NRC FORM 195 (2-76)

U.S. NUCLEAR REGULATORY COMMISSION

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TO: Mr B C Rusche	FROM: Indiana & Michigan Pwr Co New York, NY G P Maloney	DOCKET NUMBER 50-315 FILE NUMBER DATE OF DOCUMENT 3-3-77 DATE RECEIVED 3-8-77
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DESCRIPTION

Ltr notarized 3-3-77....re our 1-10-77 ltr ....trans the following:

2p

REACTOR VESSEL OVERPRESSURIZATION
DISTRIBUTION PER G. ZECH 10-21-76

PLANT NAME: Cook #1

ENCLOSURE

Info concerning implementation option re installation of acceptable long-term improvements to mitigate overpressurization transients........

½inch

DO NOT REMOVE

ACKNOWLEDGED

SAFETY — **FOR ACTION/INFORMATION** 3-11-77 ehf

BRANCH CHIEF: (5)	Ziemann
LIC. ASST:	Diggs
PROJECT MANAGER:	Fletcher

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NRC FORM 195 (2-76)

INDIANA & MICHIGAN POWER COMPANY
P. O. BOX 18
BOWLING GREEN STATION
NEW YORK, N. Y. 10004

March 3, 1977

Donald C. Cook Nuclear Plant Unit No. 1
Docket No. 50-315
DPR No. 58

Mr. Benard C. Rusche, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Rusche:

This letter is in response to Mr. Dennis L. Ziemann's letter dated January 10, 1977 which requested that we identify the implementation option we have selected for the installation of acceptable long-term improvements to mitigate overpressurization transients. There were also staff positions and requests for additional information listed in enclosure 1 to Mr. Ziemann's letter. Our responses to the staff positions and requests for information are provided in Attachment 1.

Attachment 2 presents two sketches which represent the control function - logic diagram for the circuits to be installed at the Cook Nuclear Plant to mitigate the effects of an overpressurization transient. This control scheme will be duplicated on both existing RCS wide range pressure transmitters with each train actuating one Pressurizer Power Operated Relief Valve.

This control scheme will be the long-term solution for RCS overpressure protection. The selection and ordering of hardware (e.g. - Block-Unblock switch) is currently underway. The estimated delivery for the hardware is late September 1977. The remaining equipment that is needed is already available or can be fabricated on site.

The mitigating system we have designed incorporates controls for two (2) of our existing Pressurizer Power Operated Relief Valves and the control scheme of Attachment 2. The mitigating system will represent good engineering practice and will not adversely affect plant safety or introduce potential common mode failures that could both cause a pressure transient and disable the protection system.

2347

We will have all the necessary equipment available and be prepared to do the installation work by December 31, 1977. We will install the equipment at the first system depressurization subsequent to having all of the equipment available on site. We believe this to be the most logical approach since there is no need for the mitigating system until the reactor coolant system is depressurized from operating conditions.

Your August 13, 1976 letter required us to provide "an analysis of the RCS response to pressure transients." We expect that the analysis, when completed by Westinghouse, will show that our existing relief capacity is adequate to mitigate all the anticipated pressure transients. We will advise you as to the results of the analysis when they become available.

Very truly yours,

G. P. Maloney
Vice President

GPM:mam
Attachment

Sworn and subscribed to before me on this 3rd day of March 1977 in New York County, New York

Kathleen Barry
Notary Public
KATHLEEN BARRY
NOTARY PUBLIC, State of New York
No. 41-4606792
Qualified in Queens County
Certificate filed in New York County
Commission Expires March 30, 1977

cc: G. Charnoff
R. J. Vollen
R. C. Callen
P. W. Steketee
R. Walsh
R. S. Hunter
R. W. Jurgensen - Bridgman

Attachment 1

Response to Item 1:

Discussion of PWR overpressurization transients will be conducted as a part of the Licensed Operator Requalification Program. The program is conducted on a one shift per week basis. The required personnel attendance is normally all RO and SRO licensed operators on shift. For the purpose of overpressurization discussions, attendance will also be required of shift auxiliary equipment operators and C&I Technicians. The requalification program, year three, will start in March, 1977.

a) It is estimated that all plant operators will have participated in formal discussions of overpressure transients by May 15, 1977.

b) A formal discussion of overpressure transients at PWR's will be conducted as a part of the licensed operator requalification program. The discussions will cover in detail the following subjects:

1. The causes of past pressure transients that have occurred at other PWR facilities.

2. Plant conditions at the time of the pressure transients.

3. The mitigating action that could have been or was taken.

4. The preventive measures that could have been taken to avoid the event.

5. The steps that have been taken to prevent similar occurrences.

6. Cook Nuclear Plant similarities and differences with other PWR facilities and how these relate to overpressure events which occurred at these other facilities.

c) We have been able to collect details of eleven overpressure events at PWR facilities. These have been analyzed as to their possibility of occurrence at the Cook Nuclear Plant and the results are summarized in the following table.

Table of Overpressure Events
(Item 1(c) & (d))

Donald C. Cook Nuclear Plant

Plant Name	Cause of Event	Event Nature	Water-Solid Operation*	Steam-Bubble Operation**
Indian Point 3	Component Failure	RHR-Letdown Isolation	Credible	Not-Credible
Indian Point 2 (2 events)	Component Failure	Loss of Instrument Air	Not-Credible (Redundant Dryer Trains)	Not-Credible
St. Lucie 1	Defective Procedure	RCP Start	Credible	Not-Credible
D. C. Cook 1	Defective Procedure	Letdown Isolation	Credible	Not-Credible
Beaver Valley 1 76-13/3L	Personnel Error	SIS plus Letdown Isolation	Insufficient Information to draw a conclusion.***	
Point Beach 2	Defective Procedure	Letdown Isolation Deliberate Operator Action	Credible	Not-Credible
Beaver Valley 1 76-11/3L	Personnel Error	Letdown Isolation Electronic Instr. Spike	Insufficient Information to draw a conclusion.***	

*This column reflects the Cook Nuclear Plant prior to the revision of the three major operating procedures described in Item 2 response.

**This column reflects the Cook Nuclear Plant with its current operating procedures which provide for a steam cushion in the pressurizer.

***The description of the event in the available reports did not contain enough information to be able to determine the event credibility at the Cook Nuclear Plant.

[illegible]

Table cont'd....

Donald C. Cook Nuclear Plant

Plant Name	Cause of Event	Event Natures	Water-Solid Operation*	Steam-Bubble Operation**
St. Lucie 1	Component Failure	Hot Functional Test	Insufficient Information to draw a conclusion.***	
Zion 1	Operator Error	Letdown Isolation	Credible	Not-Credible
Turkey Point 3	Insufficient Information	RHR-Letdown Isolation	Credible	Not-Credible
Indian Point 2	Defective Procedure	RCP Start	Credible	Not-Credible

[illegible]

d) To reduce the likelihood of occurrence of the events which are listed as credible in the preceeding table, we have revised three major operating procedures to include non-water-solid operation of the reactor coolant system (RCS) and have installed an "operator alert" alarm. The description of the details of the procedural changes and the alarm function are provided in the response to Items 2 and 5 below.

