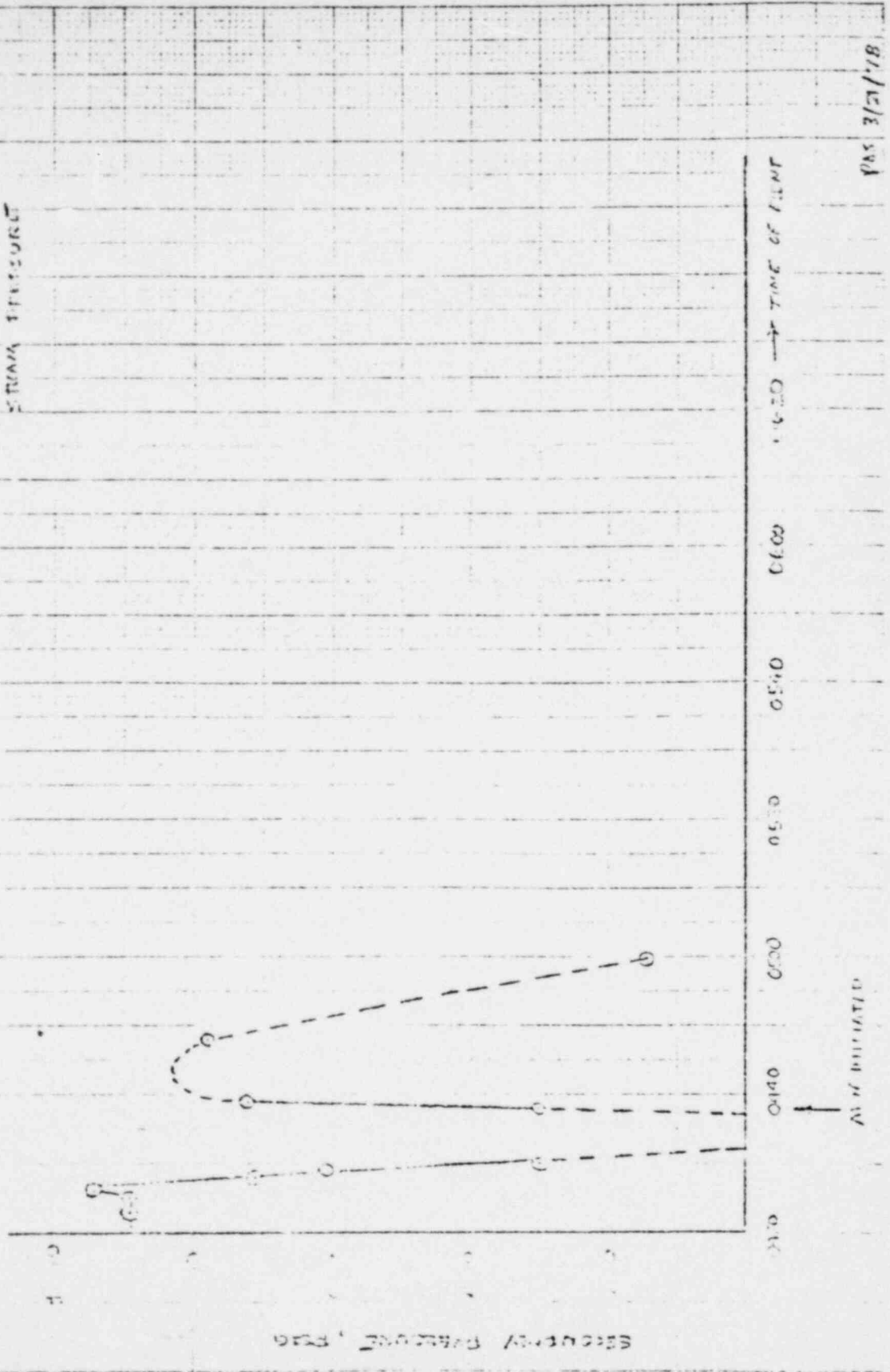
TIME OF EVENT

46-1022

FIGURE 2

SMVD RAPID COOLING 2/20/78

STRAIN RATE

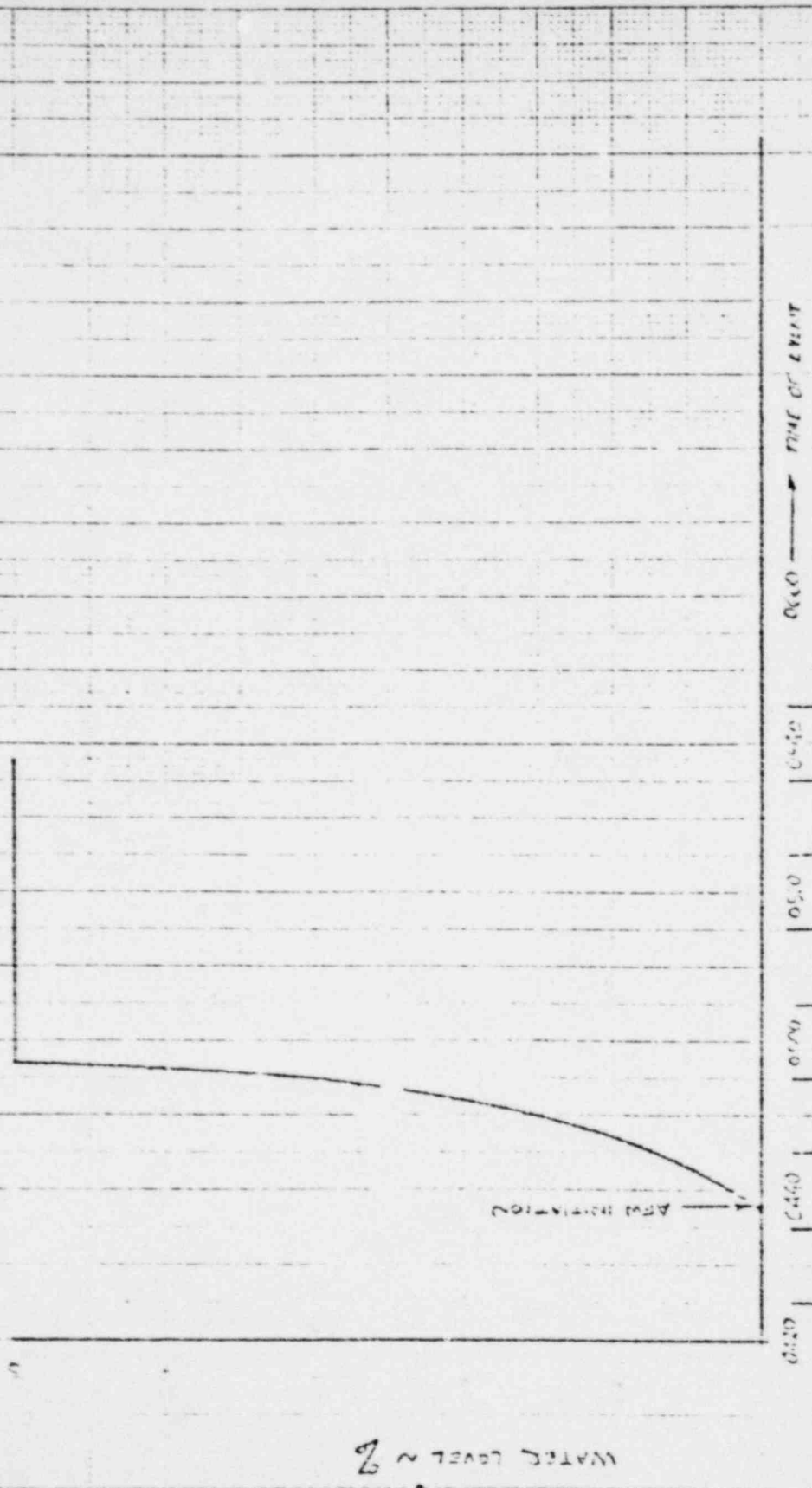


MAR 3/21/78

FIGURE 3

SMUD RAPID COOLDOWN 1/20/78

OTSG WATER LEVEL



WATER LEVEL ~ Z

