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INVESTIGATION OF N REACTOR SCRAM OF SEPTEMBER 30, 1970

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INVESTIGATION OF N REACTOR SCRAM OF SEPTEMBER 30, 1970

I. INTRODUCTION

During the early phases of startup and power ascension activities on September 30, 1970, the Hanford N Reactor received an automatic shutdown (scram) signal from its safety system which monitors reactor coolant flow. The reactor neutron absorbing safety rod system, which should normally function to achieve reactor shutdown, failed to automatically respond to the scram signal by inserting all withdrawn rods into the reactor core. Failure of the neutron poison rods to automatically insert caused the automatic activation of the reactor safety backup shutdown system which achieved an immediate and safe shutdown of the reactor.

Investigation of the September 30 scram of N Reactor was accomplished by investigating teams from both the Atomic Energy Commission and Douglas United Nuclear, Inc. This report covers the investigations, findings, and corrective actions taken by DUN as a result of the September 30 scram of N Reactor.

Failure of the rod system to respond to the scram signal was conclusively determined to have been the result of a "sneak" circuit permitted by the shorting failure of multiple in-series diodes specifically designed into the rod control circuitry to prevent this type of sneak circuit. It was also clearly established that at no time was the reactor in or near an unsafe condition.

II. N REACTOR DESCRIPTION

The N Reactor plant is located in the 100-N Area within the Atomic Energy Commission's Hanford reservation. The site is immediately adjacent to the Columbia River and is about 39 miles north of Richland, Washington.

The reactor is a graphite-moderated, light-water cooled, dual-purpose reactor which produces plutonium, and other important isotopes for the AEC and by-product steam which is piped to the Washington Public Power Supply System No. 1 turbine-generator facility where electrical power is generated.

A tabular summary of the most significant design parameters is given in Table I. A brief description of the principal systems is presented below.

A. Primary Coolant System

N Reactor is fueled with zirconium alloy clad metallic uranium fuel elements which are contained in 1003 horizontal zirconium alloy

TABLE I

SUMMARY OF NPR DESIGN PARAMETERS

Lattice

Spacing	8" horizontal x 9" vertical
Pattern	
Tubes per horizontal row:	34
Tubes per vertical row:	32 installed, 34 total
Omitted from each corner:	21 as installed, 21 total
Number of tubes:	1003 installed, 1072 total

Graphite

Overall dimensions	
Length:	39 feet, 5 inches
Width:	33 feet, 0 inches
Height:	33 feet, 3-1/2 inches
Weight of graphite:	3300 tons raw, 1800 tons finished
Density of graphite:	1.7 g/cc material, 1.29 g/cc as stacked

Shield

Thermal, front and rear:	8" cast-iron blocks
Rest of thermal shield:	1" boron-steel plate
Biological Shield	
Front and rear:	40" serpentine and iron aggregate concrete
Sides:	43" high density concrete
Top:	65" high density concrete
Bottom:	102" regular concrete

Safety Systems

86 horizontal, water-cooled, hydraulically-driven control rods
40 entering from left side; 46 entering from right side
32" horizontal by 36" vertical spacing, 29.5 foot travel
11 horizontal by 9 vertical pattern
B4C absorber in Titanium sheath material

107 vertical ball channels; 4" x 4" square opening
32" x 32" spacing
10 wide by 12 long pattern
Ceramic balls of Alumina; Samarium oxide as neutron absorber

TABLE I

(Cont'd)

Graphite Cooling System

Side to side orientation, 640 tubes
Every filler layer vertically and on 16" centers, horizontally
Zircaloy-2 tubes, 3/4" O.D.; carbon steel end fittings; pigtail connectors
Normally 5000 gpm recirculating process water
Once-through raw water back-up of maximum reliability

Process Tube and Fittings

Zircaloy-2; 2.7" I.D. and 0.25" wall; 53' long, ribless
End connection to carbon steel nozzle assembly
 Inlet end; rolled joint to short nozzle; dynamic gas seal
 Outlet end; rolled joint to long nozzle; static gas seal

Reactor Manifolding

Individual tube nozzle-to-header connectors on both inlet and outlet
Carbon steel; 2.105" I.D., 2.625" O.D.; average length of 53'
Piping to and from 109-N:
 16 parallel lines, 8 on each side
 Carbon steel, nominal 18" diameter