[illegible]
[illegible]
[illegible]
[illegible]
[illegible]
[illegible]
[illegible]
[illegible]

Response to Item 2:

Three major operating procedures in which the plant would experience a water solid condition have been re-written. These procedures cover plant heatup, cooldown, and reactor coolant system fill and vent. As a result of the revisions to these operating procedures described below, it is not anticipated that the Cook Nuclear Plant will be operated or maintained in a water solid condition. Therefore the possibility of an overpressurization during these plant conditions has been greatly reduced.

a) Revised Plant Operating Procedures:

1. Filling and Venting Reactor Coolant System-OHP 4021.002.001

Prior to revising this procedure the pressurizer (PZR) and coolant loops were filled to a condition in which four air pockets remained in the steam generator tubes. The RCS pressure was then increased to about 400 psig using the charging pumps. While maintaining this pressure with the charging pumps and letdown through RHR, each reactor coolant pump (RCP) was bumped and the RCS was vented. After this venting the RCS was in a water solid condition using charging and letdown to control pressure and RHR system to control temperature. After the first vent the reactor coolant pumps were run again for 1 minute and the system vented. This cycle was repeated two (2) more times for 5 and 10 minute runs on the reactor coolant pumps. After completing the ten minute run/vent cycle the fill and vent procedure was considered complete and the plant remained in this condition until plant heatup, at which time a pressurizer bubble was drawn during heatup.

The revised procedure fills the RCS using the charging system to the condition where four air pockets remain in the steam generator tubes. At this point instead of pressurizing the system, bumping reactor coolant pumps and venting to a water solid condition, the pressurizer heaters are energized and a steam bubble is formed in the pressurizer. Pressurizer temperature is then increased to bring system pressure up to about 325 psig for reactor coolant pump operation. The

pressurizer steam space is put on sample recirculation to the volume control tank to remove air and N_2 from pressurizer. The reactor coolant pump run/vent cycle is now performed using steam bubble for pressure control. After the last RCP is run, the reactor coolant system pressure is reduced to 50-100 psig for final vent, to remove more N_2 from solution at lower pressure. After venting at 100 psig the fill and vent procedure is complete and the system will remain in this condition until heatup. Using this revised fill and vent method, which makes use of the air pockets in the four steam generator tubes, the RCS will never be in a water solid condition. Previous experience has shown that this procedure provides a "soft" system while drawing the steam bubble.

2. <u>Plant Heatup from Cold Shutdown to Hot Standby-OHP 4021.001.001</u>

This procedure was revised so that initial conditions for the heatup are that the RCS pressure is being maintained by a pressurizer steam bubble. The steam bubble has already been established during the fill and vent, therefore, the heatup procedure is not involved with water solid operation under any conditions.

3. <u>Plant Cooldown from Hot Standby to Cold Shutdown-OHP 4021.001.004</u>

Prior to revising the cooldown procedure the RCS was placed in a water solid condition during the last phase of plant cooldown. The first phase of plant cooldown remains the same, since water solid condition is not a concern during this part of the cooldown. The RCS temperature is reduced to about 150°F in the reactor vessel and loops, and 425°F in the pressurizer (which maintains system pressure at about 325 psig), by dumping steam to the condenser and placing RHR system in service at 425 psig/350°F. During the first phase of cooldown the following pertinent steps are taken at the indicated system conditions.

i) System Pressure < 1000 psig:

Accumulator isolation valves IMO-110, 120, 130, 140 are closed and breakers are racked out and tagged.

ii) System temperature < 350°F:

SI pump breakers are racked out and tagged.

Note: A revision is now being made to add the following valves to condition (ii), (system temp. < 350°F).

Close the following valves, rack out and tag supply breakers for each:

SI Pump discharge Valves	-	ICM-260
		ICM-265
Cold Leg SI Valves	-	IMO-51
		IMO-52
		IMO-53
		IMO-54

The steps listed above will completely disable the SIS system.

A major revision was made to the last phase of the cooldown procedure to cool the pressurizer down and depressurize the system using Auxiliary spray with a maximum level in pressurizer of 85%, instead of filling the pressurizer to water solid condition for cooldown. The revised procedure completes the plant cooldown as follows:

Plant Conditions:

Loop and vessel temperature - 150°F

Pressurizer temperature - 420°F

Pressurizer pressure - 325 psig

1 reactor coolant pump running, RHR in service.

First the running RCP is secured and the PZR heaters are de-energized. Using Auxiliary spray the PZR level is slowly increased, cooling the PZR and depressurizing

the system. The PZR level is allowed to increase to 85% maximum level and recirculated using Auxiliary and letdown through RHR until a steam space temperature of 250°F to 275°F is reached. When this condition is established, the PZR level is reduced by increasing letdown through RHR system and the charging system is secured. Draining of the RCS to necessary level for maintenance/and/or refueling can be started at this point and the cooldown is considered complete.

For all other conditions which do not require opening of the RCS a low pressure steam bubble will be maintained during the cold shutdown condition.

b) With the revision of the above three procedures, it is not anticipated that the RCS will be operated or maintained in a water solid condition at any time during further operations.

c) It is not anticipated that the need will exist to operate the RCS in a water solid condition.

Response to Item 3:

The safety injection system (SIS) pumps will be de-energized during cold shutdown operation as required. However, our current Technical Specifications (3.5.2 & 3.5.3) require that surveillance testing of the SIS be performed during cold shutdown and hence the SIS pumps must be operated during this mode. We will not be able to de-energize the SIS pumps during this testing period but they will be de-energized at all other times during cold shutdown.

a) Figures 1 and 2 (attached) show the SIS flow paths into the RCS.

b) Figures 3 and 4 show the head-flow characteristics of both (North & South) SIS Pumps.

c) The valves and pumps to be closed and de-energized are shown on Figure 1 by being circled with a dotted line. Items are identified as North and South ("N" and "S") pumps, valves V-9 (ICM-265) & V-10 (ICM-260) and valves IMO-51, IMO-52, IMO-53, IMO-54.

d) The present revision of the plant cooldown procedure requires the accumulator valves to be closed and the breakers for each to be racked out and tagged. The SIS pumps are also de-energized and the supply breakers racked out and tagged.

   A temporary sheet has added the following valves to the cooldown procedure:

   Cold leg injection valves IMO-51, 52, 53, and 54

   SI pump discharge valves ICM-265 and 260 (V-9 & V-10 respectively)

   The change requires these valves to be closed and the breakers for each racked out and tagged.

e) All equipment identified in 3(c) will remain closed and/or de-energized during cold shutdown operation, except when technical specification surveillance testing is being performed. It is felt that by closing and de-energizing

the cold leg injection valves IMO-51, 52, 53, 54 that SIS injection would be prevented in the event of an inadvertent operation of the SIS during cold shutdown condition.

f) Safety Injection Pump "S" is supplied from circuit breaker T11A1 located in the Bus A and Bus B 4kV switchgear room. Safety injection pump "N" is supplied from circuit breaker T11D5 located in the Bus C and Bus D 4kV switchgear room. Both switchgear rooms are on elevation 609'. These circuit breakers are normally operated from the control room. Disconnecting these circuit breakers, which makes them inoperable, is done at the 4 kV switchgear rooms.