0.600 — TIME OF EVENT

0.550

0.500

0.450

0.400

0.350

0.600 — TIME OF EVENT

0.550

0.500

0.450

0.400

0.350

0.300

0.250

0.200

0.150

0.100

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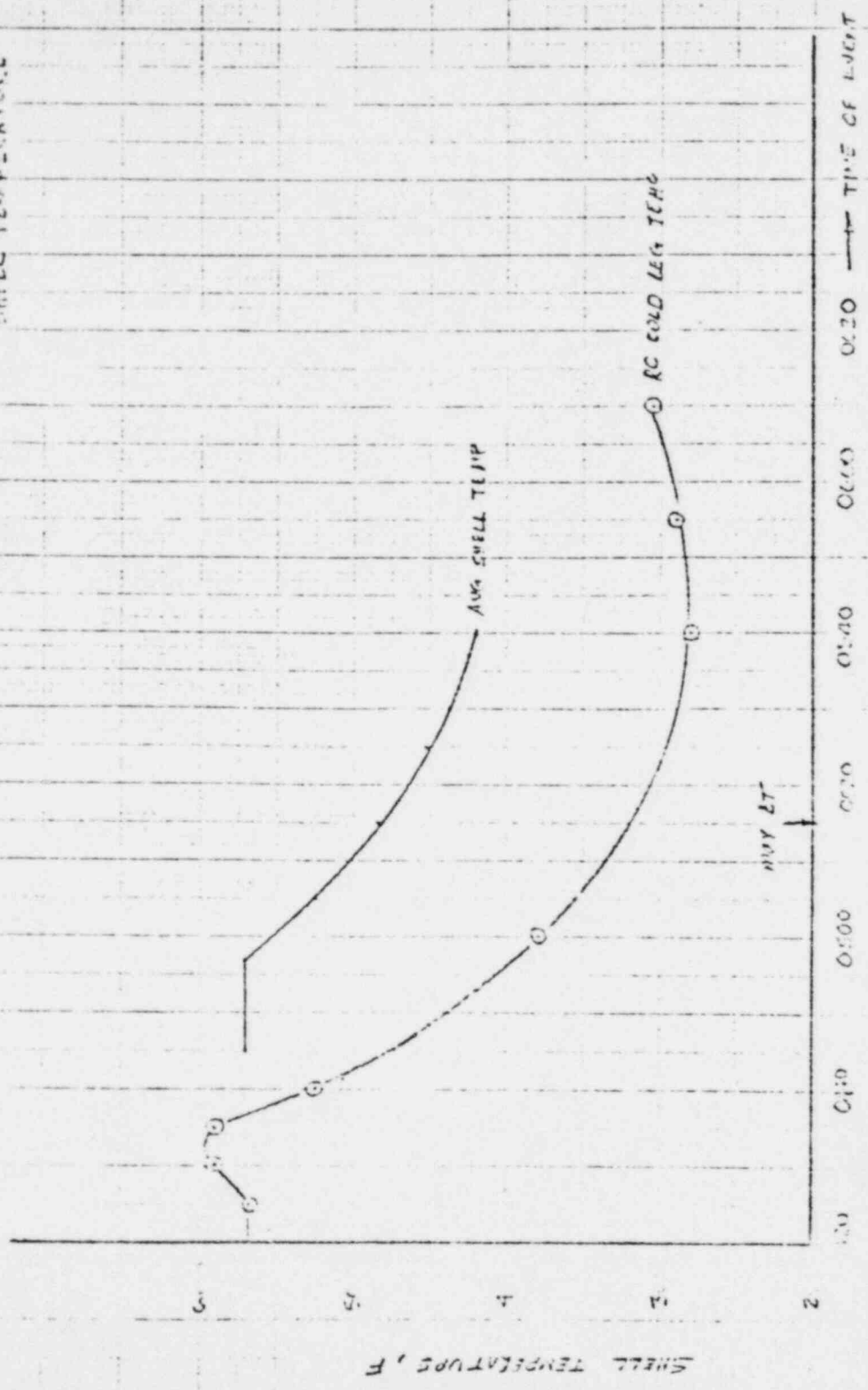
0.000