Primary Cooling System

Six parallel cells fed from reactor outlet manifold piping
Each cell contains two heat exchangers in parallel, a primary pump and valving
Pumps are horizontal, centrifugal, injection shaft seal type
Pump drives are 9000 hp steam turbines with 400 hp electric pony motors
Steam generators are horizontal, U-tube type of $\sim$16,000 ft^2 surface
 Shell design pressure is 800 psi
Surge vessel (pressurizer)-1200 ft^3 capacity, with 2350 KW heater capacity
Piping is carbon steel except for heat exchanger tubes and valve trim

Secondary Steam Loop

System design pressure of 600 psi
Sixteen dump condensers with design capacity of 12.6 x 10^6 lb/hr

River System

Condenser cooling supplied by four 100,000 gpm 3500 hp electric-driven pumps
Backup by two 35,000 gpm direct diesel-driven pumps
Emergency cooling from 200,000 gallon storage tank
 Supplied by two 35,000 gpm direct diesel-driven booster pumps
Last ditch supply to graphite cooling system
 Supplied by two 15,000 gpm direct diesel-driven pumps

TABLE I

(Cont'd)

Standby Power

Steam supply from one 575,000 lb/hr, 400 psi, oil-fired boiler and two 200,000 lb/hr, 400 psi, oil-fired boilers
Local station electric power from 15,625 KW turbine-generator set

Water Treatment

Filter plant with 1,050,000 gallon storage
Demineralized water plant with 1,000,000 gallon storage

Confinement

Primary zone designed for +5 psig, -2 psig
Sufficient vent area to dispose of steam surge
Fog spray system to control confiner to near atmospheric pressure
Filter system to remove 99 percent of particulates and >50 percent radiohalogens

process tubes which penetrate the graphite core (Figure 1). High purity water (primary coolant) is pumped through these tubes where it picks up the heat which is released from the uranium fuels.

The portion of the primary cooling system within the reactor building (105-N) consists of 16 parallel lines which conduct the cooling water from an inlet water manifold to the reactor. Each of these 16 lines terminates in a vertical header to which are attached from 54 to 66 connectors which supply coolant to each individual process tube. A similar arrangement at the reactor outlet conducts the coolant from the process tubes to an outlet water manifold. These lines are of carbon steel and are sized to provide an adequate coolant flow rate to the reactor. The primary loop design conditions are 1825 psig and 600 F. Primary coolant pressure is maintained at a sufficiently high level to prevent boiling at any point in the system.

From the reactor the primary coolant normally goes to twelve parallel steam generators in which the process heat is dissipated to the secondary steam system by boiling secondary water in the shell side of the steam generators.

The heat exchanger building (109-N) is located immediately adjacent to the reactor building and consists of six identical cells arranged in parallel. Cell 6 was added later, and is located adjacent to cell 1. Each cell contains two parallel steam generating heat exchangers, a primary coolant circulating pump and associated valves and instrumentation.

Steam-turbine drive units are used for the primary circulating pumps for each cell. Each drive turbine has a design rating of 9000 hp at 3600 rpm. Steam for the primary pump drive is supplied by the main steam generators during normal reactor operation and by standby oil-fired boilers when steam from the steam generators is not available.

In addition to the drive turbine, a 400 hp electric pony motor capable of driving each primary coolant pump at about 900 rpm is mounted on the same shaft. These motors provide sufficient water recirculation to prevent reactor damage during the scram transient and shutdown conditions which would follow complete loss of steam to the turbines. The pump-drive units are located outside the shielded cell and are connected to the pump by an extension shaft through the cell wall to provide direct connected drive.

During dual-purpose operation of N Reactor, the major fraction of the steam generated is routed to the export turbine generators at the Washington Public Power Supply System No. 1 power generating facility. A portion of the steam generated is still used to drive the primary reactor pumps and the reactor pump house turbine-generator unit and to condition the 16 reactor dump condensers to take up the full steam load in the event of an export turbine-generator trip-off.