The north safety injection pump shutoff valve, ICM-260, is supplied from MCC-1-ABV-D and the south safety injection pump shutoff valve, ICM-265, is supplied from MCC-1-ABV-A. Both motor control centers (MCC) are located on elevation 587'. The valve motor operators are normally controlled from the control room. Opening the circuit breakers to the valve controller, which will disable the valve operator, is done at the motor control center.

g) Disconnecting the circuit breakers for the safety injection pump motors will turn off both the red and green breaker position indicating lights and the white breaker closing status light. The status monitor lights are still operable and will indicate a failure of the safety injection pump motors to start if a safety injection is initiated.

The red and green valve position indicating lights will both go out on both valves. The status monitoring lights will indicate the valves closed if a safety injection is initiated.

h) The plant cooldown procedure will provide the administrative control necessary to insure that each designated valve and pump is de-energized and/or closed at the correct time during cooldown. When the equipment is placed in the indicated position the step in the procedure will be signed off. The equipment will also be tagged out in

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

accordance with the plant tagging procedure, which provides another administrative control. When the equipment is tagged, it can not be energized or its position changed without first obtaining authorization from the shift operating engineer.

When the plant is returned to power, proper equipment alignment and operability is administratively controlled by the technical specification surveillance signoff sheets. These data sheets are attached to the plant heatup and startup procedures and require that all equipment associated with a mode change are OPERABLE prior to that mode change. All equipment listed above will be checked and verified OPERABLE in this manner.

Each shift operating engineer is responsible for plant operation and control of plant equipment on his shift. It is the shift operating engineer's responsibility to insure that designated equipment is closed, de-energized and tagged out in accordance with plant procedures.

i) The accumulator isolation valves will be closed when the RCS pressure is at 1000 psig or less.

The supply breakers for the motor operated accumulator isolation valves are located on Motor Control Centers 1-EZC-A,B,C, and D. These MCC's are located in the electrical bay mezzanine. The switches for the isolation valves are located on the control room panel SIS. The supply breakers are opened and closed locally at the MCC's.

Disabling the valve motor operators will cause the red and green valve position lights to go out. The annunciator "accumulator valve not fully open" will be operable when operating above P11 interlock and the status monitor lamps will be operable.

j) The overall impact on plant operation, to lower accumulator nitrogen pressure for each cooldown, would be the possible increase in time to either the cooldown or heatup cycle due to depressurization or pressurization of accumulators.

Also the increased usage and availability of N_2 could effect plant operation.

There is also a remote chance that the N_2 would have to be vented to the gas system due to high gaseous activity which would overload the waste gas system.

We believe that closing the isolation valves, de-energizing and tagging the breakers supply power to the motor operated valves is sufficient protection.

[illegible]
[illegible]

[illegible]
[illegible]
[illegible]

[illegible]
[illegible]
[illegible]

<u>Response to Item 4</u>:

a) The charging system and the safety injection system are routinely tested while in cold shutdown.

b) For extra measures taken see the response to Item 3.

<u>Response to Item 5</u>:

A high pressure alarm ("operator alert") used during low RCS temperature operation is presently installed at the Cook Nuclear Plant.

a) The "operator alert" alarm was installed prior to the December 23, 1976 refueling outage.

b) The system modifications are shown in Figure 5 (attached).

c) The alarm setpoint, mode of annunciation and sensor are shown in Figure 5.

d) It is not anticipated that the Cook Nuclear Plant will be operated in a water-solid condition.

Response to Item 6:

a) RHR Design pressure is 600 psig.

b) Figure 2 identifies the RHR isolation valves relevant to overpressure incidents by means of two hexagons. The valves are Copes-Volcan, double disk, motor operated gate valves.

c) RHR - Interlocks, setpoints, alarms:
RHR pump suction valves: IMO-128 and ICM-129.

1. IMO-128 is interlocked with RCS wide range pressure channel NPS-122. When pressure drops below 425 psi, a permissive signal allows the operator to manually open this valve. If the pressure increases above 600 psi, this valve is closed automatically.

   An RCS high pressure alarm in the control room is energized whenever pressure is above 600 psi, as sensed on either wide range pressure channel, NPS-121 or NPS-122. An RCS low pressure alarm is energized whenever both pressure channels are below 425 psi.

2. ICM-129 is interlocked with RCS wide range pressure channel NPS-121. Operation and setpoints are as above.

d) The nominal stroke time for these valves is 35 seconds.

e) The setpoint and capacity of the relevant RHR safety valve is 450 psig and 900 gpm, respectively. This safety valve is designated on Figure 2 by an asterisk(*).

f) The other alarms and setpoints associated with RHR not discussed in (c) above are as follows:

Pressure alarms; IPA-310, IPA-320

RHR pump discharge pressure high alarms annunciate in the control room above 590 psi.

Response to Item 7:

We feel that adequate provisions have beenpprovided by means of the revised procedures described in the response to Item 2.

a) Using our revised procedures for RCS fill and vent, plant cooldown and heatup, the RCS will no longer be in a water solid condition during any of these operations. During RCS fill and vent, air bubbles are maintained in the steam generators while a pressurizer steam bubble is established. The reactor coolant pumps (RCP's) are not operated prior to PZR steam bubble formation. System pressure is increased for RCP operation by increasing PZR temperature to about 440°F. Pump run and vent cycles are conducted at these conditions. With this procedure the RCP's are not operated in a water solid condition.

The RCS is not placed in a water solid condition at anytime during cooldown. The PZR is cooled down and RCS depressurized using Auxiliary spray and slowly filling the PZR to a maximum level of 85% and recirculating until conditions are met for draining the PZR/RCS.

Either air or N_2 blanket is placed on system at start of drain down. If the RCS is not to be opened during a unit shutdown, a low pressure (i.e. 50-75 psig) bubble will be maintained in the PZR during shutdown precluding a water solid condition.

Using these revised procedures, RCP operation during water solid conditions is no longer a question.

b) A PZR steam bubble is established for all conditions listed in Item 7(a) <u>prior</u> to the start of the first reactor coolant pump.

c) Listed below are the limits associated with system temperature before the first RCP can be started. This question is not applicable to the Cook Plant due to the response of (b) above.

1. RCS below 150°F without seal water or with RCP seal leakage rate at 5 gpm or less and about 35 gpm of component cooling water at inlet temperature of about 80°F is flowing through thermal barrier.

2. A pressure differential of at least 275 psi is available across the #1 controlled leakage seal. This is met if reactor coolant pressure is above 375 psig., and temperature of 350°F or less.

