123 Main Street
White Plains, New York 1(
914 681.6200

NewYorkPower Authority

J. Phillip Bayne
Executive Vice President
Nuclear Generation

November 7, 1983
IPN-83-91

U. S. Nuclear Regulatory Commission
Office of Nuclear Reactor Regulation
Washington, D. C. 20555

Attention: Darrell G. Eisenhut, Director
Division of Licensing

Subject: Indian Point 3 Nuclear Power Plant
Docket No. 50-286
Required Actions Based on Generic Implications of Salem ATWS Events (Generic Letter 83-28, dated July 8, 1983)

Dear Sir:

Attachment A to this letter provides an item by item response to the information requested in the subject Generic Letter.

As indicated in the enclosed attachment, the Authority is a member of both the Westinghouse Owner's Group (W-OG) and a Nuclear Utility Task Action Committee (NUTAC) sponsored by INPO. Accordingly, for some items our schedules are consistent with these industry groups projected completion schedules.

Should you or your staff have any questions regarding this matter, please contact Mr. P. Kokolakis of my staff.

G.M. Wilverding
for J. P. Bayne
Executive Vice President
Nuclear Generation

Enc: Attachment A

cc: attached

State of New York
County of Westchester

Subscribed and Sworn to before me
this 7th day of November, 1983

Doreen Pisco

Notary Public
DOREEN PISCO
Notary Public, State of New York
No. 4737373
Qualified in Westchester County
Term Expires March 30, 19[illegible]

A001
1/1

cc: Resident Inspector's Office
Indian Point Unit 3
U. S. Nuclear Regulatory Commission
P.O. Box 66
Buchanan, New York 10511

ATTACHMENT A

IP3 Status Regarding Required Actions Based on Generic Implications of Salem ATWS Events

1.1 Post Trip Review (Program Description and Procedure)

1.1.1 The criteria for determining the acceptability of restart.

Response: The decision to restart the unit following a trip or an unscheduled or unexplained power reduction is contingent on a determination of the circumstance, analysis of the cause and a determination that operation can be resumed and proceed safely as required by Administrative Procedure AP-21 (Conduct of Operations).

10CFR 50.36 (c)(1)(i), referenced in Technical Specification Section 6.7, requires that the reactor be shutdown in the event a safety limit is violated and that operation shall not be resumed until authorized by the Commission.

1.1.2 The responsibilities and authorities of personnel who will perform the review and analysis of these events.

Response: The Shift Supervisor has the responsibility and authority to perform the review and analysis of plant transients including an ATWS event in accordance with Administrative Procedures, AP-21 (Conduct of Operations).

The Shift Technical Advisor is responsible for performing engineering evaluations of plant operations and procedures and in general provides technical support to the Shift Supervisor.

Plant Emergency Procedure PEP-EL-1 provides for the plant to be restarted with the permission of the Operations Superintendent or his designated alternate in the event of a blackout during a system-wide emergency, providing that Technical Specification Section 3.7 is satisfied.

The decision to restart the plant is based on the Shift Supervisor's recommendation and discussions with the Operations Superintendent, Superintendent of Power and the Resident Manager if necessary.

The duties and responsibilities of the Shift Supervisor, the Shift Technical Advisor, the Operations Superintendent, Superintendent of Power and the Resident Manager are described below.

Shift Supervisor (SS)

a. The Shift Supervisor has the responsibility and authority to command the control room and to supervise the operation of the plant during normal operation and accident conditions.

b. The Operations Superintendent, the Superintendent of Power or other licensed personnel designated in the Emergency Plan may elect to assume overall responsibility to supervise the operation of the plant. All commands and directions pertaining to the plant operation will be transmitted through the Shift Supervisor.

c. The Shift Supervisor has the responsibility and authority, unless specifically relieved by higher authority, to determine the circumstance, analyze the cause, and determine that operation can proceed safely before the reactor is returned to power after a trip or an unscheduled or unexplained power reduction.

d. The Shift Supervisor has the responsibility and authority to provide direction for returning the reactor to power following a trip or an unscheduled or unexplained power reduction.

e. The Shift Supervisor has the responsibility to ensure that (1) the operability of redundant nuclear safety-related equipment is verified, by test or inspection, prior to the removal of any safety-related equipment from service for maintenance and (2) the operability of safety-related equipment is verified when it is returned to service following maintenance and/or testing.

As indicated in our response to item 1.1.7, the Authority is currently developing a systematic safety assessment post trip review procedure which will be implemented prior to startup from the ongoing outage. The present practice used in the event of an unscheduled reactor shutdown is outlined below.

The Shift Supervisor assumes command of the control room and performs the following tasks:

a. Verifies that the plant is brought to a stable condition.

b. Reviews all automatic actuations, as delineated in the Plant Emergency Procedures (PEP), to ascertain that equipment functioned in accord with their design criteria and initiates corrective measures as required.

c. Notifies the Operations Superintendent, the Superintendent of Power, the Resident Manager (the Operations Superintendent may elect to notify the Resident Manager himself) Corporate Public Relations, the Nuclear Regulatory Commission (NRC) and the NRC Resident Inspector of the occurrence.

d. Prepares a Significant Occurrence Report (SOR).

e. Determines the cause of the reactor shutdown and takes appropriate corrective action if necessary.

f. Provides periodic updates to the NRC as necessary.

g. Determines that all Technical Specification and procedural requirements are met.

h. Confers, as necessary, with the Operations Superintendent, Superintendent of Power and the Resident Manager regarding corrective actions taken and for a determination that the plant can be restarted and that operations can proceed safely.

Shift Technical Advisor (STA)

The STA is responsible to the Shift Supervisor for those functions related to plant safety and to the Technical Services Superintendent for his administrative and engineering duties. The STA provides technical support to the Shift Supervisor and advises him on actions to terminate or mitigate the consequences of an accident condition. In addition, he has the responsibility to perform engineering evaluations of plant operations and procedures and to advise the Shift Supervisor of any problems relating to plant operations. During accident conditions he is to act solely in an advisory capacity to the Shift Supervisor and will not have responsibility for manipulation of controls.

During normal operation the STA will be utilized to assess operational information, investigate unusual or abnormal events and make recommendations to appropriate management level personnel with regard to his findings.

The STA may advise the Shift Supervisor on Technical Specification implications as requested by the Shift Supervisor.

Operations Superintendent

The Operations Superintendent directs plant operations through the Shift Supervisors. The Operations Superintendent ensures plant operation within the requirements of the Technical Specifications and all other regulatory requirements in accordance with approved procedures. He is responsible for day to day operations of the plant and timely submittal of work requests. The Operations Superintendent reviews routine operating data to assure safe plant operation. The Operations Superintendent is responsible for assuring that personnel under his supervision are aware of and comply with the appropriate requirements of the Quality Assurance Program.

The Operations Superintendent provides the liaison between the shift and plant staff organizations for safe, efficient operation and maintenance. The Operations Superintendent is a member of the Plant Operating Review Committee and reports to the Superintendent of Power. The Operations Superintendent may relieve the Shift Supervisor during normal operation or accident conditions.

Superintendent of Power

The Superintendent of Power is responsible to the Resident Manager for the functional operation of the plant. The Superintendent of Power supervises plant operations and staff organization activities and assures safe, efficient plant operation within the requirements of regulatory agencies and Technical Specifications and in accordance with the Quality Assurance Program.

The Superintendent of Power is responsible for assuring that qualified manpower and required equipment and/or facilities are provided to operate, test and maintain the station.

The Superintendent of Power is Vice Chairman of the Plant Operating Review Committee. During the absence of the Resident Manager, the Superintendent of Power will assume his responsibilities and functions, including Chairman of the Plant Operating Review Committee.

Resident Manager

The Resident Manager has overall responsibility for safe, efficient and dependable operation of the plant consistent with providing continuing protection to the environment. He is empowered to implement all administrative controls in conformance with applicable regulatory requirements regarding the facility and has responsibility for the coordination of all station functions through the Superintendent of Power, Plant Superintendents and other key personnel. He is responsible for the selection and training of personnel, administrative implementation of plant security and relations with official bodies within the scope of the Authority practice. The Resident Manager ensures the on site administration and adherence to the Quality Assurance Program.

During periods when the Resident Manager is unavailable, the Superintendent of Power will assume his responsiblities. In the event both are unavailable, he may delegate this responsibility to other supervisory personnel.

The Resident Manager reports directly to the Executive Vice President-Nuclear Generation. The Resident Manager is Chairman of the Plant Operating Review Committee and a non-voting member of the Safety Review Committee.

Technical Specifications require that the Chairman of the Safety Review Committee (SRC) and the Executive Vice President-Nuclear Generation be notified of any safety limit violation within 24 hours. A safety limit violation report shall be prepared by the Plant Operating Review Committee (PORC) and submitted to the Chairman of the SRC and the Executive Vice President-Nuclear Generation by the Resident Manager. Reports of reviews encompassed by Technical Specification Section 6.5.2.7 and delineated in Section 1.1.6 of this report shall be prepared, approved and forwarded to the Executive Vice President-Nuclear Generation within 14 days following completion of the review in addition to minutes of all SRC meetings. (Technical Specification Section 6.5.2.10).