0.000

1/20/78

FIGURE 4

SMUD RAPID COOLDOWN (3/20/78)
SHELL TEMPERATURE



Bill Barrett

3/20

8:15

RCS Chemistry

AFTER TRAVIS

I₁₃₁ 3.88×10^{-2} I₁₃₃ 4.52×10^{-2} I₁₃₅ 2.32×10^{-2} gross β 3.16×10^{-1} I $\sim 10 \times$ normalgross β last week ave $\sim 2.2 \times 10^{-1}$ gross α in progress. have in AM

SEQUENCE OF EVENTS IN PLOG

5 PAGES

- 1 3 pages - memory trip review
- 4ch RC Pressure Trade
- 5ch RCS TEMP

John McGowan
H (416) 944-2936

9:00

30min

Region 5 - Bill Johnson

NEW Review & Recommendation

KMC issue

No NLE April 2000

Bill Gravel

Don Bletcher

A-Cold Log

PRF work up

04:29:30	582.8
04:30:10	587.8
04:30:30	590.9
04:31:20	593.1
04:32:20	594.9
04:33:30	595.1
04:34:20	594.9
04:35:30	590.4
04:36:30	571.7
04:37:30	573.1
04:38:30	566.6
04:39:30	558.6
40:30	528.3

PRF
Experience Muffler
at 2155.

04:26:45	585.2
	588.2
	586.1
	586.6
	587.1
	587.1
	587.6
	587.6
	588.1
	588.1
	588.8
04:29:30	588.8

PRT Hi Temp 04:25:59 210° on
disc. #
RV-1B

NR 2125, 2119, 2114, 2115
WR 2103, 2107, 2028

04:28:29 20FF High disc. temp.
Electromatic

PRT data

8:01 pumped down
8:15 secured
10:44 Press. = 118 psi
Added disin. water, down to 20 psi
Incr. P to 2155, PRT press. increased
1900-2000 lbs.

D inverter after
J inverter
circuit breakers
open upon
trip of one
power supply -

Gross α

$$1.11 \times 10^{-8} \text{ mCi/ml} \pm 7.43 \times 10^{-9}$$

Gamma Scan

Net 239 - 106.1 KEV

ACTIVITY 9.7×10^{-5} mCi/ml2 σ limit $\pm 3.01 \times 10^{-5}$

$$\text{Gross } \alpha \quad 1.11 \times 10^{-8} \text{ mCi/ml} \pm 7.43 \times 10^{-9}$$

→ 2000st 532 ONE CODE RELIAX LEAKAGE

87.4" incrs in PZR LOWEST LEVAC

1975st 15 min
2000st 20 min

HOW MANAGING PZR LEVEL

TOP AT 0425

SECURE 4th ZCP EARLY (1)
2000 / PZR LEVELS adjustment 432 ALSO

5:14 ← 502 HTI

		↓ 1.10"	
0438	13 →	557	1586
0439	14 →	546.5	1510
PLANS 270 AT 27 27	15 →	435	1475

FULL RANGE A 256

0438 25
0440 166

0425:59

0428:29

SEQUENCE OF EVENTS

TIME	EVENT
0425:35	LOST NNI PWR SUPPLY
0425:46	TURB TRIP / RX TRIP (HP)
0426:17	HPI "B" ON (MANUAL)
0427:23	" " NORMAL (OFF)
0431:25	RX LOW PRESS TRIP (
0437:56	TRIP "B" SFAS ANALOG
0437:57	" "A" " " " " WR PRESS
0437:58	HPI MIA ON / C SFAS 1543 1541
0443:	1670, 1665, 1662 WR RES PRESS 1545
0443:56	A HPI SECURED
0446:09	B LPI SECURED
0446:39	A " " "
0449:54	A HPI INITIATED (MANUAL)
0451:25	"D" RCP SECURED
0451:52	A HPI SECURED
0455:23	A HPI INITIATED
05:05:30	" B HPI SECURED
05:05:42	B HPI INITIATED
05:23:04	B HPI SECURED
05:33:19	B HPI INITIATED
05:34:04	B HPI SECURED

AT 04:44:16 - CALCULATED HT UP/O/S RATE 550°/HR
 AT 04:45:16 " " " " 291.1°/HR
 SWITCH SWITCH DURING CALC

05:34: RESTORED NNI X&Y PWR SUPPLIES

AUX SW WITH SFAS

	3	4	5	6	7	8	A	B	A cont	B cont	C cont	D cont
4:19:45	-4.05	-4.00	72.5	71.3	71.6	71.9	579.7	579.1	564.9	564.1	564.1	567.5