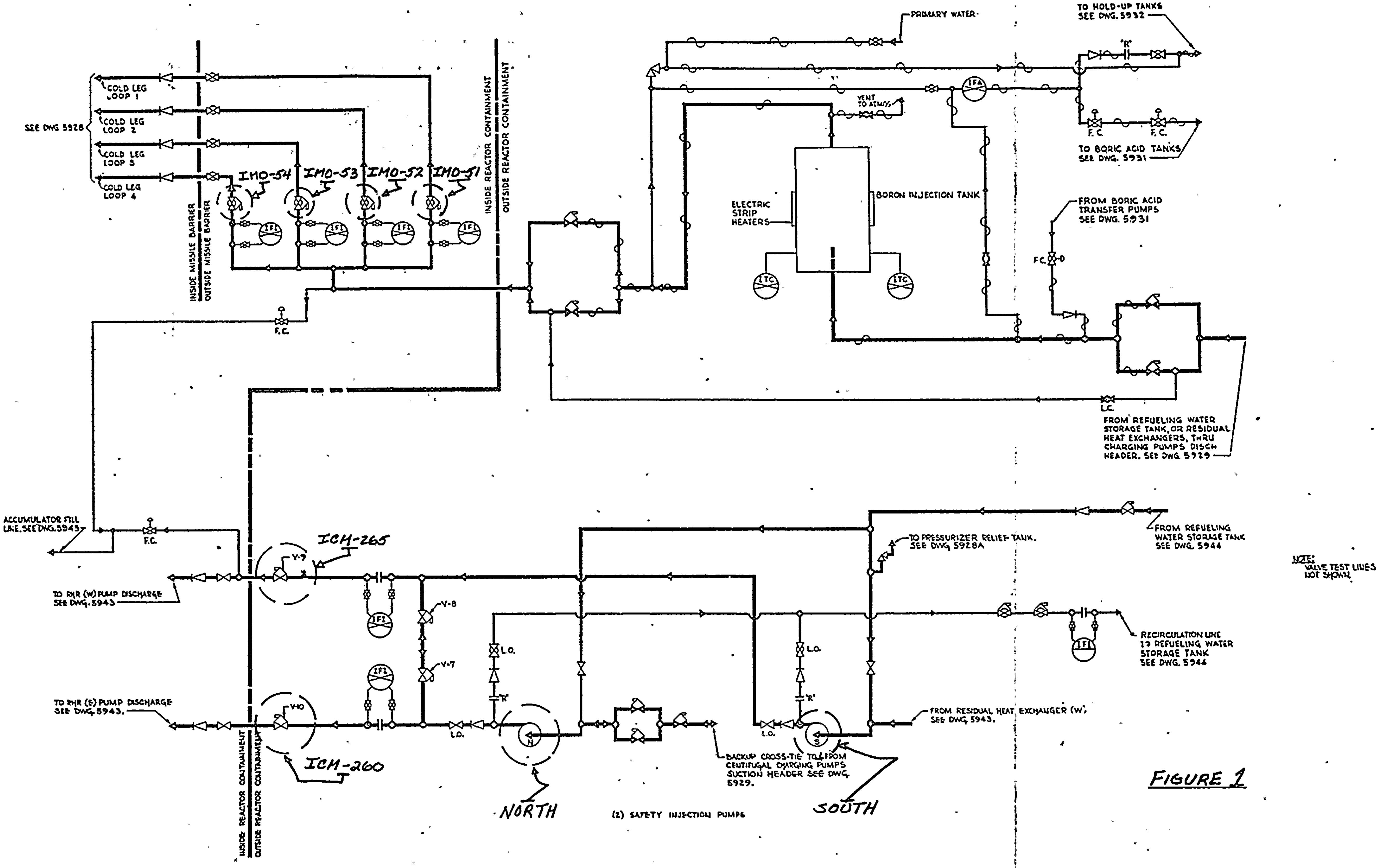
d) RCS temperature profile instruments:

1. Individual hot and cold leg reactor coolant temperature are recorded over the range of 0-700°F.

2. RHR pump discharge temperature is recorded over a range of 0-400°F.

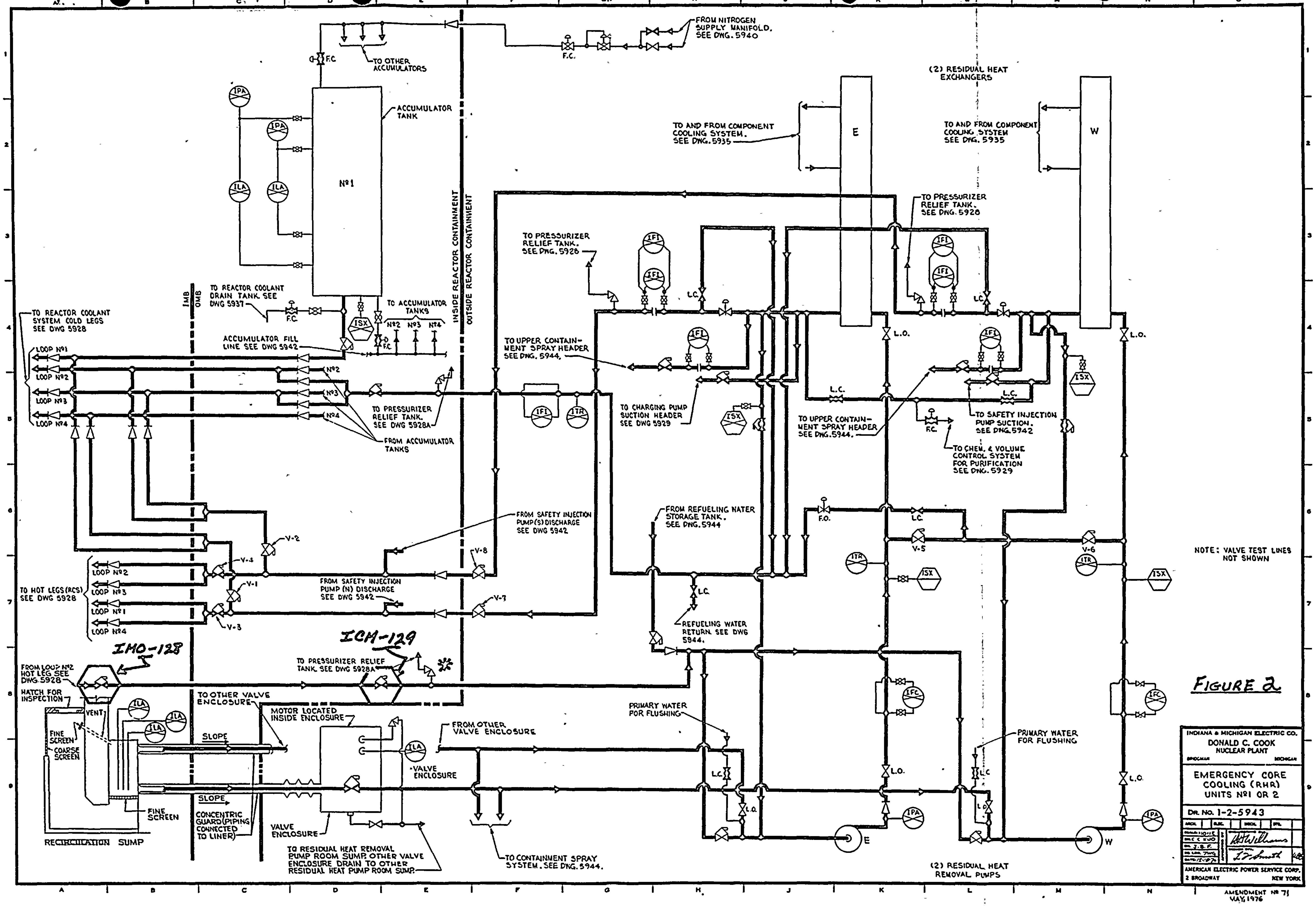
e) With steam bubble operation it is not necessary to establish isothermal conditions in the RCS for RCP operation.

f) Pressure spikes are not a problem on RCP starts when a PZR steam bubble has been established. Several uses of the revised procedures as described in the response to Item 2 have shown this to be the case.