1.1.3 The necessary qualification and training for the responsible personnel.

Response: Plant Technical Specification (Administrative Controls, Section 6) requires that members of plant staff identified in section 1.1.2 (above) meet or exceed the minimum qualifications of ANSI N18.1 - 1971 for comparable positions with the exception of the Shift Technical Advisor who shall be qualified as described below. The Technical Specification further requires that a retraining and replacement training program for the plant staff shall be maintained under the direction of the training coordinator and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10CFR Part 55.

Shift Supervisor

In accord with ANSI N18.1 - 1971 and Administrative Procedure AP-1 (Plant Staff Organization), the Shift Supervisor is required to hold a high school diploma or equivalent; four years of responsible power plant experience, of which a minimum of one year shall be nuclear power plant experience. A maximum of two of the remaining three years of power plant experience may be fulfilled by academic or related technical training on a one-for-one time basis. Training of Shift Supervisors is that needed to obtain and maintain a senior reactor operator license as required by 10CFR55.

Shift Technical Advisor

In accord with the Plant's Technical Specification (Administrative Controls), the Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design and response and analysis of the plant for transients and accidents.

Operations Superintendent

ANSI N18.1-1971 requires that the Operations Superintendent shall have a minimum of eight years of responsible power plant experience of which a minimum of three years shall be nuclear power plant experience. A maximum of two years of the remaining five years of power plant experience may be fulfilled by satisfactory completion of academic or related technical training on a one-for-one time basis. At the time of initial core loading or appointment to the active position, the Operations Superintendent shall hold a Senior Reactor Operator's License. If the Superintendent of Power is licensed at the senior reactor operator level, he may fulfill this requirement until a newly appointed Operations Superintendent obtains his license.

Resident Manager and Superintendent of Power

ANSI N18.1-1971 requires that the resident manager shall have ten years of responsible power plant experience, of which a minimum of three years shall be nuclear power plant experience. A maximum of four years of the remaining seven years of experience may be fulfilled by academic training on a one-for-one time basis. To be acceptable, this academic training shall be in an engineering or scientific field generally associated with power production. The resident manager or superintendent of power shall have acquired the experience and training normally required for examination by the NRC for a Senior Reactor Operator's License whether or not the examination is taken.

If the Superintendent of Power meets the nuclear power plant experience and NRC examination requirements established for the resident manager, the requirements of the plant manager may be reduced, such that only one of his ten years of experience need to be eligible for NRC examination.

Either the Resident Manager or the Superintendent of Power should have a recognized baccalaureate or higher degree in an engineering or scientific field generally associated with power production.

1.1.4 The source of plant information necessary to conduct the review and analysis.

Response: A listing of plant information necessary to conduct the review and analysis is provided in Table 1.

1.1.5 The methods and criteria for comparing the event information with known or expected plant behavior.

Response: Plant Emergency Procedures (PEP) and Off Normal Operating Procedures (ONOP) delineates automatic actuations and required operator actions for various abnormal events. These procedures require the operator to verify that certain automatic actuations have occurred as specified in the procedures.

Administrative Procedure AP-1 further requires the Operations Superintendent and Shift Supervisors to assure that plant operations comply with limits of the Technical Specification.

1.1.6 The criteria for determining the need for independent assesment of an event and guidelines on the preservation of physical evidence (both hardware and software) to support independent analysis of the event.

Response: Current plant procedures and Technical Specifications require independent assessments by an engineering task force or Plant Operating Review Committee (PORC) in the following situations: (1) seismic events which exceed the 2% response curve for the containment structure (PEP-S-1), (2) safety limit and Technical Specifiction violations and (3) reportable occurrences requiring 24 hour notification to the NRC (Technical Specifications Sections 6.5.1.6 and 6.6.1 and 6.7.1)

Technical Specifications Section 6.5.2.7 further requires that a review be performed by the Safety Review Committee (SRC) in the following circumstances:

a. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in 10 CFR50.59.

b. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

c. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

d. Events requiring 24 hours written notification to the Commission.

e. All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.

f. Reports and meeting minutes of the Plant Operating Review Committee.

Administrative Procedure AP-18 (Table A) and Technical Specification (Section 6.10) requires that plant records and logs of facility operation and all reportable occurrences to the Commission be maintained for a minimum of five years and that Significant Occurrence Reports be maintained for a period of six years.

In addition, in accordance with AP-18, Section 5.3.3, records of significant value in determining the cause of an accident or malfunction of an item are maintained for the lifetime of the plant.

1.1.7 The systematic safety assessment procedures complied from items 1.1.1 through 1.1.6 which are to be used in conducting the evaluation.

Response: The Authority believes that current plant procedures, specifications and practices address the intent of the criteria outlined in items 1.1.1 through 1.1.6 of Generic Letter 83-28. A systematic safety assessment post trip review procedure is presently being developed and it will be implemented prior to returning to service (expected January 1984) from the ongoing electrical generator outage.

1.2 Post-Trip Review-Data and Information Capability

1.2.1 Capability for assessing sequence of events (on-off indicators)

1.2.1.1 Brief description of equipment.

Response: See Table 1.

1.2.1.2 Parameters monitored.

Response: See Table 2.

1.2.1.3 Time discrimination between events.

Response:

Plant Computer	Time Discrimination
Alarm Type Write Up	one minute
Trend Type Write Up	30 sec.-99 mins.
Sequence of Events Records	one cycle (on 60 Hz basis)

1.2.1.4 Format for displaying data and information

Response: See Table 2.

1.2.1.5 Capability for retention of data and information

Response: Hard copy of plant computer printout are retained in plant files in accordance with Regulatory Guide 1.88 Rev. 2, 1976. (See also response to item 1.1.6 above.)

1.2.1.6 Power Sources.

Response: See Table 3.

1.2.2 Capability for assessing the time history of analog variable needed to determine the cause of unscheduled reactor shutdowns, and the functioning of safety-related equipment.

1.2.2.1 Brief description of equipment.

Response: See Table 1.

1.2.2.2. Parameters monitored, sampling rate, and basis for selecting parameters and sampling rate.

Response: Analog variables listed on the log typewriter (Table 2, sheets 1a through 1c) are collected hourly. These are readings designated by Westinghouse, and concurred with by the Authority to be representative of normal operating and shutdown conditions of the plant.

The analog variables listed on the Post Trip Review (Table 2, Sheet 2) were determined by Westinghouse to provide the most useful information concerning a plant trip. The value of each variable is listed approximately every 8 seconds for a period beginning two minutes prior to the trip and ending three minutes after the trip. These variables include various plant temperatures, pressures, powers, levels and flow rates.

In addition, variables may be logged or the Trend Typewriter for periods from 30 seconds to 99 minutes on one of six blocks. Each block may contain up to 18 variables which are selected by the operator. The data is recorded on printout paper, and the trending continue until stopped by the operator.

See Table 2.

1.2.2.3 Duration of time history (minutes before trip and minutes after trip).

Response:

Plant Computer/Record	Time Before Trip	Time After Trip
Alarm Type Write Up	continuous	continuous
Report Type Write Up (post trip review program)	120 secs.	180 secs.
Log Type Write Up	continuous (hourly)	continuous (hourly)
Transient Bus Recorder	0 secs.	continuous
Strip Chart Recorder	continuous	continuous

1.2.2.4 Format for displaying data including scale (readability) of time histories.

Response: The Log Typewriter displays time in military units (1200, 1300, etc.). The Alarm Typewriter displays time the same way. The Post Trip Review displays the time of trip in hours, minutes, seconds (HMSS) format and displays times for analog variable readings in MMSS.T format (i.e. minutes, seconds, tenths of seconds).

See Table 2 for format samples.

1.2.2.5 Capability for retention of data, information, and physical evidence (both hardware and software).

Response: Hardcopy of plant computer outputs and recorder graphs are retained in plant files in accord with Regulatory Guide 1.88 Rev. 2, 1976. (See also response to item 1.1.6 above.)

1.2.2.6 Power Sources.

Response: See Table 3.

1.2.3 Other data and information provided to assess the cause of unscheduled reactor shutdowns.

Response: None.

1.2.4 Schedule for any planned changes to existing data and information capability.

Response: Emergency Response Facilities Data Acquisition and Display System (EFDADS), which includes the SPDS, is scheduled to be operational at the end of cycle 4/5 refueling outage as indicated in our letter IPN-83-26, dated 4/18/83 and transmitted in response to Supplement 1 to NUREG-0737 (Post-TMI Requirements).