5/20/00
 2:10
 2:10

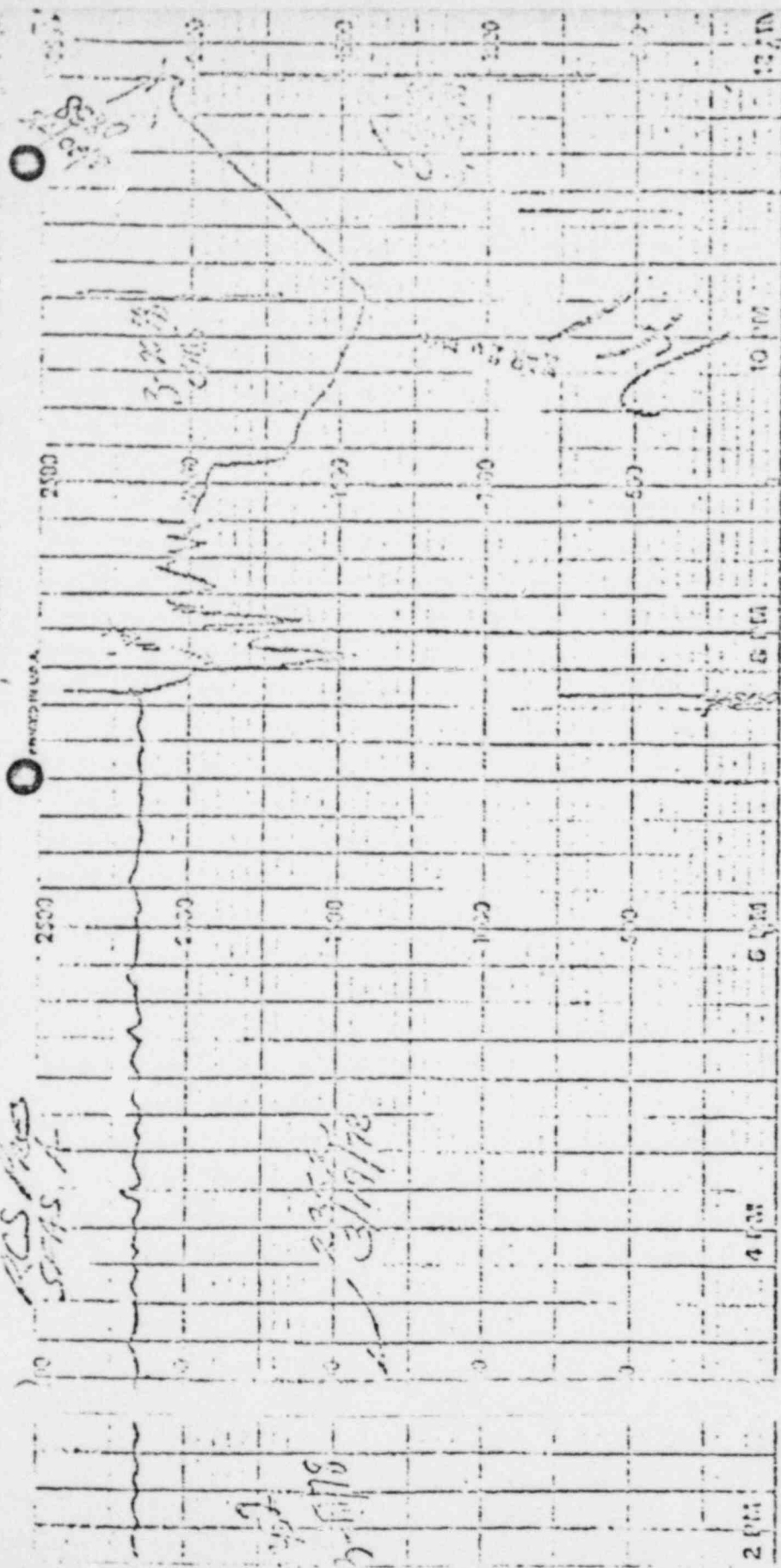
G100 - AVE PER LEVEL
T009 - PZR TEMP
● 23 - A RCS FLOW
G122 - B " "
T067 - A FW TEMP
T208 - B FW TEMP
L035 - A OTSG LVL
L036 - B " "
G12 - IMBALANCE