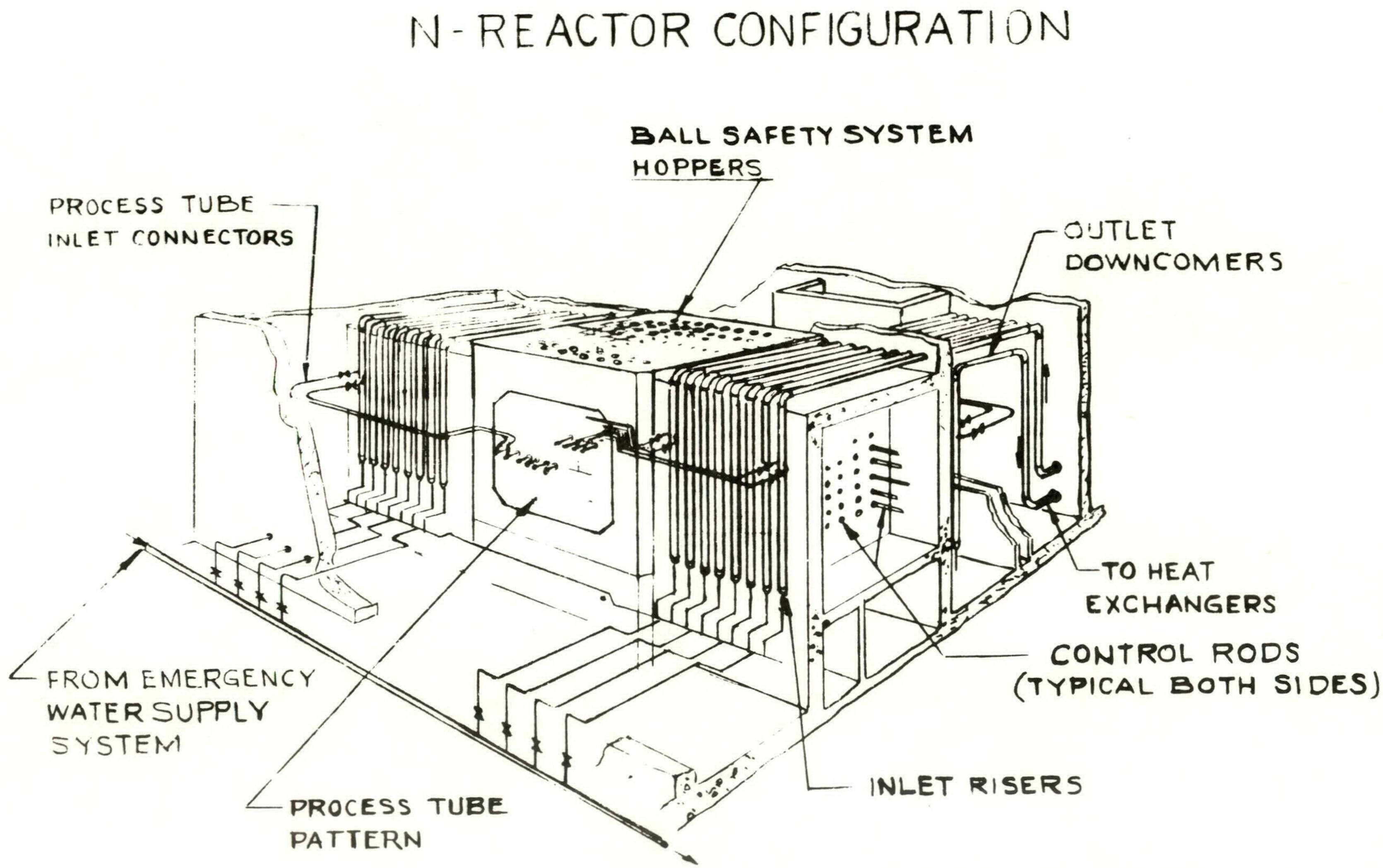


Figure 1. N Reactor Configuration

B. Horizontal Control and Safety Rod System

The reactor graphite moderator stack is penetrated by horizontal side-to-side channels for the control rod system, in a 32-inch horizontal by 36-inch vertical spacing. The 86 horizontal control control rods are water cooled and hydraulically driven and can be extended through the active zone of the reactor. Approximately half of them enter from each side of the reactor graphite block. This rod system provides both operating control and reactor shutdown nuclear safety functions.