TABLE 1

Equipment used to diagnose the cause of unscheduled reactor shutdowns

(A) <u>On - Off Indications</u>
Alarm Type Print-out
Trend Type Print-out
Sequence of Events Records

(B) <u>Analog Variables</u>
Alarm Type Print-out
Report Type Print-out (Post Trip Review Program)
Log Type Print-out
Transient Bus Recorders
Strip Chart Recorders

(C) <u>Indicating Instruments & Devices</u>
First Out Reactor & Turbine Trip Annunciators
Annunciators (In General)
Instrument Bus Transient Recorders
Indicating Lights (Rod Bottom)
Switch Positions and Disagreement (amber) Lights
Indicators (Meters, etc.)
Valve Position Indicators
Valve Position Lights
Breaker Position Switch
Breakers
Counters

(D) <u>Relays</u>
Protective Relays (Control Room, Plant and Buchanan)
Lock out relays
Relays in General

(E) <u>Human Factors</u>
Observations of components by Plant Personnel
Control Room Logs

TABLE 2 (SHEET 1 of 3)

INDIAN POINT STATION UNIT NO. 3

(Sheet 1a)

PERIODIC LOG SHEET

DATE ______________ PAGE ________ OF ________

TIME	U N I T																				
	CYCLE BURNUP		NET GENERATION	AUXILIARY CONSUMPTION		GROSS GENERATION	THERMAL OUTPUT		AUXILIARY LOAD		NET LOAD	UNIT NET EFFIC	NUCLEAR POWER	FLUX THERM POWER MIS-MTCH	REACTOR COOLANT TEMP			TREF	STEAM GEN.		1ST STG TURBINE PRESS
				UNIT	STATION		STEAM GEN.	REACTOR	UNIT	STATION					HOT LEG	COLD LEG	AVE T AVE		FW TEMP	AVE OUTLET PRESS	
	MWD/MTM	EFPD	M W H						M W			%			°F					PSIG	
	LAST CALC.		SINCE LAST 1 HOUR						LAST 1 MINUTE AVERAGES												

TIME	UNIT			CHEMICAL AND VOLUME CONTROL									PRESSURIZER					RADIATION MONITORS							
	CONTAIN-MENT PRESS	COMPONENT COOLING SUCTION TEMP		CHARG-ING FLOW	LET DOWN FLOW	V C TANK TEMP	CHARG-ING TEMP	REGEN HX LETDOWN OUT TEMP	NON-REG HX LETDN OUT TEMP	NON-REG HX LETDN OUT PRESS	BORIC ACID INJECTED	PRIMARY MAKE-UP WATER ADDED	PRESS	LEVEL	LEVEL SET POINT	LEV SP DEV FROM COMP SP	SURGE LINE TEMP	COND AIR EJECTOR	STM GEN BLOW DOWN	COMPONENT COOLING		PLANT VENT		CONTAINMENT AIR	
		NO. 31	NO. 32																	NO. 31	NO. 32	PARTICULATE	GAS	PARTICUL	GAS
	PSIG	°F		G P M		°F				PSIG	GAL		PSIG	%			°F	DEKA-COUNTS 1 MINUTE							
	LAST 1 MINUTE AVERAGES										SINCE LAST HOUR		LAST 1 MINUTE AVERAGES					LAST 1 MINUTE AVERAGES							

R E A C T O R C O R E

POWER RANGE DETECTOR SIGNALS

MAX CALCULATED VALUES

(Sheet 1b)

TIME	REACTOR CORE																										
	POWER RANGE DETECTOR SIGNALS																					MAX ASSEM REL POWER		MAX T/C TEMP °F	CALCULATED VALUES		
	UPPER DETECTOR				LOWER DETECTOR				PERCENT POWER				AXIAL OFFSET				AVE AXIAL TILT %	RADIAL TILT									
	41 VOLTS	42 VOLTS	43 VOLTS	44 VOLTS	41 VOLTS	42 VOLTS	43 VOLTS	44 VOLTS	41%	42%	43%	44%	41%	42%	43%	44%		41%	42%	43%	44%	T/C ID	F ΔH		COOLANT BORON PPM	XENON DEFECT %	SH DO
	LAST 1 MINUTE AVERAGES																					LAST CALC.		INST.	LAST CALC.		

TIME	REACTOR CORE																	ΔT SETPOINTS									
	T/C QUADRANT POWER TILT								ROD BANK POSITIONS									OVERTEMPERATURE ΔT					OVERPOWER ΔT				
	$\frac{1\,|\,2}{4\,|\,3}$				6 = NORTH				CONTROL				SHUTDOWN				PART LENGTH	RC LOOP				COM-PUTED	RC LOOP				COM-PUTED
	1	2	3	4	5	6	7	8	A	B	C	D	A	B	C	D		31	32	33	34		31	32	33	34	
	BASED ON T/C AVERAGED OVER 1 MINUTE								STEPS									PERCENT									

TIME	LOOP 31													LOOP 32												
	REACTOR COOLANT			STEAM GENERATOR					RC PUMP MTR BRG TEMP				LOWER SEAL BRG TEMP °F	REACTOR COOLANT			STEAM GENERATOR					RC PUMP MTR BRG TEMP				LOWER SEAL BRG TEMP °F
	LOOP T AVE °F	LOOP ΔT °F	COLD LEG RTD °F	LEVEL %	FW FLOW CORRECTED KLB/HR	STEAM PRESS PSIG	STM FLOW CORRECTED KLB/HR	THERMAL OUTPUT MW	UPPER RADIAL °F	UPPER THRUST °F	LOWER RADIAL °F	LOWER THRUST °F		LOOP T AVE °F	LOOP ΔT °F	COLD LEG RTD °F	LEVEL %	FW FLOW CORRECTED KLB HR	STEAM PRESS PSIG	STM FLOW CORRECTED KLB HR	THERMAL OUTPUT MW	UPPER RADIAL °F	UPPER THRUST °F	LOWER RADIAL °F	LOWER THRUST °F	
	LAST 1 MINUTE AVERAGES													LAST 1 MINUTE AVERAGES												

(Sheet 1c)

TIME	LOOP 31													LOOP 32												
	REACTOR COOLANT			STEAM GENERATOR					RC PUMP MTR BRG TEMP				LOWER SEAL BRG TEMP °F	REACTOR COOLANT			STEAM GENERATOR					RC PUMP MTR BRG TEMP				LOWER SEAL BRG TEMP °F
	LOOP T AVE °F	LOOP ΔT °F	COLD LEG RTD °F	LEVEL %	FW FLOW CORRECTED KLB/HR	STEAM PRESS PSIG	STM FLOW CORRECTED KLB/HR	THERMAL OUTPUT MW	UPPER RADIAL °F	UPPER THRUST °F	LOWER RADIAL °F	LOWER THRUST °F		LOOP T AVE °F	LOOP ΔT °F	COLD LEG RTD °F	LEVEL %	FW FLOW CORRECTED KLB/HR	STEAM PRESS PSIG	STM FLOW CORRECTED KLB/HR	THERMAL OUTPUT MW	UPPER RADIAL °F	UPPER THRUST °F	LOWER RADIAL °F	LOWER THRUST °F	
	LAST 1 MINUTE AVERAGES													LAST 1 MINUTE AVERAGES												

TIME	LOOP 33													LOOP 34												
	REACTOR COOLANT			STEAM GENERATOR					RC PUMP MTR BRG TEMP				LOWER SEAL BRG TEMP °F	REACTOR COOLANT			STEAM GENERATOR					RC PUMP MTR BRG TEMP				LOWER SEAL BRG TEMP °F
	LOOP T AVE °F	LOOP ΔT °F	COLD LEG RTD °F	LEVEL %	FW FLOW CORRECTED KLB/HR	STEAM PRESS PSIG	STM FLOW CORRECTED KLB/HR	THERMAL OUTPUT MW	UPPER RADIAL °F	UPPER THRUST °F	LOWER RADIAL °F	LOWER THRUST °F		LOOP T AVE °F	LOOP ΔT °F	COLD LEG RTD °F	LEVEL %	FW FLOW CORRECTED KLB/HR	STEAM PRESS PSIG	STM FLOW CORRECTED KLB/HR	THERMAL OUTPUT MW	UPPER RADIAL °F	UPPER THRUST °F	LOWER RADIAL °F	LOWER THRUST °F	
	LAST 1 MINUTE AVERAGES													LAST 1 MINUTE AVERAGES												

Post Trip Review of Selected Parameters

(Computer Print-out Sample)

0851 POST TRIP REVIEW

N0049A	N0050A	N0051A	N0052A	TIME1	P0398A	P0399A	Q0340A	T0496A	TIME2
0.1	0.2	0.1	0.1	4837.1	9.0	2.8	0.6	547.7	4837.1
0.1	0.2	0.1	0.1	4839.2	9.4	3.9	0.0	547.7	4039.2
0.1	0.2	0.0	0.1	4841.1	7.6	1.0	0.3	[illegible]	4841.2
0.1	0.1	0.1	0.1	4843.2	11.2	3.9	0.3	[illegible]	[illegible]
0851POST-TRIP DATA - TRIP TIME 004044									
0.1	0.2	0.1	0.1	4845.1	11.2	3.2	0.3	546.1	4845.2
0.1	0.2	0.1	0.1	4847.2	9.7	2.8	0.3	547.8	4847.2
0.1	0.2	0.0	0.1	4849.1	9.7	2.1	0.3	547.9	4849.2
0.1	0.2	0.1	0.2	4851.2	11.6	3.2	0.3	548.1	4851.2