04:24:00 - 179.4 - 648.2 - 69.24 - 68.96 - 437.0 - 431.8 - 170.6 - 176.7 -

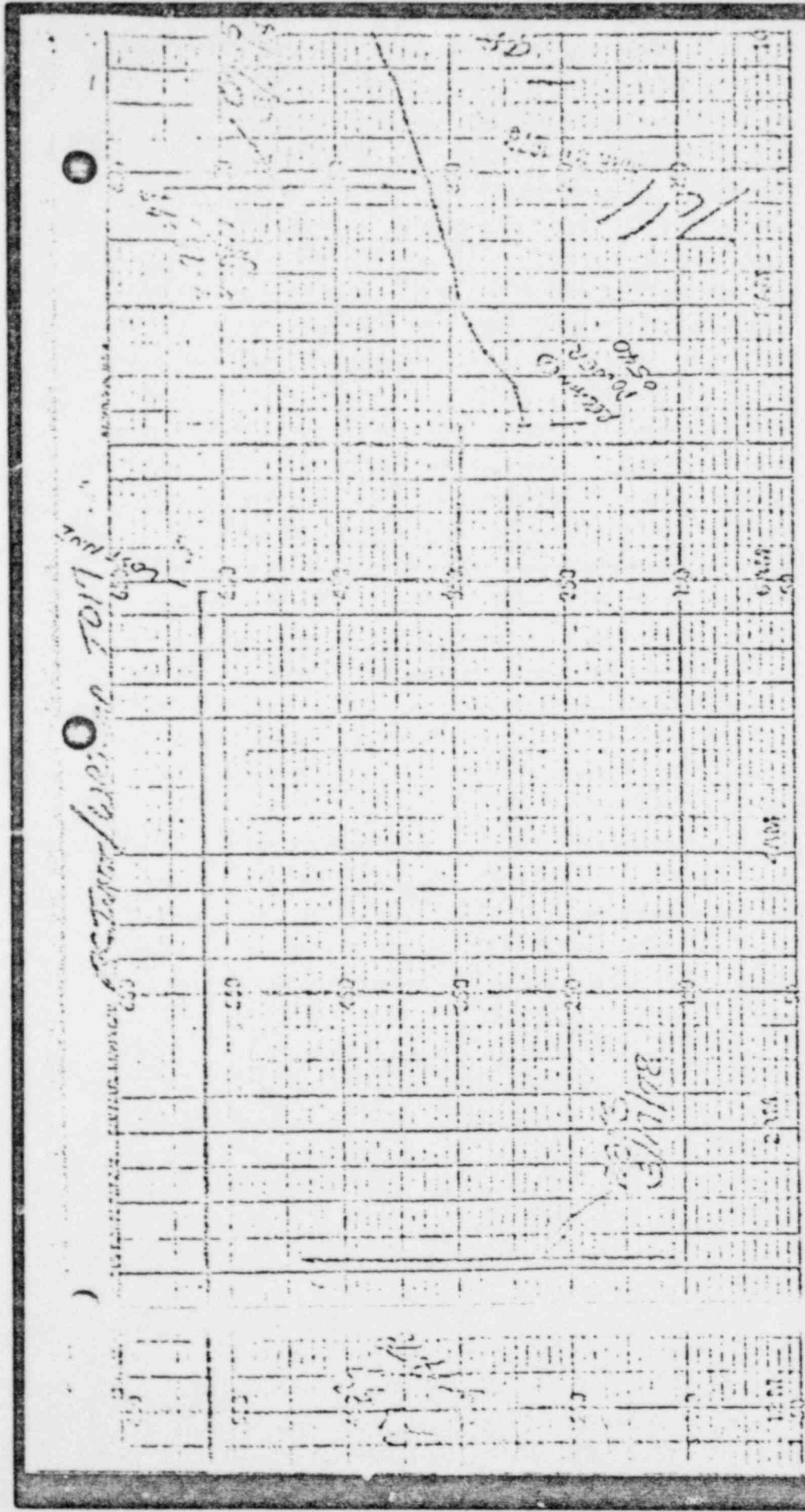


Table with multiple columns of numerical data and a right-hand column containing values such as .00, .01, .02, .03, .04, .05, .06, .07, .08, .09, .10, .11, .12, .13, .14, .15, .16, .17, .18, .19, .20, .21, .22, .23, .24, .25, .26, .27, .28, .29, .30, .31, .32, .33, .34, .35, .36, .37, .38, .39, .40, .41, .42, .43, .44, .45, .46, .47, .48, .49, .50, .51, .52, .53, .54, .55, .56, .57, .58, .59, .60, .61, .62, .63, .64, .65, .66, .67, .68, .69, .70, .71, .72, .73, .74, .75, .76, .77, .78, .79, .80, .81, .82, .83, .84, .85, .86, .87, .88, .89, .90, .91, .92, .93, .94, .95, .96, .97, .98, .99, 1.00. The table is oriented vertically on the page.

RCS
SAS. 1-



11 11 11



THE BABCOCK & WILCOX COMPANY

POWER GENERATION GROUP

To Distribution List D

From G. M. Olds - Manager, Nuclear Service Department
J. C. Deddens - Manager, Engineering Department

BDS 623-5

Cust.

File No.
or Ref.

Subj. SMUD NNI Incident Recovery Team

Date March 21, 1978

This letter is cover one customer and one subject only.

The Rancho-Sequoia plant suffered a loss of the NNI System for approximately an hour on 3/20/78. A severe plant transient resulted.

A task force is appointed via this memo to reconstruct the event and to investigate any significant effects on the NSS equipment involved. The task force is expected to derive B&W recommendations to the customer based on their findings. In addition, they will assist SMUD in gaining NRC concurrence with returning the plant to service.

The task force will be headed by B. A. Karrasch, who will work through J. T. Janis with the SMUD organization. Bruce will be assisted by J. J. Kelly.

The Licensing representative will be C. S. Banwarth.

J. M. Burnett will coordinate Component Engineering participation. P. A. Sherburne and J. R. Burris have started OTSG calculations.

J. A. Castanas will coordinate all activities of the Control and Instrumentation Section.

R. W. Winks and A. W. Brown are proceeding to the plant to assist with the investigation.

Others may be added to the task force as the work proceeds.

Top priority should be assigned to these activities in order to return the plant to service as soon as possible.

GMO:JCD:RF

- | | |
|---------------------|-----------------|
| cc: J. H. MacMillan | J. T. Janis |
| D. W. Montgomery | C. S. Banwarth |
| R. W. East | J. M. Burnett |
| J. C. Deddens | P. A. Sherburne |
| J. H. Taylor | J. R. Burris |
| J. D. Finney | J. A. Castanas |
| D. H. Ray | K. E. Suhrke |

[Handwritten signature and scribbles]

March 23, 1978

Mr. R. J. Rodriguez
Manager, Nuclear Operations
Sacramento Municipal Utility District
6201 S. Street
Sacramento, California

Subject: Rancho Seco Nuclear Generating Station - Unit No. 1
Evaluation of NSS Cooldown Transient

Dear Mr. Rodriguez:

B&W has reviewed the data provided by SMUD regarding the March 20, 1978 reactor trip and resulting cooldown transient, and have performed the following evaluations:

1. Evaluation of RCP seal performance data prior, during and subsequent to the transient.
2. Evaluation of transient conditions with respect to RCP's and CRDM's.
3. Evaluation of transient conditions with respect to fuel assemblies.
4. Evaluation of transient conditions with respect to RV, RC piping, pressurizer, and OTSG's.