Shown in Figure 2 is a simplified schematic of a typical rod hydraulic system. Each rod has a slow speed, hydraulic-powered drive system for fine control of reactor flux distribution and power level. Alternating current (AC) solenoids are actuated to control the normal in and out motion of the rods.

In addition to the normal control system, a DC-powered scram solenoid for each rod provides an independent means to rapidly drive rods into the reactor in emergency or "scram" conditions. When a rod scram signal occurs, each rod scram solenoid is de-energized which bleeds off the pressure from two pilot-operated scram valves. This allows high pressure oil stored in each rod's accumulator to drive the rod into the reactor at high speed. When called upon, the automatic shutdown system overrides the normal control system and rapidly inserts all active rods into the reactor. The safety shutdown system is designed such that loss of electrical power will de-energize the scram solenoid valves and accumulator pressure will scram the rods.

Electrically, a rod can be placed into any one of five operating modes by means of switches on the rod selector panel in the reactor control room.

1. Safety - Rods assigned to this operating mode are reserved for the safety shutdown function only and cannot be used for control. At startup, all safety rods must be withdrawn completely before any other rods can be pulled for control purposes.

2. Manual - Rods assigned to this operating mode are used for reactor power adjustments and are run in and out of the reactor by assigning them to one of six manually-operated rod-gang switches on the reactor operator's console.

3. Setback - During reactor operation, if certain parameters are exceeded, an automatic reduction in reactor power is achieved by control rods which are also assigned to setback. These setback rods are automatically inserted into the reactor in a series of short steps on a setback signal.

4. Withdrawal - For maintenance purposes rods are assigned to this operating mode so they can be withdrawn from the reactor