COLD LEG LOOP 1
COLD LEG LOOP 2
COLD LEG LOOP 3
COLD LEG LOOP 4
SEE DWG 5928
IMO-54
IMO-53
IMO-52
IMO-51
INSIDE MISSILE BARRIER
OUTSIDE MISSILE BARRIER
INSIDE REACTOR CONTAINMENT
OUTSIDE REACTOR CONTAINMENT
F.C.
PRIMARY WATER
VENT TO ATMOS
TO HOLD-UP TANKS SEE DWG. 5932
TO BORIC ACID TANKS SEE DWG. 5931
ELECTRIC STRIP HEATERS
BORON INJECTION TANK
FROM BORIC ACID TRANSFER PUMPS SEE DWG. 5931
L.C.
FROM REFUELING WATER STORAGE TANK, OR RESIDUAL HEAT EXCHANGERS, THRU CHARGING PUMPS DISCH HEADER. SEE DWG 5929
ACCUMULATOR FILL LINE. SEE DWG. 5943
ICM-265
V-9
V-8
V-7
V-10
TO RHR (W) PUMP DISCHARGE SEE DWG. 5943
TO RHR (E) PUMP DISCHARGE SEE DWG. 5943.
ICM-260
L.O.
TO PRESSURIZER RELIEF TANK. SEE DWG 5928A
FROM REFUELING WATER STORAGE TANK SEE DWG. 5944
RECIRCULATION LINE TO REFUELING WATER STORAGE TANK SEE DWG. 5944
FROM RESIDUAL HEAT EXCHANGER (W) SEE DWG 5943.
BACKUP CROSS-TIE TO/FROM CENTRIFUGAL CHARGING PUMPS SUCTION HEADER SEE DWG. 5929.
NORTH
SOUTH
(2) SAFETY INJECTION PUMPS
NOTE: VALVE TEST LINES NOT SHOWN.

FIGURE 1



FROM NITROGEN SUPPLY MANIFOLD. SEE DWG. 5940
TO OTHER ACCUMULATORS
ACCUMULATOR TANK
Nº1
INSIDE REACTOR CONTAINMENT
OUTSIDE REACTOR CONTAINMENT
IMB
OMB
TO REACTOR COOLANT DRAIN TANK SEE DWG 5937
TO ACCUMULATOR TANKS
Nº2 Nº3 Nº4
TO REACTOR COOLANT SYSTEM COLD LEGS SEE DWG 5928
LOOP Nº1
LOOP Nº2
LOOP Nº3
LOOP Nº4
ACCUMULATOR FILL LINE SEE DWG 5942
TO PRESSURIZER RELIEF TANK. SEE DWG 5928A
FROM ACCUMULATOR TANKS
TO UPPER CONTAINMENT SPRAY HEADER SEE DWG. 5944.
TO PRESSURIZER RELIEF TANK. SEE DWG. 5926
TO CHARGING PUMP SUCTION HEADER SEE DWG 5929
TO AND FROM COMPONENT COOLING SYSTEM. SEE DWG. 5935
(2) RESIDUAL HEAT EXCHANGERS
TO PRESSURIZER RELIEF TANK. SEE DWG. 5926
TO SAFETY INJECTION PUMP SUCTION. SEE DWG. 5942
TO CHEM. & VOLUME CONTROL SYSTEM FOR PURIFICATION SEE DWG. 5929
FROM SAFETY INJECTION PUMP(S) DISCHARGE SEE DWG 5942
FROM REFUELING WATER STORAGE TANK. SEE DWG. 5944
NOTE: VALVE TEST LINES NOT SHOWN
TO HOT LEGS (RCS) SEE DWG 5928
FROM SAFETY INJECTION PUMP (N) DISCHARGE SEE DWG 5942
REFUELING WATER RETURN SEE DWG 5944.
IMO-128
ICM-129
FROM LOOP Nº2 HOT LEG SEE DWG. 5928
HATCH FOR INSPECTION
TO PRESSURIZER RELIEF TANK. SEE DWG 5928A
TO OTHER VALVE ENCLOSURE
MOTOR LOCATED INSIDE ENCLOSURE
FROM OTHER VALVE ENCLOSURE
VALVE ENCLOSURE
VENT
FINE SCREEN
COARSE SCREEN
SLOPE
CONCENTRIC GUARD (PIPING CONNECTED TO LINER)
RECIRCULATION SUMP
TO RESIDUAL HEAT REMOVAL PUMP ROOM SUMP. OTHER VALVE ENCLOSURE DRAIN TO OTHER RESIDUAL HEAT PUMP ROOM SUMP.
TO CONTAINMENT SPRAY SYSTEM. SEE DWG. 5944.
PRIMARY WATER FOR FLUSHING
(2) RESIDUAL HEAT REMOVAL PUMPS
FIGURE 2
INDIANA & MICHIGAN ELECTRIC CO.
DONALD C. COOK NUCLEAR PLANT
EMERGENCY CORE COOLING (RHR) UNITS Nº1 OR 2
DR. NO. 1-2-5943
AMERICAN ELECTRIC POWER SERVICE CORP. 2 BROADWAY NEW YORK
AMENDMENT Nº 71 MAY, 1976