As a result of these evaluations, B&W concurs with SMUD's intent to return Rancho Seco to power operation at a power level at or below 75% full power with the following recommendations:

1. The following maneuvering limits be applied for this plant startup:
 - a. The maximum rate of power increase below 20% full power shall be 10% per hour.
 - b. The maximum rate of power increase between 20% and 40% full power shall be 30% per hour.
 - c. Above 40% full power, escalation shall be limited to 3% per hour.
2. Increased surveillance of the loose parts monitoring system for at least a one week period.

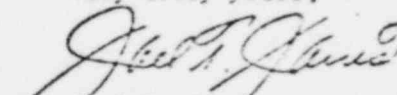
3. Performance of an operability check of on-line and redundant NMI instrumentation.
4. Establishment of a procedure for restoring NMI power in the event of a power loss and a commitment to establish by April 7, 1978 operator instructions for immediate action to limit NSS transient if NMI power cannot be immediately restored.
5. Surveillance of primary and secondary radiochemistry on a minimum of a daily basis for at least one week following startup.

B&W NPGD's concurrence with operation of Rancho Seco above 75% full power is contingent upon our QA evaluation of data input provided by SMUD and analyses performed by B&W Engineering, and you will be informed of completion of those activities and our concurrence in a timely manner.

In addition, B&W is performing an evaluation to determine the effects of the transient conditions on the Reactor Coolant System accumulative usage factor and will advise you of these results. As a further measure of verification of OTSG tube integrity, B&W requests additional inspection of the OTSG tubes during your next refueling outage. Those inspection recommendations will be forwarded to SMUD along with B&W's proposed refueling inspection plan prior to the outage.

If you have any questions or require additional information, please advise.

Very truly yours,


Joel T. Janis
Service Manager

JTJ/hh

cc: JJ Mattimoe
DG Raasch
RP Oubre'
JH Johnston

bcc: RM Ball
R. Berchin
JA Castanes
JC Daddens
EA Karaasch
ER Kane
JH MacMillan
GM Olds
JD Phinney

AL Dunlap
JH Taylor

Record Center NSS-11 T1.2

SECRET

SACRAMENTO MUNICIPAL UTILITY DISTRICT □ 691 S Street East 11112, Sacramento, California 95811 (916) 442-2711

RFO 78-28

March 28, 1978

Mr. Joel Jenis
Subcock & WALCOX
P. O. Box 1150
Lynchburg, Virginia 24505



Dear Joel:

Per discussion with you on Monday, March 27, you requested a chronology of our rise to power over the past weekend. Listed below is that chronology showing the rate of change of power level for the three rises in power.

Criticality @ 1607	3/24/78
0-20% 8%/hr	3/24/78
20-42.5% 6%/hr	3/24/78
Turbine/Reactor Trip at 2230	3/24/78

Criticality @ 0111	3/25/78
0-20% 6%/hr	3/25/78
20-40% 26%/hr	3/25/78
40-55% 3%/hr	3/25/78
55-62% essentially 0%/hr	
over a 20-hour period	
extending into 3/27/78	

Reactor reduced to 10^{-6}	
intermediate range amps	3/26/78
0-20% 10%/hr	3/26/78
20-50% 17%/hr	3/27/78
50-55% 3%/hr	3/27/78
55-72% 12%/hr	3/27/78

You will note that on the rise to power from 55 to 72% on Monday, March 27, the rate of change was at 12% per hour. This exceeded the recommended rate of change given to SMUD in your letter of March 23, 1978. I request SMU review this rate of change and respond to SMUD on any effect which it may have had. I would appreciate your response as soon as possible.

Respectfully,

R.P. Oubre

ENCLOSURE

TO

Gen. Meyer

FROM

DeMarr

CUST.

FILE NO. OR REF.

SUBJ.

SMUD SIR

DATE

3.21.78

BACKGROUND

FOLLOWING A TRIP THE REACTOR COOLANT TEMPERATURE AND PRESSURE DECREASED. GEN. MEYER HAS DEMANDS THE CHANGE IN TEMPERATURE AND PRESSURE AND ITS EFFECTS OF FUEL ROD CLADDING STRESS APPEAR INSIGNIFICANT AND CAN BE TREATED IN A MANNER SIMILAR TO TOLUOL DEPRIVATION. THERE IS A HIGH PROBABILITY THE SOME FUEL ASSEMBLIES LIFTED SINCE 4 PUMPS WERE OPERATING AT TEMPERATURE DOWN TO 400F FOR A PERIOD UP TO 10-12 MINUTES. THIS IS NOT A SERIOUS PROBLEM (SEE COMMENTS ACTION LIST RESOLUTION). HOWEVER, AS A CAUTIOUS PRECAUTION, STARTUP SHOULD BE IN ACCORDANCE WITH MMR'S ASSUMING THAT FUEL HAS BEEN HANDLED, I.E. 39% / 150% ABOVE 75% POWER WITH FUEL HOLD MODE AT POWER LEVEL CUT OFF.

COOLANT ACTIVITY SHOULD BE MONITORED CAREFULLY AND PARTICULAR ATTENTION SHOULD BE PAID TO LOOSE PARTS WITHIN THE SYSTEM.

MECHANICAL MANEUVERING RECOMMENDATIONS

RANCHO SECO

CONT: CT SMUD

REVISION 0

DATE JUNE 3, 1977

The following are the recommended maneuvering limits for SMUD, Cycle 2:

- 40-55A
1. The maximum rate of power increase below 20% full power shall be 10% per hour.
 2. Above 20% power, normal operating procedures (Tech. Spec Limits) will apply unless the reactor has operated at less than 20% power for more than 48 hours.
 3. If the power level has been below 20% full power for greater than forty-eight (48) hours, the maximum rate of power increase above 20% full power shall be 30% per hour with a five (5) hour hold at 20% full power below the power level cutoff and a five (5) hour hold at the power level cutoff. These holds can run concurrently with holds required by the Technical Specification.
 4. During the initial power escalation at cycle startup or immediately following a control rod interchange, the initial escalation above the 75% full power shall be limited to 3% per hour, with a five (5) hour hold at the power level cutoff. This hold can run concurrently with Technical Specification holds where applicable.
 5. With the exception of item 4 above, no restrictions are placed on required physics startup tests.