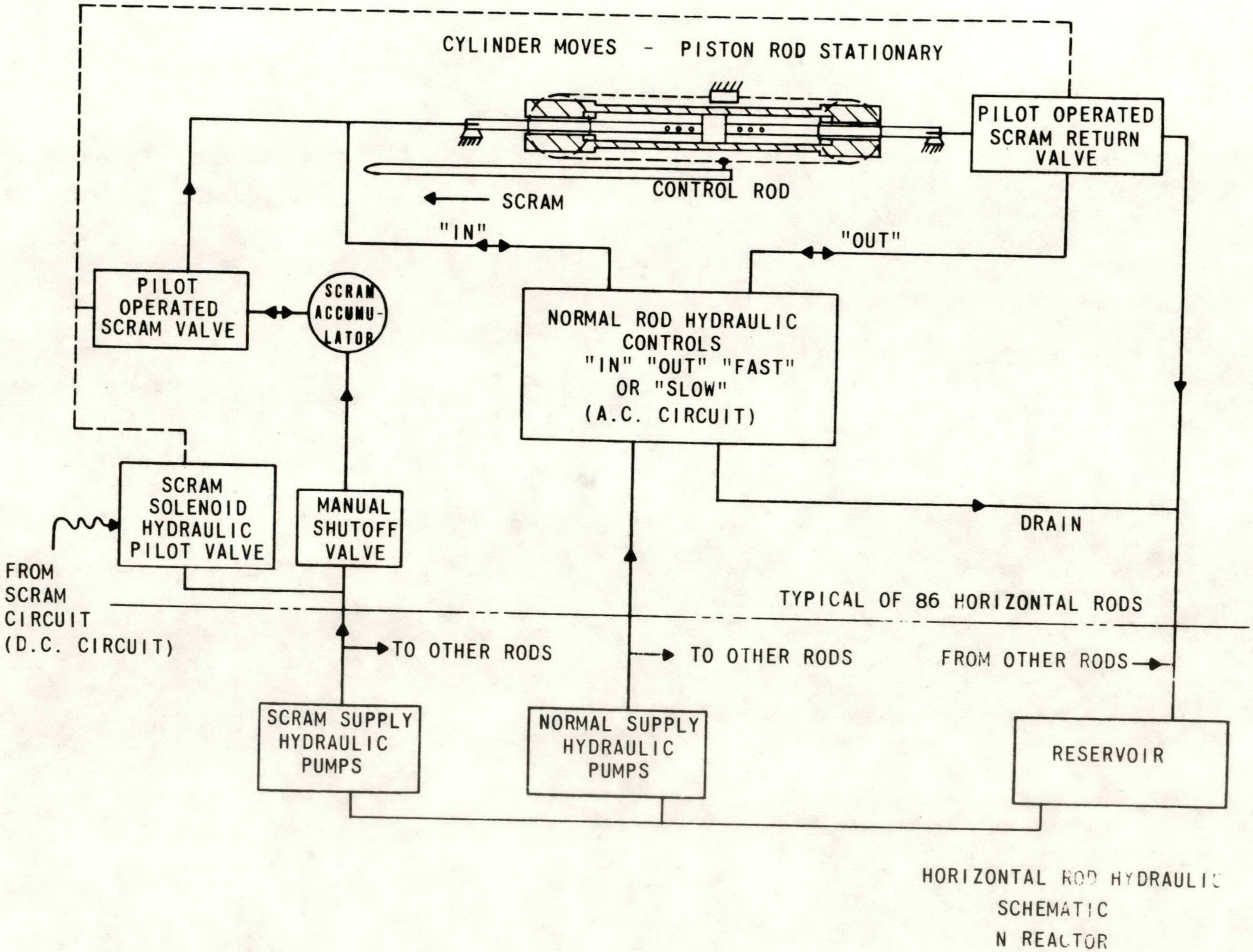


Figure 2. Horizontal Rod Hydraulic, Schematic, N Reactor

with or without the safety circuit tripped. In this mode, rods can be withdrawn by the normal control system but cannot be inserted into the reactor either by the normal control system or the scram system.

5. Off - This assignment is used to lock out a rod electrically so that the rod cannot be either withdrawn or inserted by either the normal or scram systems. Normally, the rod is full out when this assignment is made.

C. Ball Safety System

The ball safety system provides a completely independent nuclear control system to back up the rapid shutdown (scram) function of the horizontal rods. The control elements for this system are 3/8-inch diameter ceramic balls containing a neutron absorber. Ball hoppers are imbedded in the top biological shield in a 32-inch square pattern over each of 107 vertical channels in the reactor graphite moderator stack. The hopper gates, held closed by battery-powered solenoids, are designed to open in case of control power failure, as well as upon appropriate safety circuit signal, to allow the balls to enter the channels by gravity.

Both the ball and rod safety systems each have adequate strength to maintain the reactor subcritical in any credible accident situation. In addition, each system has a speed of insertion sufficient to prevent fuel melting under the most extreme accident conditions even if either of the systems should fail during a nuclear excursion.

D. Flow Monitoring

The inlet of each process tube is equipped with a venturi and instrumentation for monitoring coolant flow on a tube-by-tube basis. Primary coolant flow information is indicated and recorded as necessary for operational control purposes. In addition, reactor shutdown signals are initiated by flow rates for individual fuel channels which are found to be either below or above pre-determined limits. The flow monitors for these critical points are all designed to have response times sufficiently short to insure safe reactor shutdown.

The outlet assembly for each process channel includes equipment for monitoring outlet water temperature and sample points for the fuel element cladding failure detectors. Each outlet connector has a remotely-operated three-way valve which permits diversion of each tube's outlet water from the recirculating cooling stream to a contaminated water disposal system to minimize spread of radioactive materials through the primary cooling loop in event of a fuel element failure.

Individual process channel outlet temperatures are indicated and recorded for operational purposes and to back up the nuclear instrumentation in detecting excessive local power density in a particular region of the reactor.

E. Nuclear Instrumentation

Nuclear instrumentation measures the level, the distribution, and the rate of change of neutron flux density in the reactor. Monitoring data and/or safety circuit trip signals are provided, as appropriate, over the full range of neutron flux levels from those existing during shutdown conditions to those characteristic of full production level operation.

It is a significant factor, in regard to nuclear safety, that the NPR lattice is designed so that reactivity is decreased by any loss of coolant from the process channels.