CONTRACTOR ______

CUSTOMER *WESTINGHOUSE APD*

SPIN NO. *AER SIAPSI-01* P.O. *54-E-795-038 SUPP*

IMPELLER PATTERN *M-6116* *M-6117/18*

MAXIMUM DIAMETER *8 5/8* *8 5/8 / 8 5/8*

RATED DIAMETER *8 5/8* *8 5/8 / 8 5/8*

MINIMUM DIAMETER *8 1/4* *8 1/4 / 8 1/4*

TEST PERFORMANCE CURVE NO. *34554C*

SIZE *2 1/2* TYPE *JTCH* STAGES *10*

R.P.M. *3570* DATE *2-19-70*

PUMP NUMBER *45491 Unit 1-Nor*

PERFORMANCE ALSO APPLIES TO PUMP

NUMBER ______

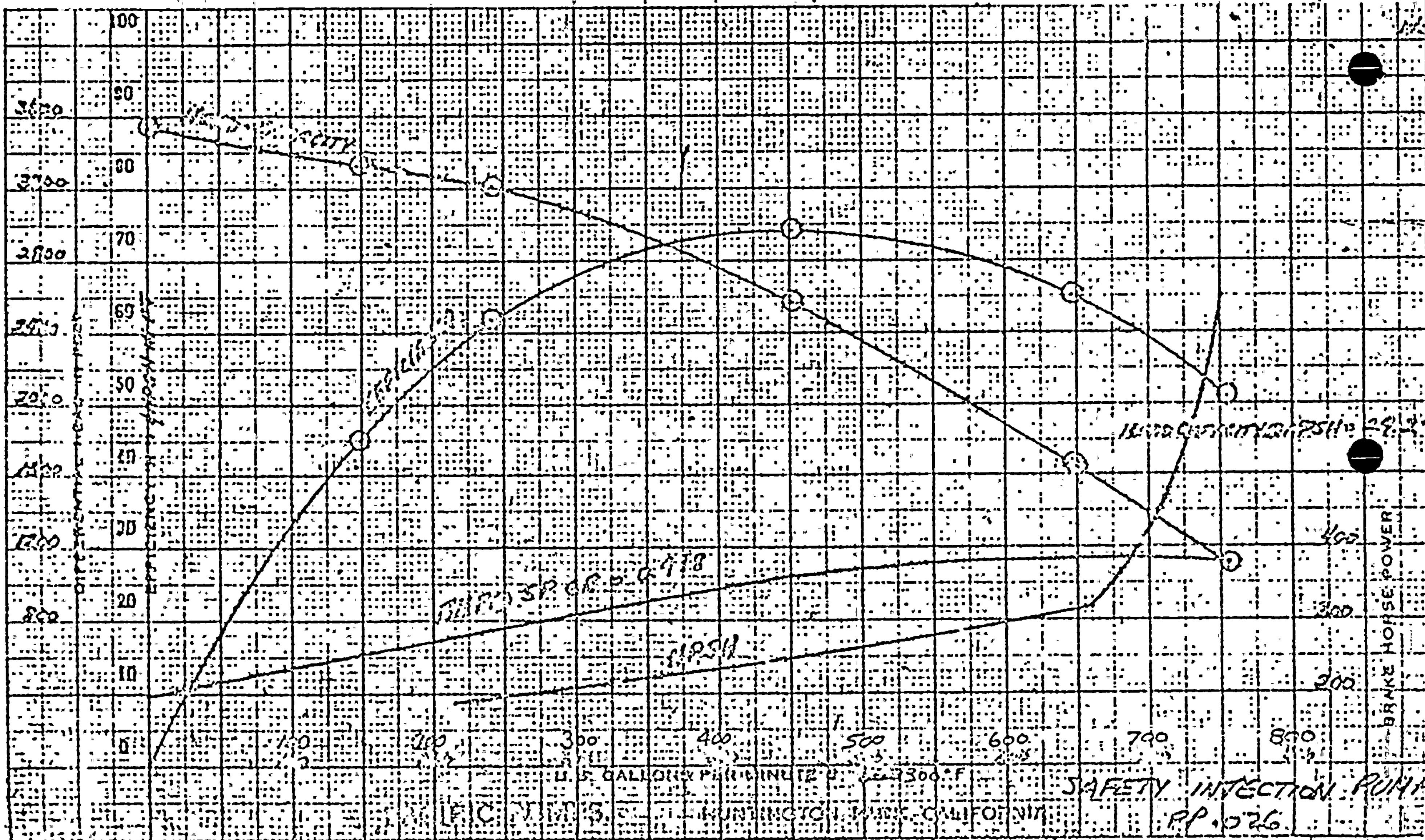


FIGURE 3

CONTRACTOR ____________________

CUSTOMER WESTINGHOUSE APD

SPM NO AMP-SIAPSI-02 P D 54-E-705-03 B SUPP

IMPELLER PATTERN M-6146 M-6147/48

MAXIMUM DIAMETER 8 5/8 8 5/8 8 5/8

RATED DIAMETER 8 5/8 8 5/8 8 5/8

MINIMUM DIAMETER 8 1/4 8 1/4 8 1/4

TEST PERFORMANCE CURVE NO. 54534 F

SIZE 2 1/2 TYPE JTCH STAGES 10