N0031A	N0032A	N0035A	N0036A	N0041A	N0042A	N0043A	N0044A	N0045A	N0046A	N0047A	N0048A	N0049A	N0050A
0.2606	0.2453	0.0000	0.0000	0.01	0.00	0.00	-0.00	-0.00	-0.00	0.00	0.00	0.1	0.1
0.2202	0.2453	0.0000	0.0000	0.01	0.00	0.00	-0.00	0.00	0.00	0.00	0.00	0.0	0.1
0.1811	0.2537	0.0000	0.0000	0.01	0.00	0.00	-0.00	-0.00	-0.00	0.00	0.00	0.0	0.0
0.2406	0.2404	0.0000	0.0000	0.01	0.00	0.00	-0.00	-0.00	0.00	0.00	0.00	0.1	0.1
0.2732	0.3044	0.0000	0.0000	0.01	0.00	0.00	-0.00	-0.00	0.00	0.00	0.00	0.0	0.1
0.2262	0.3323	0.0000	0.0000	0.01	0.00	0.00	0.00	-0.00	0.00	0.00	0.00	0.1	0.1
0.2807	0.3212	0.0000	0.0000	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.1	0.2
0.2826	0.2660	0.0000	0.0000	0.01	0.00	0.00	0.00	0.00	0.00	0.00	0.00	0.1	0.1
0.2262	0.2923	0.0000	0.0000	0.01	0.00	0.00	0.00	-0.00	0.00	0.00	0.00	0.1	0.1
0.2660	0.2100	0.0000	0.0000	0.01	0.00	0.00	0.00	-0.00	0.00	0.00	0.00	0.1	0.1
0.1937	0.2031	0.0000	0.0000	0.01	0.00	0.01	0.00	0.00	0.00	0.01	0.00	0.1	0.1
0.2537	0.2486	0.0000	0.0000	0.01	0.00	0.01	0.00	0.00	0.00	0.00	0.00	0.1	0.1
0.2520	0.2963	0.0000	0.0000	0.01	0.00	0.01	0.00	0.00	0.00	0.00	0.00	0.1	0.2
0.2864	0.3148	0.0000	0.0000	0.01	0.00	0.01	-0.00	-0.00	0.00	0.00	0.00	0.1	0.1
0.1715	0.3345	0.0000	0.0000	0.01	0.01	0.01	0.00	-0.00	0.00	0.00	0.00	0.1	0.2
0.1571	0.3483	0.0000	0.0000	0.01	0.01	0.01	0.00	0.00	0.00	0.00	0.00	0.1	0.2
0052POST-TRIP DATA - TRIP TIME 004044													
0.1990	0.2943	0.0000	0.0000	0.01	0.00	0.01	0.00	0.00	0.00	0.00	0.00	0.1	[illegible]
0.2624	0.2642	0.0000	0.0000	0.01	0.01	0.01	0.00	0.00	0.00	0.00	0.00	0.1	[illegible]
0.2202	0.3360	0.0000	0.0000	0.01	0.01	0.01	0.00	0.00	0.00	0.01	0.00	0.1	[illegible]
0.23[illegible]4	0.362[illegible]	0.0000	0.0000	0.01	0.01	0.01	0.00	0.00	0.00	0.00	0.00	[illegible]	[illegible]
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]								

Table 2 (Sheet 3 of 3)

Alarm Type Write Up Listing

(Computer Print-Out Sample)

0811	RETRN	LO	T0497A	AI	RCL AUCT DT		0.0	L	0.0	PC
0811	ALARM	LO	T0497A	AI	RCL AUCT DT		-0.4	L	0.0	PC
0816	ALARM	CI	Y0007D	CI	REAC MAIN TR BKR B	NT TR				
0816	ALARM	CI	Y0026D	CI	REAC AUX TR BKR A	NT TR				
0816	RETRN	CI	Y0007D	CI	REAC MAIN TR BKR B	TR				
0816	RETRN	CI	Y0026D	CI	REAC AUX TR BKR A	TR				
0820	ALARM	CI	Y0007D	CI	REAC MAIN TR BKR B	NT TR				
0820	ALARM	CI	Y0026D	CI	REAC AUX TR BKR A	NT TR				
0820	RETRN	CI	Y0007D	CI	REAC MAIN TR BKR B	TR				
0821	RETRN	CI	Y0026D	CI	REAC AUX TR BKR A	TR				
0903	RETRN	LO	T0497A	AI	RCL AUCT DT		0.0	L	0.0	PC
0904	ALARM	LO	T0497A	AI	RCL AUCT DT		-0.1	L	0.0	PC
0909	ALARM	CI	Y0027D	CI	REAC AUX TR BKR B	NT TR				
0909	ALARM	CI	Y0006D	CI	REAC MAIN TR BKR A	NT TR				
0909	RETRN	CI	Y0027D	CI	REAC AUX TR BKR B	TR				
0909	RETRN	CI	Y0006D	CI	REAC MAIN TR BKR A	TR				
0915	RETRN	LO	T0497A	AI	RCL AUCT DT		0.0	L	0.0	PC
0915	ALARM	LO	T0497A	AI	RCL AUCT DT		-0.5	L	0.0	PC
0922	ALARM	CI	P0483D	CI	PRESSURIZER HI P CAUS RE	TR				
0922	ALARM	CI	P0481D	CI	PRESSURIZER HI P 2 PART RE	TR				
0922	ALARM	CI	P0480D	CI	PRESSURIZER HI P 1 PART RE	TR				
0922	RETRN	CI	P0483D	CI	PRESSURIZER HI P CAUS RE	NT TR				
0922	RETRN	CI	P0481D	CI	PRESSURIZER HI P 2 PART RE	NT TR				
0922	RETRN	CI	P0480D	CI	PRESSURIZER HI P 1 PART RE	NT TR				
0923	ALARM	CI	Y0006D	CI	REAC MAIN TR BKR A	NT TR				
0923	ALARM	CI	Y0027D	CI	REAC AUX TR BKR B	NT TR				
0923	RETRN	CI	Y0027D	CI	REAC AUX TR BKR B	TR				
0923	RETRN	CI	Y0006D	CI	REAC MAIN TR BKR A	TR				
0924	ALARM	CI	Y0007D	CI	REAC MAIN TR BKR B	NT TR				
0924	ALARM	CI	Y0026D	CI	REAC AUX TR BKR A	NT TR				
0924	RETRN	CI	Y0026D	CI	REAC AUX TR BKR A	TR				
0924	RETRN	CI	Y0007D	CI	REAC MAIN TR BKR B	TR				
0935	ALARM	CI	P0483D	CI	PRESSURIZER HI P CAUS RE	TR				
0935	ALARM	CI	P0481D	CI	PRESSURIZER HI P 2 PART RE	TR				
0935	ALARM	CI	P0480D	CI	PRESSURIZER HI P 1 PART RE	TR				
0935	RETRN	CI	P0483D	CI	PRESSURIZER HI P CAUS RE	NT TR				
0935	RETRN	CI	P0481D	CI	PRESSURIZER HI P 2 PART RE	NT TR				
0935	RETRN	CI	P0480D	CI	PRESSURIZER HI P 1 PART RE	NT TR				
0936	ALARM	CI	Y0026D	CI	REAC AUX TR BKR A	NT TR				
0936	ALARM	CI	Y0007D	CI	REAC MAIN TR BKR B	NT TR				
0936	ALARM	CI	P0483D	CI	PRESSURIZER HI P CAUS RE	NT TR				
0936	RETRN	CI	Y0026D	CI	REAC AUX TR BKR A	TR				
0936	RETRN	CI	Y0007D	CI	REAC MAIN TR BKR B	TR				
0936	ALARM	CI	P0481D	CI	PRESSURIZER HI P 2 PART RE	TR				
0936	ALARM	CI	P0480D	CI	PRESSURIZER HI P 1 PART RE	TR				
0936	RETRN	CI	P0483D	CI	PRESSURIZER HI P CAUS RE	NT TR				
0936	RETRN	CI	P0481D	CI	PRESSURIZER HI P 2 PART RE	NT TR				
0936	RETRN	CI	P0480D	CI	PRESSURIZER HI P 1 PART RE	NT TR				
0937	ALARM	CI	Y0027D	CI	REAC AUX TR BKR B	NT TR				
0937	RETRN	CI	Y0027D	CI	REAC AUX TR BKR B	TR				
0937	ALARM	CI	Y0026D	CI	REAC AUX TR BKR A	NT TR				
0937	RETRN	CI	Y0026D	CI	REAC AUX TR BKR A	TR				
0950	ALARM	CI	Y0027D	CI	REAC AUX TR BKR B	NT TR				
0950	RETRN	LO	T0497A	AI	RCL AUCT DT		0.0	L	0.0	PC
0951	ALARM	LO	T0497A	AI	RCL AUCT DT		-0.3	L	0.0	PC
0951	ALARM	CI	Y0026D	CI	REAC AUX TR BKR A	NT TR				
0952	RETRN	CI	Y0026D	CI	REAC AUX TR BKR A	TR				
[illegible]	RETRN	CI	Y0027D	CI	REAC AUX TR BKR B	TR				
[illegible]	ALARM	CI	Y0026D	CI	REAC AUX TR BKR A	NT TR				
[illegible]	ALARM	CI	[illegible]	CI	REAC AUX TR BKR B	NT TR				
[illegible]	RETRN	CI	Y0026D	CI	REAC AUX TR BKR A	TR				
[illegible]	[illegible]		[illegible]	CI	REAC AUX TR BKR B	TR				
[illegible]	ALARM	CI	Y0026D	CI	REAC AUX TR BKR A	NT TR				