F. Confinement System

The primary confinement zone which surrounds the entire primary cooling system is designed to maintain structural integrity with internal pressures ranging from +5 psig to -2 psig. Vents are designed with adequate area to dissipate the maximum rate of steam evolution without exceeding these internal pressure limits and with closures capable of resealing the vent openings within the time interval between dissipation of the steam surge and initiation of melting of the reactor fuel. In addition, a fog spray system would condense the steam vapor to reduce the pressure within the primary confinement zone and also remove a significant fraction of the contamination from the confined air. Controlled release of the confined gases is through a filter system designed to remove a high percentage of the particular matter as well as a substantial fraction of the radioactive halogens.

G. Water Supply System

Untreated water from the Columbia River is supplied to the dump condensers as well as to the condensers on the primary-drive turbines and on the reactor local turbine-generator. This condenser cooling water is then returned to the river. Untreated water is also supplied to a water treatment facility to provide plant requirements for filtered, sanitary, and demineralized water.

H. Emergency Cooling System

A separate, independent, untreated water system is provided for emergency once-through cooling of the reactor. Two diesel-driven pumps located in the river pump house supply this water to a 200,000 gallon storage tank and to the suction of the three diesel-driven high-lift pumps which are provided to deliver the cooling water to the bottom of the front face headers, where it may enter the primary cooling system, through check valves, when the system pressure falls below a specified level. This emergency water supply is intended for use only in case the normal recirculating treated water-heat removal system is disabled by loss of all sources of pumping power or by a major rupture in the primary cooling system piping. The 200,000 gallon emergency water storage tank can also serve as the water source for the confiner fog spray system as well as for emergency coolant supply for the moderator cooling system.

III. INCIDENT DESCRIPTION

Final pre-startup checks for N Reactor, per Mandatory Check List 48-145-000-390-101, were begun on September 29, 1970. All pre-start checks, including a functional scram insertion test of three randomly selected control rods were accomplished without difficulty.

Following the successful completion of pre-startup checks, N Reactor startup was initiated on September 30, 1970, at approximately 1:20 a.m. Full withdrawal of the 32 neutron poison rods assigned as safety rods was then begun as required prior to withdrawing the control rods and beginning the approach to reactor criticality. Safety rod withdrawal in small groups of five or six proceeded normally until withdrawal of the last group of safety rods was attempted. At that time, rod No. 59 would not respond to withdrawal commands from the reactor operator's console in the 105-N control room. Startup was stopped after all safety rods were withdrawn with the exception of rod 59.

Immediate troubleshooting indicated that the inability to extract rod 59 was apparently due to failure of a hydraulic component in the rod drive mechanism. To avoid a delay estimated at two to four hours in the continuation of reactor startup, the shift manager made the decision to physically withdraw the rod from the reactor and designate it out of service as permitted by Process Standards.

An adjacent rod, No. 58, was assigned to safety as a substitute, and rod 59 was assigned to "off" with the rod assignment switch on the rod service selector panel in the 105-N control room. The reactor physicist covering the startup was consulted and concurred with the aforementioned action. Reactor startup was resumed.

Initial indication of criticality was noted at 2:22 a.m. The reactor power was held at a nominal 450 mw_{th} to accomplish an inspection of 105 Building Zone 1. While the inspection was proceeding, normal pre-power ascension checks were successfully completed and the primary cooling system pumps were increased to the normal full speed of 3600 rpm in anticipation of the increase in reactor power. Flow monitor settings for full-flow operation were accomplished immediately following the flow increase.

At 5:22 a.m., an annunciator alarm was received, indicating that the vacuum of the No. 6 primary coolant pump drive turbine steam condenser had exceeded 7 inches of mercury absolute. (Normal running vacuum is 1 to 3-1/2 inches mercury absolute.) The turbine operator was attempting to place the auxiliary steam ejector ("hogger") in operation in an attempt to lower the condenser vacuum when the vacuum reached the trip point of 15 inches of mercury absolute and the No. 6 primary pump drive turbine was automatically shut down. The No. 6 primary pump electric pony motor assumed the load of driving the primary pump; however, the switch from the main drive turbine (3600 rpm) to the pony motor (900 rpm) created a flow transient which resulted in simultaneous low-flow trips on several process tubes on the right side (cell 6 side) of the reactor. With the flow monitor low flow trip points set for high-flow operation, the low-flow trips are the normal and expected result of a primary pump slowing down to pony motor drive speed.