R P.M. 3570 DATE 1-16-70

PUMP NUMBER 15474 Unit 1-South

PERFORMANCE ALSO APPLIES TO PUMP

NUMBER ____________________

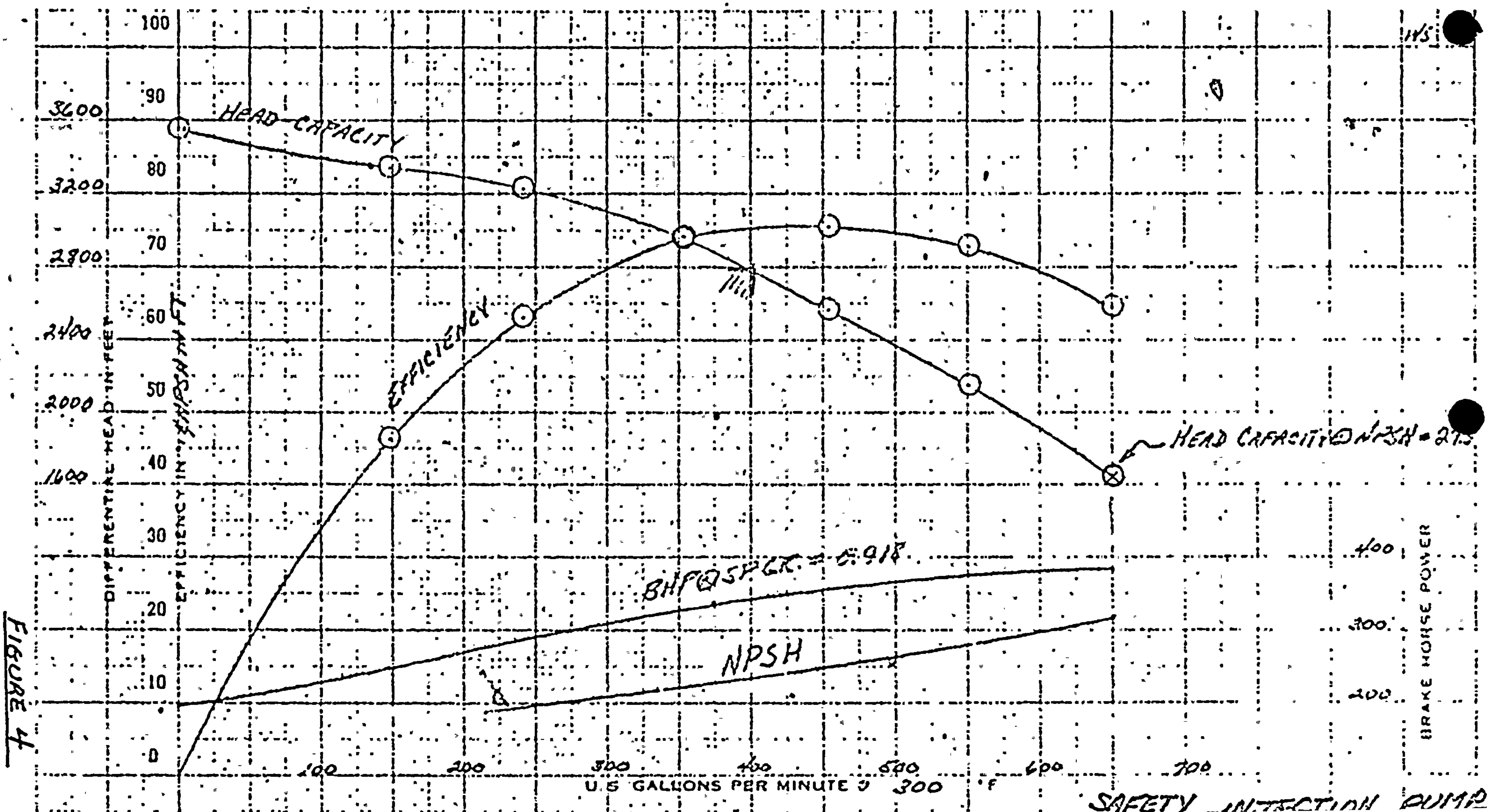


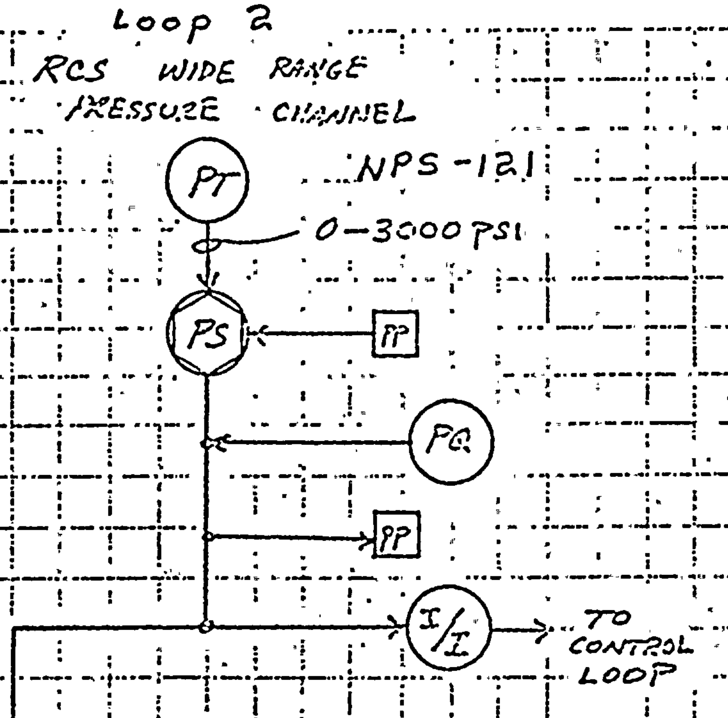
FIGURE 4

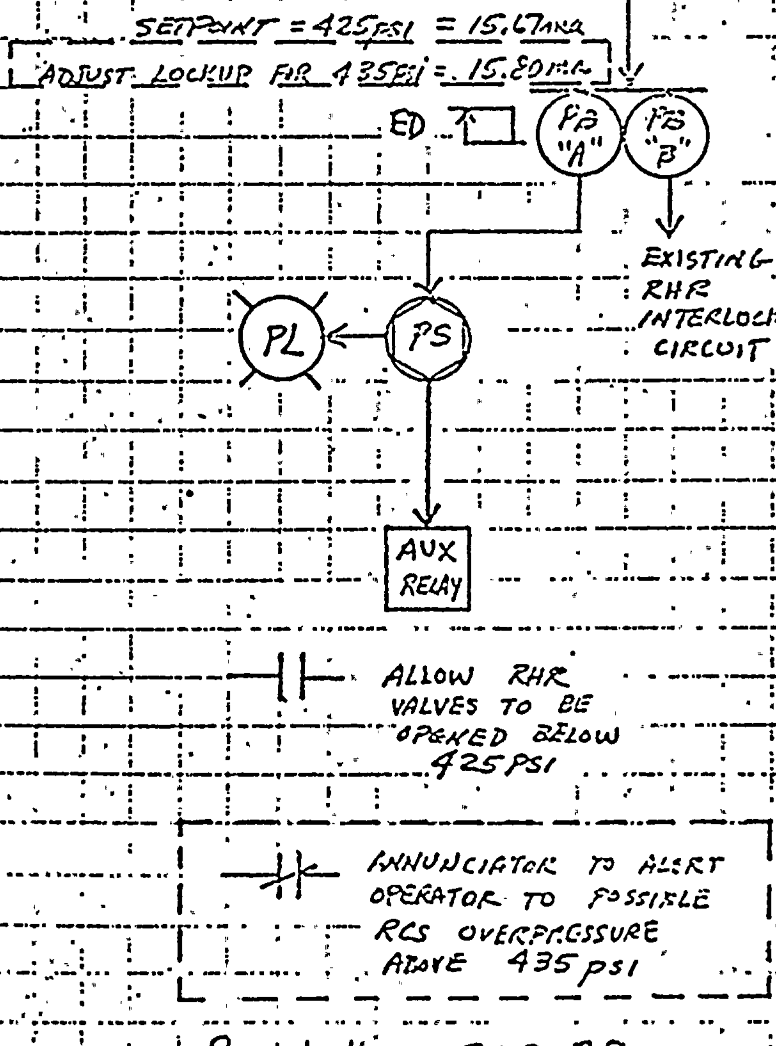
SAFETY INJECTION PUMP

AMERICAN ELECTRIC POWER SERVICE CORP.
2 BROADWAY
NEW YORK

DATE 11/5/76 BY D.E. GOLINO CK. ____
COMPANY I&M POWER G.O. ____
PLANT D.C. COOK

SUBJECT RCS OVERPRESSURE ALARM - RFC-DC-12-1332





Panel drop PZR-39

FIGURE 5

NOTES:

1) PRESENT RHR INTERLOCK SETPOI[NT] WILL NOT BE AFFECTED.

2) ALARM SHOULD BE MOUNTED IN THE "U" SECTION OF THE CONTROL ROOM.

3) THE "LOCKUP" ADJUSTMENT ON THE RHR PERMISSIVE INTRLK WILL BE NARROWED FROM 1% TO 0.325%. THIS IS THE POINT AT WHICH THE BISTABLE RESETS ON INCREASING PRESSURE.