V. B. J. ...
FUEL ENGINEERING SECTION MANAGER

THE BAWCOCK & WILCOX COMPANY
POWER GENERATION GROUP

To	G. M. Olds, Manager, Nuclear Service	
From	B. A. Karrassi, Manager, Plant Integration	BDS 442.3
Cust.	177 - All	File No. or Ref.
Subj.	SMUD Cooldown Incident	Date March 29, 1978

This letter is for one customer and one subject only.

On March 20, 1978, SMUD experienced a loss of power to a substantial portion of the non-nuclear instrumentation (see details - Attachment 3). Although the Reactor Protection System (RPS) and Safety Features Actuation System (SFAS) functioned properly, SMUD still experienced the most severe thermal transient on any B&W plant to date. The subsequent investigation pointed out that additional guidance to our operating customers in the area of limiting potential events of this nature is warranted. Accordingly, the Engineering Department recommends that our operating plant customers be informed of this incident, and suggestions be made on how to minimize the plant thermal transient for loss of MWI and other similar events. Attachment 1, a sequence of events, and Attachment 2, a series of descriptive curves, are provided to assist you in preparation of a customer letter.

In addition, the following recommendations should be made to assure proper operator action for events of this nature:

1. Operators should be trained to recognize a loss of power to all or a majority of their MWI (indicators fail to mid-range, automatic or manual transfer to alternate instrument strings brings no response, etc.). The loss of power is emphasized here rather than the failure of any one instrument or control signal. These minor events are adequately covered in our present simulator course.
2. Given that the operator can determine that he has lost power to all or most of the MWI, he should know the location of the power supplies and power supply breakers and have a procedure available to regain power.
3. If the fault cannot be cleared (e.g., the breakers to the power supplies reopen), he should have a list of alternate instrumentation available to him. Some possibilities are:
 - a) SFAS panels
 - b) RPS panels
 - c) FCI
 - d) SRCI

- e) Remote shutdown panels
- f) Local gages
- g) Plant Computer

Note that each plant will be different in detail, but the list should be developed in advance and the operators thoroughly trained in its usage.

4. The above instrumentation sources should also be keyed to certain critical variables to help the operator select his priorities during the emergency condition. It is recognized that no procedure can cover all the possible combinations of non-nuclear instrumentation failures, however, if the operator knows he has an instrument problem (as opposed to a LOCA or steam line break, for example), he can limit the transient by controlling only a few variables. These are:

- a) Pressurizer level (via HPI or normal makeup pumps)
- b) RCS pressure (via pressurizer heaters, spray, E/M relief valve)
- c) Steam generator level (via feed flow, feedwater valves)
- d) Steam generator pressure (via turbine bypass system)

The pressurizer level and RCS pressure assure that the Reactor Coolant System is filled and the steam generator level and pressure assure adequate decay heat removal.

In our opinion, the preferred solution is to install safety grade steam generator level instrumentation, start auxiliary feedwater on a low level steam generator signal, and control steam generator level automatically, as is required on our IDOTSG plants. This may not be a practical jackfit solution for the operating plants, however, this solution should be suggested as a possible way to assure minimal lost capacity days due to equipment failure. Plant Integration will be happy to assist you in the preparation of the customer letter.

B. Kharasch

BAK:jl

cc: J. C. Deddens	J. A. Castanes
D. H. Roy	P. A. Sherburne
R. M. Ball	J. R. Burris
T. M. Schuler	R. W. Moore
C. W. Pryor	E. W. Winks
L. J. Stanek	J. T. Janis
K. E. Subrke	A. W. Brown
J. S. Tulenko	B. J. Shephard
J. W.
L. H. ...	C. L. ...
J. J. Kelly	H. S. Palme
J. D. Phinney	J. M. Burnette
D. F. Hallman	G. A. Meyer

Attachments

The operator was in the process of changing a 115Vt bulb in a turbine header pressure transfer switch when he dropped the bulb into the switch and shorted the switch to ground. A protective fuse in the circuit had no chance to blow before the power supply monitor circuit sensed the short on the bus and opened the breakers to the "Y" and "Z" power supplies. This resulted in the loss of approximately half the MMI signals ("X" power supply remained energized) and the ability to transfer from "bad" to "good" signals. It should be pointed out that the utility changed the original ESW design in the area of the power supplies. If the original ESW design were installed, the incident would have resulted in the loss of the ability to transfer (select) signals only. No active signal would have lost power.

SEQUENCE OF EVENTS - 04:25(AM) to 05:34(AM)

<u>TIME</u>	<u>EVENT</u>
4:25:35	- Lost NNI power supply cabinets 5,6, & 7 (B&W Channel "Y")
	- This caused a loss of T _{II} signals to the ICS. BTU limits ran back feedwater, resulting in a loss of feedwater (actual Rx power was 72%).
4:25:46	- Reactor trip on high pressure, turbine trip on interlock.
	- Pressurizer code relief setting was known to be low (=2225 psig). The electronic relief was isolated due to previous leakage problems. The data indicates primary pressure went =2400 psig =>code relief valve lifted.
4:26:17	- Operator starts HPI pump "B" to maintain prr. level.
4:28:23	- Operator stops HPI pump "B".
4:30	- OTSG "B" goes dry. Data indicates that "A" OTSG probably also went dry.
4:34:25	- RC pressure =1900 PSI. (low pressure trip set-point).
4:34	- Operator increases speed of a MFP and feeds "A" OTSG. This starts RCS on pressure and temperature decrease.
4:37:16	- SFAS actuation at 1600 psig. This starts HPI, LPI and initiates aux. feed. Aux. FW valves are opened to full open position. The system makes no automatic attempt to control steam generator water level.
4:40	- RC pressure at 1475 psig. It starts to recover from this point due to HPI. T _{ave} = 528°F.
4:43:56	- "A" HPI pump secured.
4:46:09	- LPI secured.
4:49:54	- "A" HPI initiated. From this point on, the operator started and stopped HPI pumps as necessary to maintain pressurized level.