The low-flow indications resulted in an automatic reactor scram (shutdown) signal at approximately 5:25 a.m. Reactor shutdown (subcriticality) was confirmed by the reactor neutron flux level and power level instrumentation. Unexpectedly, control room instrumentation also indicated that the neutron poison balls of the backup safety shutdown system had been released into the reactor. (See Table II for detailed event timing.) It was also observed that the neutron poison rod system, which normally functions to shut the reactor down, had not responded to the automatic scram signal. The 56 safety and control rods which were out or partially out of the reactor at the time of the "scram" remained in their "pre-scram" positions. This resulted in the automatic activation of the backup ball safety system. All other scram functions appeared to have occurred normally. (See Table III.) Since the power and flux instrumentation clearly indicated that the reactor was safely shutdown by the ball safety system, the decision was made not to insert the control rods manually to avoid the possibility of disturbing evidence which might help in determining the cause of the problem.

The reactor physicists confirmed the ability of the ball safety system to adequately hold the reactor subcritical. The ball safety system provides a safety factor of approximately two in maintaining the reactor subcritical in a worst case, cold, xenon free condition.

N Reactor management personnel were informed of the situation, and instructions were confirmed to maintain the plant in its existing condition to insure that no evidence related to the cause of the rod problem would be lost.

An inspection of Zone 1 was made immediately after shutdown. All ball hoppers were visually inspected to confirm the number of hoppers dumped. Although control room instrumentation had indicated that ball hoppers 3, 7, 19, and 48 had not dumped, the visual inspection revealed that only hopper No. 3 had not dumped. Ball hopper No. 40 had been taken out of service for maintenance prior to startup and, therefore, all but two out of 107 hoppers of balls were in the reactor providing a shutdown safety factor of approximately two.

An inspection of outer rod rooms confirmed that the 56 safety and control rods were still maintaining their pre-scram positions. All rod control hydraulic systems valve positions and pressures were found to be normal. In the event of a scram signal, circuit breakers 1K1 and 1K2 are designed to open and de-energize the individual scram solenoids on all rods. When de-energized, the scram solenoids release high-pressure hydraulic fluid from the scram accumulators which produce a high-speed insertion of all

TABLE II
PRE-SHUTDOWN EVENTS

Time	Date	Action
1250	9/27/70	Shutdown from 4000 Mw_{th} and 811 Mw_E
0129	9/30/70	Started withdrawal of safety rods. (A total of 32 rods were assigned to safety.)
0150	9/30/70	Experienced difficulty pulling rod 59.
0200	9/30/70	All safety rods out except 59.
		Obtained approval from physics group to substitute 58 rod for 59 rod as a safety rod.
		No. 59 rod withdrawn by hand from reactor.
		Assigned 59 rod to "off" in control room and assigned 58 rod as safety rod.
0211	9/30/70	No. 58 rod pulled to complete safety rod withdrawal.
0222	9/30/70	First indication of rising period (46 + rods withdrawn).
0255	9/30/70	150 Mw_{Th} level reached. Holding power level for system heat up.
0345	9/30/70	Increasing power level.
0415	9/30/70	Reached 450 Mw_{Th} power level. Hold 450 Mw_{Th} power level while leak inspection was being conducted.
0420	9/30/70	Raised high en masse trip on flow monitor in preparation for turbine speed and flow increase.
0425-0445	9/30/70	Increased turbine speed to 3600 rpm.
0450	9/30/70	Raised low en masse trip on flow monitor.

TABLE II
(Cont'd)

Time	Date	Action
0450-0522	9/30/70	Holding stable power level and preparing for continued power increase.
		a. Checked all cells for oil leaks.
		b. Checked secondary makeup flow.
		c. Ran flow and temperature maps.
		d. Adjusted flux bar settings.
		e. Checked diversion system operation.
		f. Set zone temperature monitor.
		g. Reset power average range trips.
		h. Hand calculated power level.
		i. Obtained total control limits.
		j. Other duties in prparation for power ascension.
0522:06		No. 6 turbine trouble annunciator
		a. Operator going to inspect turbine.
0525:25		Turbine No. 6 tripped off to pony motor.
05:25:29.07		Low flow trip before delay
		a. Process tubes 1370, 0571, 0873, 1873, 2773
05:25:29.60		Process tube low flow
		a. Reactor scram signal initiated.
05:25:29.65		Rod scram
		a. Initiates reduction in speed of all operating primary pump turbines as well as other post-scram actions.