4) ALARM WILL BE ENERGIZED DURING NORMAL OPERATION

5) LOOP ACCURACY 1/2% = 15 PSI

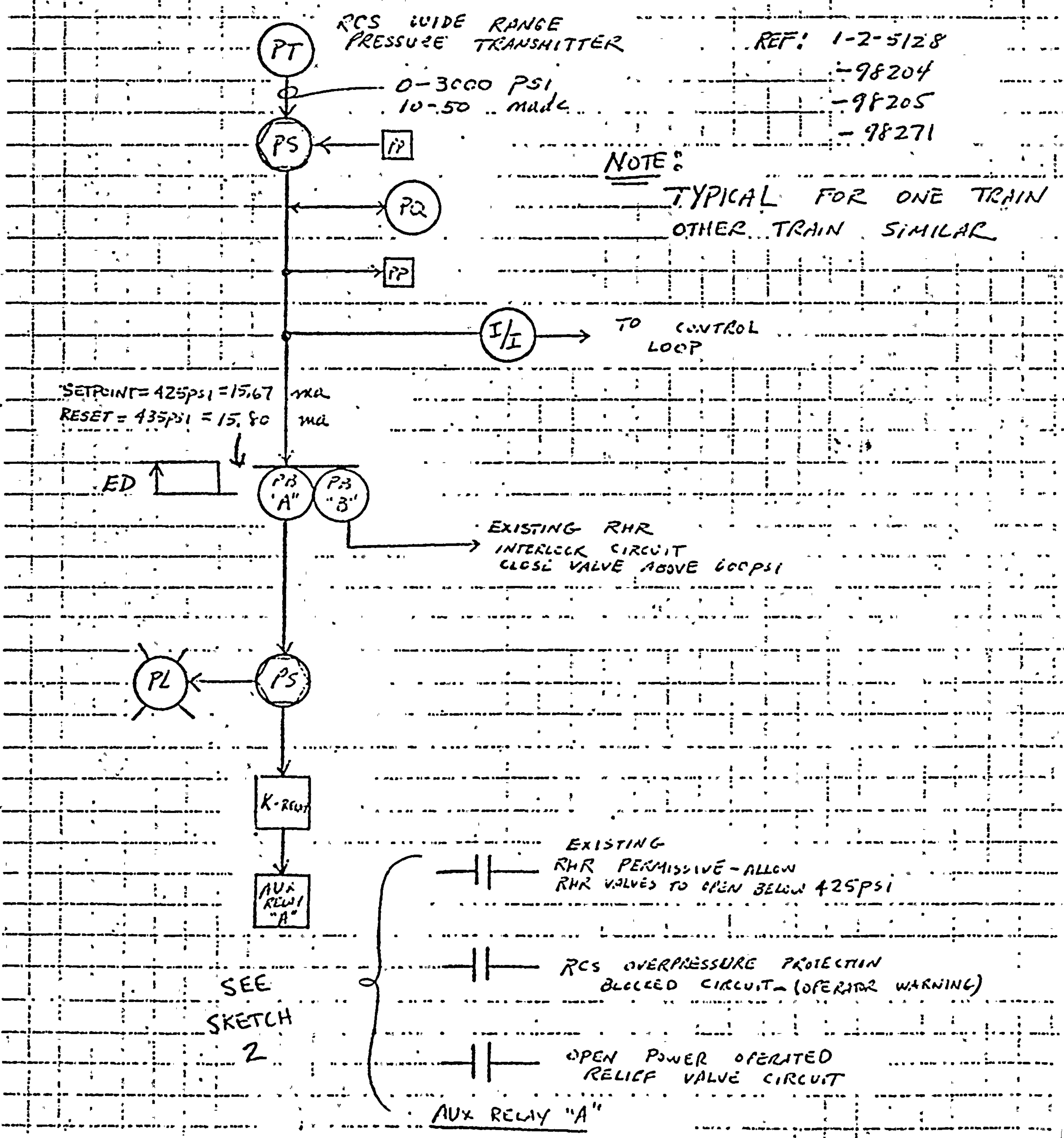
FORM GE-8

ENGINEERING D[illegible]T.
AMERICAN ELECTRIC POWER SERVICE CORP.
2 BROADWAY
NEW YORK

SHEET 1 OF 2
DATE 1/[illegible]/77 BY D.F. GILLINCK
COMPANY I & M POWER G.O.
PLANT D.C. COOK

Attachment 2

SUBJECT RCS OVERPRESSURIZATION
PROPOSAL - SKETCH 1

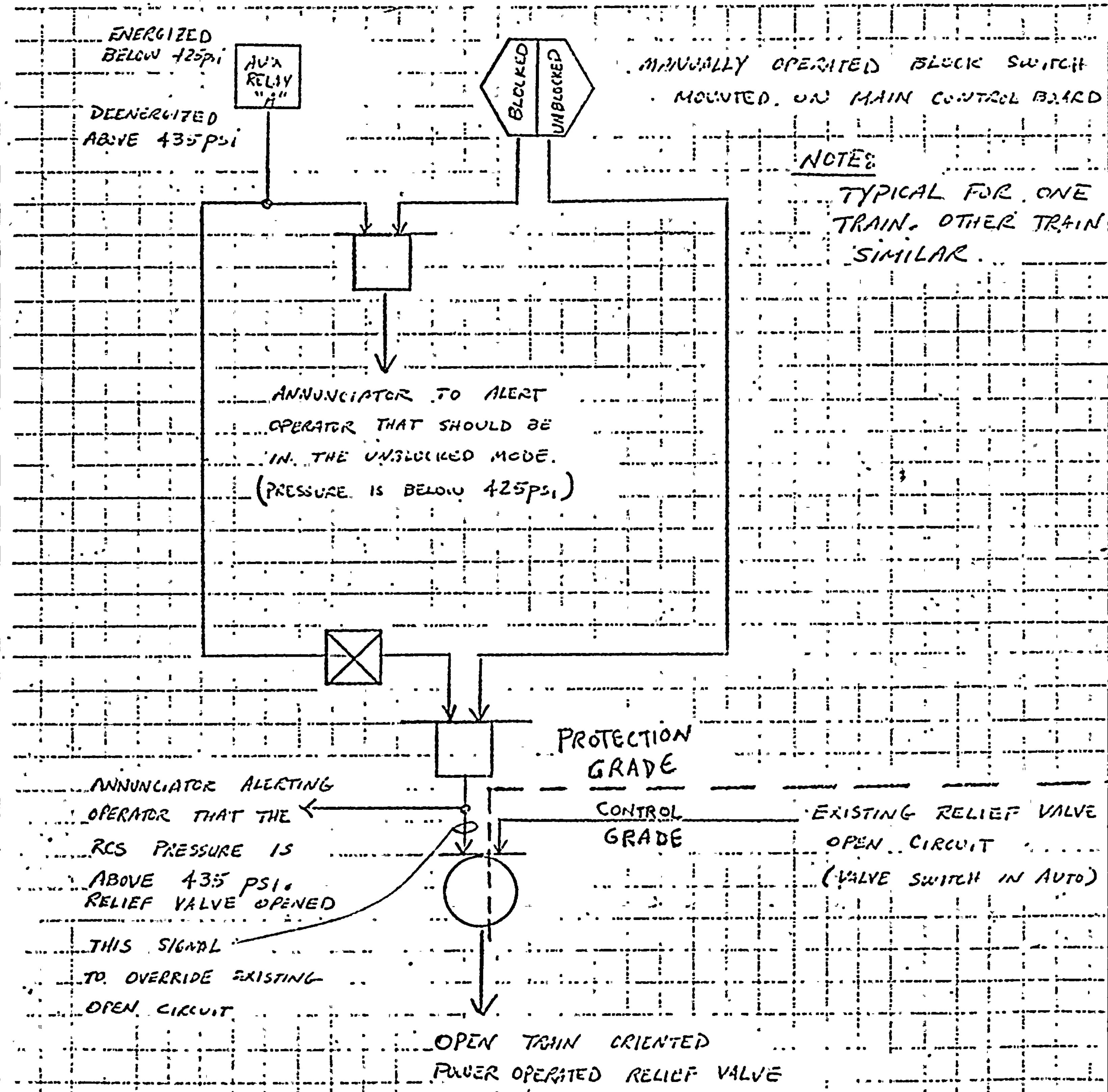


FORM GE-8

ENGINEERING DEPT.
AMERICAN ELECTRIC POWER SERVICE CORP.
2 BROADWAY
NEW YORK

SHEET 2 OF 2
DATE 1/ 77 BY P.E. GOLINOCK CK. ____
COMPANY I & M POWER G.O. ____
PLANT D.C. COOK

SUBJECT RCS OVERPRESSURIZATION - SKETCH 2



[illegible]