TIMEEVENT

4:51:25

- Secured RCP-D ($T_{ave} = 435^{\circ}F$)
This reduced #RCP's to three

4:47:27

- OTSG "A" water level - 599.7"

Speculate that ~2 ft of tubes are not flooded
(at top) due to steam line arrangement.

5:00:00

- Hourly computer log printout
Steam temp $380^{\circ}F$ (OTSG "B")
Steam pressure 171 psig (OTSG "B")

Assuming $T_{ave} = T_{sat} \Rightarrow T_{ave} = 380^{\circ}F$

5:13:47

- Power restored to NNI cabinets 5, 6, & 7

$T_{ave} = 285^{\circ}F$

RCS Pressure = 2000 psig

Both OTSG full level ranges pegged high

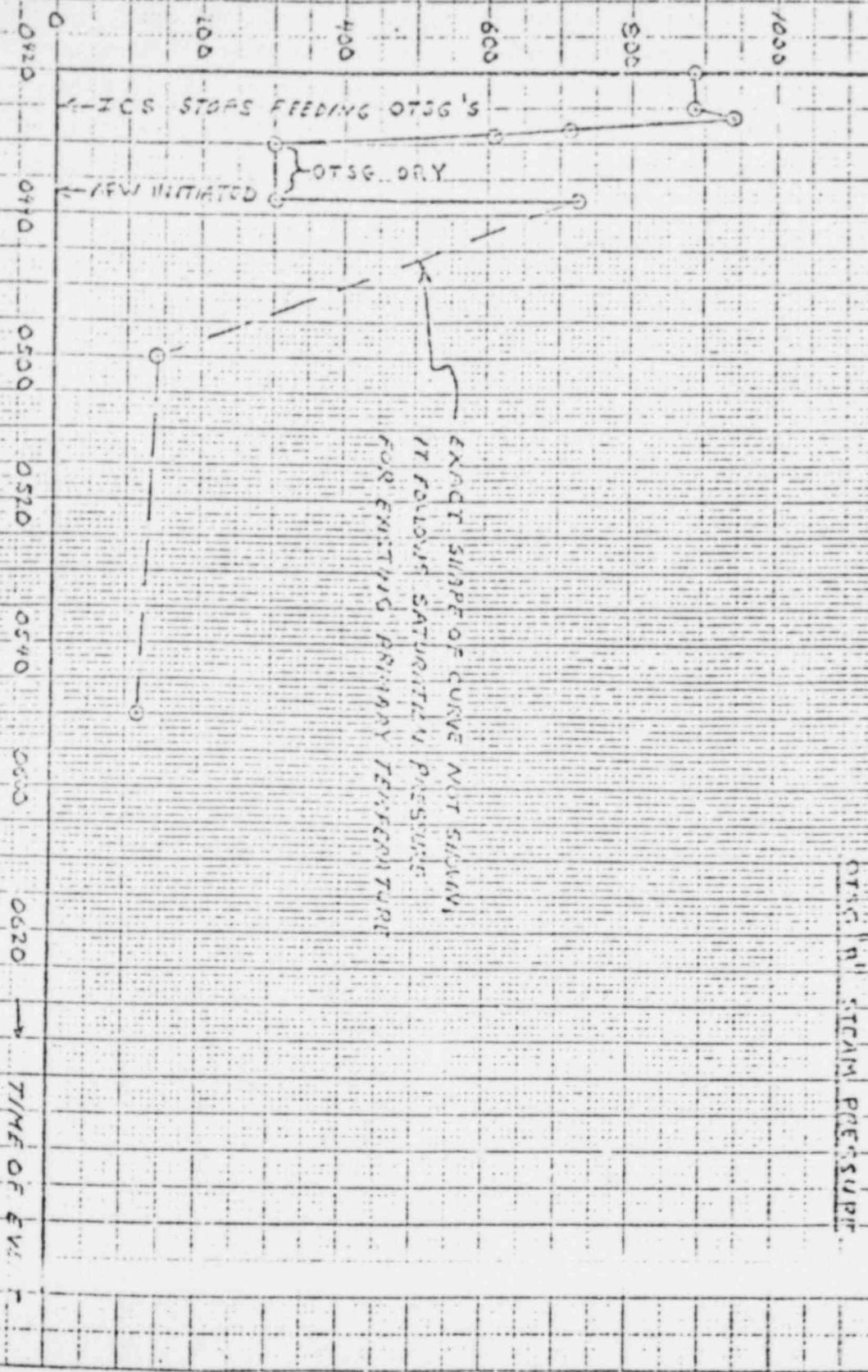
Operator begins to reduce RC pressure using
pzz. spaly.

Operator stops aux. FW pump.

SECONDARY PRESSURE, PSIG

NE 10 N 10 TO 7, PROJ. 27, N. 10 SEC. 10
 NE 10 N 10 TO 7, PROJ. 27, N. 10 SEC. 10

45 1470



46 1470

10 X 10 TO 1/4" (10 X 10) P. 3 10 THE CITY
MILWAUKEE WISCONSIN