TABLE 3

Data and Information Equipment	Power Sources
Alarm Type Write Up	MCC-39
Trend Type Write Up	MCC-39
Log Type Write Up	MCC-39
Sequence of Events	MCC-39
Protective relays (Control Room, Plant, Buchanan)	125V DC (Safety Related); 110V AC (safety related instrument bus) or voltage level and class of the equipment being protected.
First out reactor and Turbine trip annunciators	125V DC (Safety Related)
Annunciators	125V DC (Safety Related)
Strip chart recorders	110V AC Safety Related instrument bus
Instrument bus transient recorders	UPS
Switch position including disagreemnt (Amber) lights	125V DC (Safety Related)
Indicating lights (Rod bottom)	110V AC to 125V DC
Indicators (meters, etc.)	110V AC
Lock out relays	110V AC or 125 V DC

Note: Plant computers are fed from MCC-39, a non-safety related power supply, which will be interrupted only in the unlikely occurrence of both a black out and safety injection.

2.1 EQUIPMENT CLASSIFICATION AND VENDOR INTERFACE (REACTOR TRIP SYSTEM CONTONENTS)

The Authority has performed a review of the Reactor Trip System (RTS) to establish the extent of conformance with the position of the subject Generic Letter.

1. <u>Equipment Classification</u>

The Authority has verified that the Reactor Trip system is classified as safety-related in QAP 2.1, Quality Assurance Program Scope. This document is used to define plant safety related activities. It should be noted that classification exists on a system basis only, not by individual components.

The Authority considers this a conservative approach since many components that may not be required to meet the safety-related criteria are still classified as such. Components of safety-related systems may be reclassified as non-safety related in accordance with the Authority's Plant Administrative Procedures. Although the boundary of the RTS is defined through the intent of the system, a classification on a component basis will be formalized by June, 1984.

Indian Point 3 safety related systems are identified as "Category I" and listed in the QA Program.

The following information handling systems address specific requirements in support of safety-related systems including the RTS:

° Work Order Requests - "Category I" is noted on the Work Request Form in accordance with Administrative Procedure AP-9.

° Modification Packages - "Category I" along with a classification request form are included in the modification Package in accordance with AP-12.

° Procurement Documentation - "Category I" is indicated on all purchase requisitions and other documentation in accordance with applicable procedures.

° QA Program - "Category I" is indicated in the Systems List.

° Maintenance, Testing and Calibration Procedures - "Category I" is identified on all affected components.

On the following information handling documents the RTS is not specifically identified as safety related:

° Procedures are not identified as relating to safety-related activities. These procedures, however, are reviewed and approved prior to becoming effective. The record of reviews, including the PORC review and approvals including the effective date is indicated on the first page of the procedure.

° Drawings which are not marked as safety related will be stamped, as required, to indicate that they contain information dealing with safety-related systems.

- Instruction manuals which are not marked as safety related will be stamped to indicate that they contain information dealing with safety-related systems.

The Authority is presently reviewing these documents. This effort is expected to be completed by mid 1985.

2. Vendor Interface

The Authority receives pertinent information associated with the RTS from Westinghouse and other vendors. This information is received by the plant and is distributed to appropriate departments for review and assessment. Westinghouse also provides guidance on parts procurement, mandatory and recommended maintenance, and communicates with the Authority on relevant issues via Technical Bulletins.

The present method of interfacing with Westinghouse and other vendors will be formalized and a continuing vendor interface program will be established implemented and maintained in order to accomplish the following:

- To ensure that the latest Westinghouse and other vendor information for safety-related conformance is complete, current and controlled throughout the life of the plant, and that it is properly reviewed, assessed and implemented.

- To ensure that Westinghouse and other vendor information for safety-related conformance is appropriately referenced or incorporated in plant instructions and procedures.

- To ensure that the vendor interface program is closely coupled with the action of safety classification of equipment.

- To include periodic communication with vendors to assure that all applicable information is timely received.

- To ensure that a system of positive feedback with Westinghouse and other vendors for mailings containing technical information is used.

- To define the interface and division of responsibilities among the Authority and the nuclear and non-nuclear divisions of Westinghouse and other vendors to assure that requisite control of and applicable instructions for maintenance work are provided.

The Authority. through programs such as the INPO's NUTAC, will evaluate in detail the plant's existing capabilities for interfacing with Westinghouse and other vendors. Pending a timely completion of the NUTAC's efforts, the Authority will review and evaluate the results for plant specific applicability and expects to submit a vendor interface program description for the RTS to the NRC by mid 1984.

2.2 EQUIPMENT CLASSIFICATION AND VENDOR INTERFACE (ALL SAFETY RELATED COMPONENTS)

2.2.1 Equipment Classification

1. The Authority's criteria for identifying components as safety related within systems currently classified as safety- related is on a system level as discussed in the response to item 2.1.1.

2. The information handling systems used to identify safety-related components comprise of various documents such as the FSAR, the QA Program, NSSS technical manuals, vendor manuals and instructions, drawings, equipment specifications, etc.

   These original documents were developed during the plant design and were validated as part of their review and approval process.

3. In accordance with existing plant procedures such as Administrative Procedures, Departmental Procedures, Modification Procedures, Procurement Procedures, etc., a plant activity is reviewed to determine whether or not such an activity is safety-related and what constraints and requirements if any, apply.

4. The Authority's management controls utilized to verify that the procedures for preparation, validation and routine utilization of the information handling system have been followed are by such means as management's review and concurrence, PORC and/or SRC review and concurrence, QA audits and surveillance, etc.

5. The concerns related to an appropriate design verification for procurement of safety related components are addressed by our present procurement procedures. The original specification requirements shall be met as a minimum ("Procurement of like in kind"). In such cases where this method is not applicable the plant Administrative Procedures (AP-25.2) provide for the substitution of safety related components following appropriate engineering evaluation and design verification (e.g. calculations testing or analysis). Qualification testing is specified for procurement of safety-related components, as necessary, and it is evidenced, for example, by the Authority's program. This program provides assurances that the system/component will perform as intended by design during all anticipated service conditions.

2.2.2 Vendor Interface

The Authority, through programs such as the INPO's NUTAC, is evaluating in detail the plant's existing capabilities for interfacing with Westinghouse and other vendors. Pending a timely completion of the NUTAC's efforts, the Authority will review and evaluate the results for plant specific applicability and submit a vendor interface program description to the NRC by mid 1984.

3.1 Post-Maintenance Testing (Reactor Trip System Components)

A review of the Post Maintenance Test Program was performed to ensure that:

a. post maintenance testing of safety related components in the Reactor Trip System demonstrates that these safety related components can perform their intended functions in the intended manner and

b. that this post maintenance testing is required to be conducted prior to returning these components to service.

This review determined that the existing post maintenance testing was properly and adequately performed. However, Administrative Procedure revisions have been incorporated to ensure greater program confidence.

A review of the manufacturers guidance and engineering recommendations for testing of the reactor trip breakers has been performed and existing reactor trip breaker surveillance and test procedures reflect the appropriate recommendations. As per our extension request for Generic Letter 83-28 (IPN-83-75, dated 9/8/83) in order to perform a review of vendor/manufacturer test guidance for the other safety related components in the reactor trip system, it is necessary to first complete Item 2.1 of Generic Letter 83-28. Therefore, a report describing our status of conformance and a schedule to achieve full compliance with the requirements of this position is expected to be submitted to the NRC by mid 1984.

3.2 Post Maintenance Testing (All Other Safety Related Components)

A review of the Post Maintenance Test Program was performed to ensure that:

a. post maintenance testing of safety related components demonstrates that these safety related components can perform their intended functions in the intended manner and

b. that this post maintenance testing is required to be conducted prior to returning these components to service.

This review determined that the existing post maintenance testing was properly and adequately performed. However, Administrative Procedure revisions have been incorporated to ensure greater program confidence.

A review has been performed of available manufacturers and vendor test guidance for safety related components to ensure that post maintenance testing procedures reflect appropriate vendor and manufacturer guidance.
This review had also determined that all of the specific testing or test guidance which was identified by vendor manuals had already been incorporated into the applicable procedures or programs. It should be noted that very little vendor information was devoted to specific testing or test guidance and many manuals have no reference to any testing concerns.

As per our extension request for Generic Letter 83-28, the final review will not be complete until Item 2.2 ensures that vendor and manufacturer information is current and complete. Although Item 3.2.2 is now partially complete it is still necessary to first complete Item 2.2. Therefore, in accordance with the schedule established for the response to Item 2.2, a report describing our status of conformance and a schedule to achieve full compliance with the requirements of action Item 3.2.2 is expected to be provided to the NRC by mid 1984.

4.1 REACTOR TRIP SYSTEM RELIABILITY (VENDOR-RELATED MODIFICATIONS)

The Authority has reviewed the vendor-recommended reactor trip breaker modifications in accordance with the requirements of the NRC Generic Letter 83-28. In reviewing the bulletins and letters from the vendors, it was verified that the Westinghouse modifiction NCD-Elec-18 was applicable to Indian Point 3. This modification dealt with the DB-50 undervoltage trip attachment (UVTA) and the design changes made to the device in 1971. The Westinghouse commitment was to provide the modified units to all sites that have the devices. NCD-Elec-18 was the Westinghouse document generated to implement the change-out program. This modification was instituted and completed per the requirements of NCD-Elec-18. Because of the potential for inadvertent site reinstallation of pre-1972 units in the intervening 10 years, a visual reconfirmation was conducted. The visual inspection was performed by plant personnel and verification made that all UVTA's of the reactor protection system are of the proper design.

Two other Westinghouse modifications were identified during the review process. One modification concerned the DS-416 switchgear as identified in the March 31, 1983, Westinghouse Letter (NS-EPR-2744). This modification does not apply to the Authority because the Indian Point 3 Nuclear Power Plant does not employ the DS-416 breaker for the reactor trip function. The other Westinghouse modification was identified in NSD-TB-75-2 of February 20, 1975, concerning a DS circuit breaker wiring bundle problem. Because Indian Point 3 does not use the DS circuit breakers, for the reactor trip function, this too is not applicable to the Authority.

4.2 Reactor Trip System Reliability (Preventive Maintenance and Surveillance for Reactor Trip breakers)

The current maintenance procedure for the reactor trip breakers defines the necessary preventive maintenance program and lubrication requirements. This program is performed annually, except during periods of reactor operation. All the breaker manufacturer recommendations are addressed in the procedure. The results observed during this maintenance are reviewed and pertinent recommendations are made to preclude any future degradation of operability.

As indicated in our letter of 9/8/83 (IPN-83-75), life cycle testing of the breaker undervoltage and shunt trip attachments of the reactor trip switchgear is presently being conducted by Westinghouse for the Westinghouse Owners Group (W-OG). This program is aimed toward establishing the service life of these devices, and substantiating periodic test requirements with proper maintenance. The W-OG program is scheduled for completion in the second quarter of 1984. Following the Authority's evaluation for plant specific applicability, the results of this program will be factored, as necessary, into the plant maintenence and replacement program.

4.3 RPS - Automatic Actuation of Shunt Trip Attachment

A description of the conceptual design for the IP-3 proposed modification is outlined in the enclosed Appendix.

4.4 (B&W plants only)

Not applicable to Indian Point 3.

4.5 Reactor Trip System Reliability (System Functional Testing)

(1) On-line functional testing of the reactor trip system, is performed on a monthly basis. This includes testing of the breaker undervoltage and shunt trip features. The associated sensors, transmitters and control room pushbuttons are tested/calibrated on a refueling outage basis.

(2) As noted in (1) above, on-line testing is performed on the reactor trip system.

(3) The W-OG in February, 1983, submitted WCAP-10271 to the NRC for review. WCAP-10271, "Evaluation of Surveillance Frequencies and out of Service Times for the Reactor Protection Instrumentation System" documents an evaluation of the impact on RPS unavailability of current and extended surveillance intervals. The WCAP considers common mode failure, operator error, reduced redundancy during testing and equipment bypass. WCAP-10271 also considers correlative effects on plant

operation and safety including the manpower expenditure associated with surveillance, the number of inadvertent trips which occur during test and the distraction from plant monitoring on the part of the control room operator and shift supervisor associated with testing.

Supplement 1 to WCAP-10271 which was submitted to the NRC in September, 1983, is an extension of the evaluation and provides a discussion of component wearout caused by testing. The NRC review of WCAP-10271 to date has resulted in a request for additional information the NRC felt necessary to complete the review. On October 4, 1983 this information was submitted to the NRC in response to that request by the W-OG and included an overall evaluation of the impact on plant safety of RPS surveillance, a discussion of the uncertainty of the failure rates and common mode failure and more detail concerning the impact of surveillance intervals on RPS unavailability. WCAP-10271, Supplement 1, and the information provided to the NRC in defense of WCAP-10271 provides in a comprehensive form the information requested by this item.

The conclusion of WCAP-10271 and Supplement 1 is that less frequent testing of RPS components is warranted and will result in an improvement in overall plant safety and equipment reliability.

Following completion of the NRC review, the Authority will perform a plant specific evaluation and determine a submittal date of any proposed changes to the Technical Specifications.

Appendix

REPORT ON REACTOR TRIP SYSTEM

WITH SHUNT TRIP ATTACHMENTS

(NRC Generic Letter 83-28, July 8, 1983)

Section 4.3

INTRODUCTION

This report provides a description of the proposed modifications for the automatic actuation of the trip attachment of the Reactor Trip Breaker as required by the NRC's generic letter 83-28 dated July 8, 1983.

BACKGROUND

On February 25, 1983, there was a low level steam generator trip at PSE&G's Salem I PWR nuclear power plant. As a consequence, an automatic reactor trip signal was transmitted to the automatic reactor trip relay logic, but the associated undervoltage relays failed to open the two circuit breakers that feed the reactor control rods. Finally, to protect the reactor, the Salem breakers were opened manually by operating the push button switches.

Following this, the Nuclear Regulatory Commission first issued IE-Bulletin 83-01 to investigate the scram breaker maintenance programs and its operational reliability and incorporated these studies into NUREG documents - NUREG-0977 and NUREG-1000. Westinghouse also issued a Technical Bulletin (DB-50 Reactor Trip Breaker Maintenance). In order to prevent the occurrence of the Salem type event, the addition of the shunt trip mechanism was recommended. Furthermore, on July 8, 1983 NRC issued generic letter 83-28 to all licensees requesting to submit a report describing their modifications for the automatic actuation of the shunt trip attachment.

EXISTING DESIGN

The automatic trip logic at the Indian Point Unit 3 Nuclear Power Plant consists of a relay matrix as depicted in Figure 1. One relay matrix is associated with Train "A" and a similar set with Train "B".

During normal plant operation, all the relay contacts are normally closed and maintain the undervoltage relays for each train energized. These undervoltage relays become de-energized when two parallel contacts open, then the relay plungers are released from the magnetic attraction and operate a mechanical linkage that opens the Reactor Trip Breakers which cause the Reactor Control Rod Assembly to fall into the reactor core.

The manual trip is accomplished with the breaker shunt coil (TC), and is energized by either one of the two (2) parallel switches (located on the central control board) in series with the trip coil, refer to Figure 2. This manual control operates a mechanical attachment independent of the undervoltage relay. The manual trip also actuates the undervoltage trip attachment.

REFERENCES:

The following documents were reviewed and used in developing this report:

1. NUREG-1000 (SECY 83-158)
   Salem ATWS Generic Implication Task Force Report

2. Generic Letter 83-28
   Required Actions Based on Generic Implications of Salem ATWS Events

3. Westinghouse Owners Group (WOG) and Technical Specification Positions.
   a. Generic Shunt Trip Modification SER Response
   b. WOG/NRC Safety Evaluation Report
      Design Modification to provide Automatic Actuation of RT Breaker Shunt Trip Attachment (STA).

4. IP-3 Schematic Drawings and Operating Procedures

5. Westinghouse Technical Bulletin NSD-TB-83-02
   DB-50 Reactor Trip Breaker Maintenance

6. Westinghouse Instruction Book for Types DB-50 Air Circuit Breakers, 1966.

7. IEEE Std. - 352-1975 N.P.P. Generator Station Protection.

DESCRIPTION OF PROPOSED MODIFICATION

Shunt Trip Actuation

Presently the "RT" and "BY" breakers in each train have manual initiation of the shunt trip and the undervoltage (UV) mechanical linkage trip, and automatic initiation of the UV mechanical linkage trip. The power sources to the breaker shunt trip coils are safety related. As indicated in Figure 2, the red and green breaker position indication lights, which are powered from the same fuses 125 VDC supply, provide indication on the availability of power to the shunt trip circuit.

The Authority proposes to add two redundant auxiliary relays in parallel with the existing undervoltage relay (UV) of each breaker, as shown in Figure 3. The auxiliary relays will be selected such that they will function within the capacity of their associated power supplies and their contacts will be adequately sized to accomplish the shunt trip function. The auxiliary relays will have the ability to perform their intended function up to a voltage as high as approximately 115% of the normal voltage. The Authority will ensure that these relays will not cause overload conditions of the UV circuit.

Contacts from each of these auxiliary relays connected in series will initiate the actuation of the shunt trip coil automatically. Under loss of DC power to the undervoltage and the breaker circuit, both the undervoltage relays (UV) and auxiliary relays will be de-energized and the breaker trip will be initiated by the linkage mechanism as well as by the shunt trip coil simultaneously.

The circuit that will be used to implement the automatic shunt trip functions will be classified and designed as safety-related and the procurement, installation, operation, testing and maintenance of the circuitry will be in accordance with the plant quality assurance procedures.

The Authority, in participation with the Westinghouse Owner's Group, will assure that the shunt tip attachment and associated circuitry will be seismically and environmentally qualified. During the final design stages, the Authority will ensure that the separation operability and testing criteria are properly addressed.

Isolation Switch

The introduction of an isolation switch in the automatic shunt trip circuit of each breaker provides the necessary flexibility to perform on-line testing of these trip breakers (one breaker at a time) during reactor operation.

DISCUSSION

The Reactor Control Rod and Trip System logic is powered from a D. C. Emergency System (safety related power supply) with necessary redundance through two sets of series breakers (52/RTA and 52/BYB on Train-A and 52/RTB and 52/BYA on Train-B, respectively). Also, the control power supplies to these breakers are connected to the same train A and B systems with necessary separation and single failure criteria.

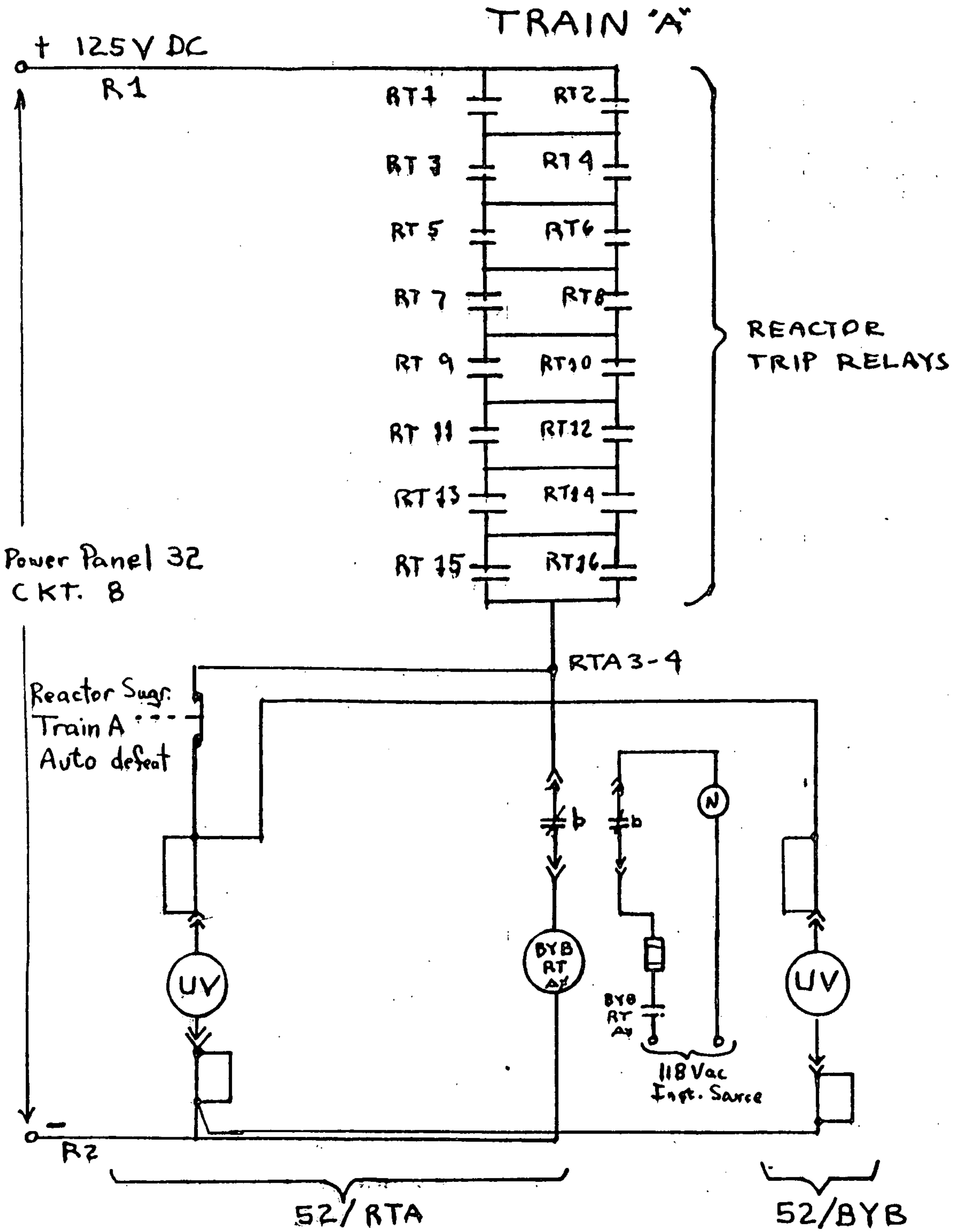
The two sets of (under-voltage) auxiliary relays will be connected in parallel to each breaker under-voltage (UV) relays, and their contacts in series, such that failure of one of these auxiliary relays cannot cause a reactor trip.

The isolation switch connected to the under-voltage auxiliary relays contacts (in the shunt trip circuit) shall be used to facilitate testing. An additional isolation switch will enable testing of the under-voltage auxiliary relays independent of the UV attachment. During the breaker racked-in position after the test, the isolation switch shall be closed only after the under-voltage auxiliary relays are picked up.

As discussed earlier in this report, the above shunt trip scheme with two sets of auxiliary (under-voltage) relays and latch switch provides the optimum controls for the necessary trip condition.

The Authority is presently conducting further reviews of the positions developed by the Westinghouse Owner's Group in response to the questions raised by the NRC staff, in their Safety Evaluation Report dated August 10, 1983. Following the completion of this review and as part of the final design effort, the Authority will inform you of our positions on the subject issues.

IP-3 Reactor Protection System
"Automatic Trip Schematic"

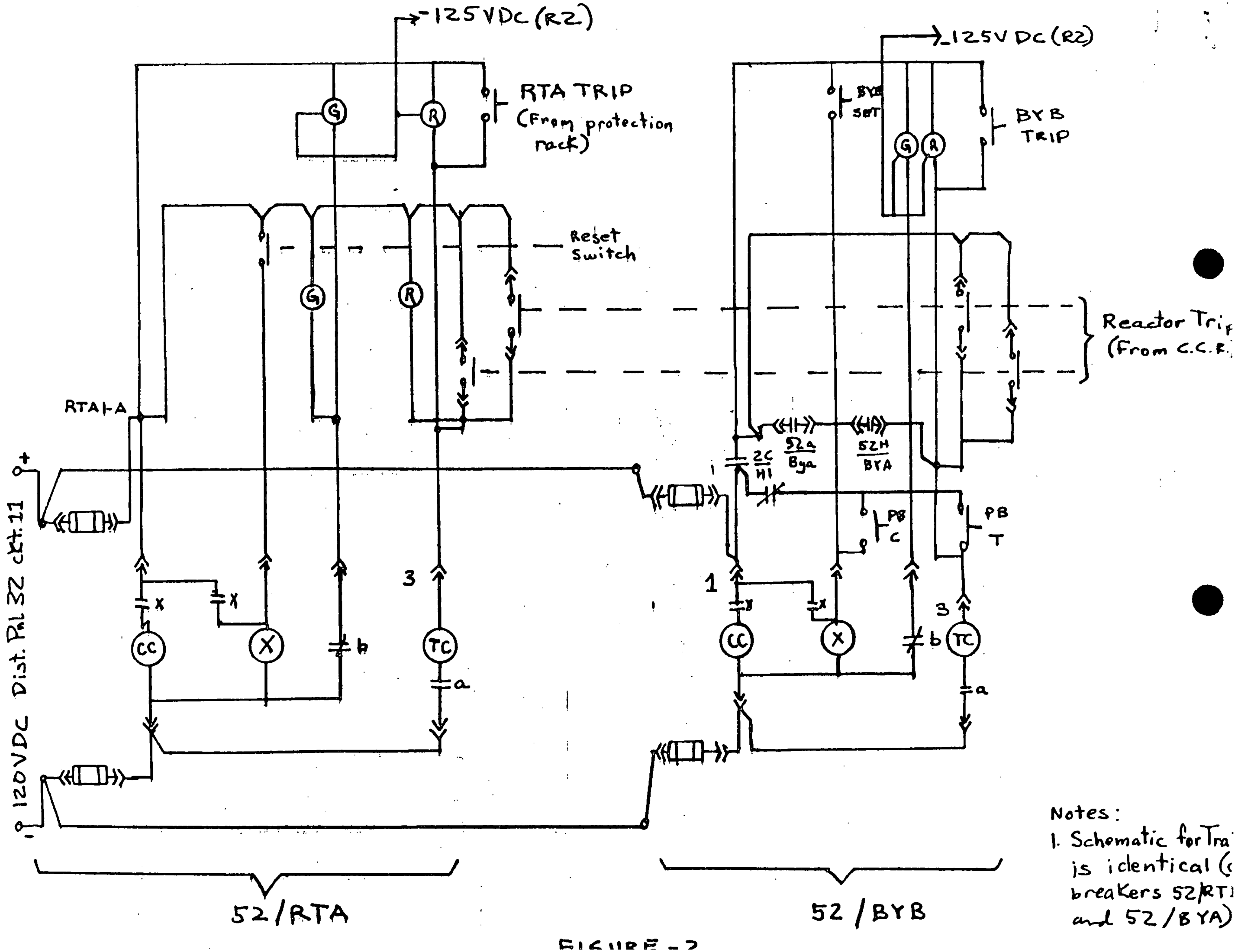


Notes:

1. Schematic for Train B is identical (circuit breakers 52/RTB and 52/BYA).

FIGURE 1

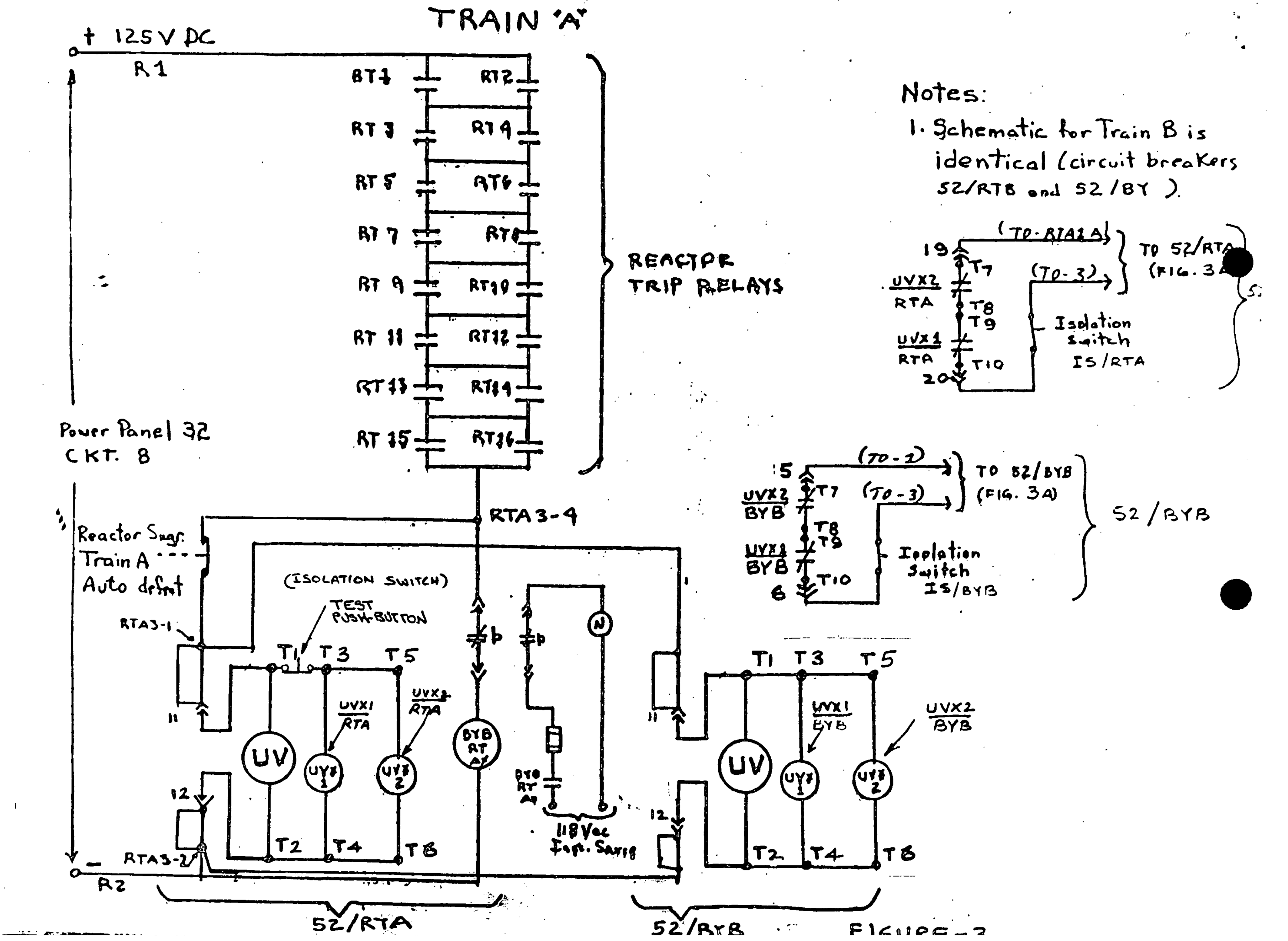
IP-3 Reactor Protection System
Manual Trip Logic



Notes:
1. Schematic for Tra
is identical (
breakers 52/RT
and 52/BYA)

FIGURE-2

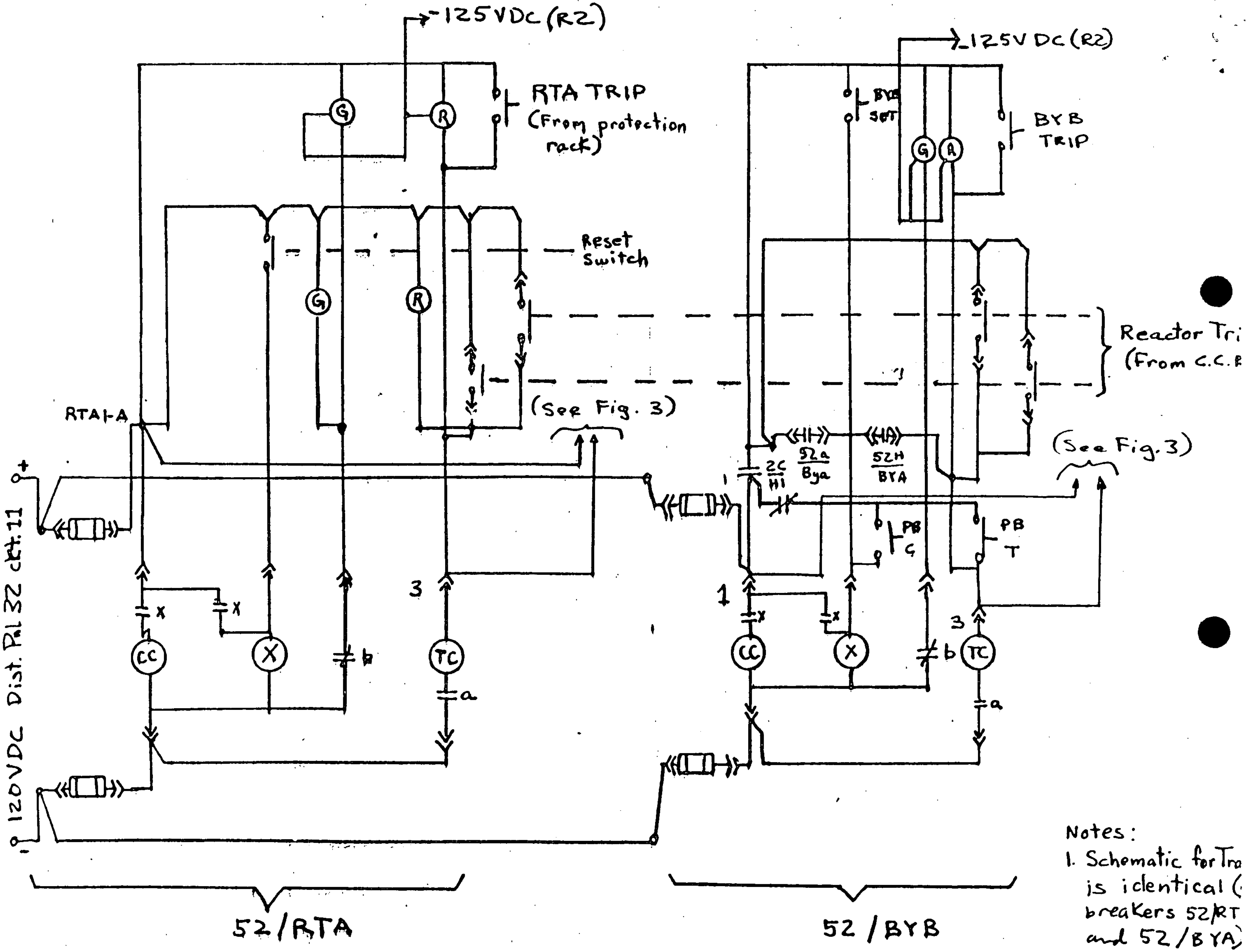
IP-3 Reactor Protection System
(PROPOSED MODIFICATION)



Notes:
1. Schematic for Train B is identical (circuit breakers 52/RTB and 52/BY).

FIGURE-3

IP-3 Reactor Protection System
(PROPOSED MODIFICATION)



Notes:
1. Schematic for Tra is identical (breakers 52/RT and 52/BYA)

FIGURE-3A