TABLE II
(Cont'd)

Time	Action
05:25:29.65	Reactor critical after scram.
05:25:30.61	182 Diesels start.
05:25:31.23	Slow rod in any column.
05:25:31.23	Ball drop - 3 slow rods.
05:25:31.32	Any ball hopper gate open
	a. 1/16-inch movement of fastest gate.

TABLE III

STATUS OF REACTOR FOLLOWING SCRAM

Normal

- Power level zero verified by neutron flux monitors, primary loop temperatures, and power calculator.
- All post-scram actions functioned.
  - Primary pump drive turbines reduced speed.
  - High-pressure injection pumps responded.
  - Pressurizer heaters shut off.
  - Steam generator bypass valves opened.
  - All boiler feed pumps operated.
  - After-heat removal pumps operated.
- Rod safety circuit relays tripped (1K1 - 2K1).
- Emergency cooling diesels operating.
- Low-flow annunciation received on flow monitor.

Abnormal

- Rods did not scram
- Ball backup to rods circuit tripped. (Normal if rods fail to scram.)
  - Four hopper gate position readouts indicated failure to open.
- All ball hoppers tripped and balls dropped except hopper 3 - visual verification.
- Low vacuum annunciation of No. 6 drive turbine.
- No. 6 primary pump on pony motor drive.
- Low-flow process tube scram.

rods into the reactor (see Attachment A, Drawing H-1-27993). Inspection of the visual indicators on the 1K1 and 1K2 safety circuit breakers in cabinet PR-27 verified that they had tripped as required when the scram signal was received. The scram solenoids were found to be warm to the touch, indicating that they were still electrically energized. The rod scram solenoids were apparently being held energized by an extraneous power source.

At this time all activities in reactor areas which might be involved with the scram problems were stopped and the areas placed under special access control conditions in anticipation of the initiation of a formal investigation.

O. C. Schroeder, Acting DUN General Manager, appointed Dr. C. W. Kuhlman, DUN Vice President and Technical Division Manager, to direct the activities of an appropriate portion of the Working Committee of the President's Review Council and accomplish the official DUN investigation of the incident. The Working Committee consisted of the following DUN personnel as members:

CW Kuhlman, Vice President, Technical Division
GC Coleman, Manager, Quality Assurance
RE Dunn, Manager, N Plant Section
RT Jessen, Manager, Engineering Section
VV Johnson, Manager, Manufacturing Engineering Section
RW Reid, Manager, Process Section
RK Robinson, Manager, Technology Section

Prior to assuming his role as Working Committee Chairman, Dr. Kuhlman assured that the existing condition of the reactor was acceptable from a nuclear safety standpoint and documented that fact in a letter to the Acting DUN General Manager.

The Working Committee was charged with the following responsibilities:

1. Plan and execute an investigation which will identify causes and reasons for the failures noted in the incident.

2. Review appropriate corrective actions and proposed modifications to assure correctness and adequacy.

3. Attest to nuclear safety adequacy of N Reactor for startup.

It was pointed out that during the period of the investigation the Working Committee would have prior review and approval over those activities occurring within N Plant which might obscure evidence related to the incident.

Initial presentations to the Working Committee by Operating and Engineering personnel summarized the existing information on the scram incident and the present status of the reactor. The functioning of the reactor systems and safety circuits appeared to be normal with the previously described exceptions.

As a result of the initial briefings and in order to proceed as rapidly and thoroughly as possible, the Working Committee appointed special teams of senior technical and operational personnel to investigate each of the problem areas in detail to determine and provide to the Working Committee the extent of the problems, the problem causes, and the recommended corrective actions which should be taken. All actions of the investigating teams which might affect reactor hardware systems were authorized by written procedures or other documentation approved by the Working Committee. All teams were cautioned to be especially watchful for items which might indicate any relationship between the separate problem areas.

The problem areas singled out for investigation by the special teams were:

1. The failure of the rod safety system rods to scram into the reactor in response to the scram trip.

2. The No. 6 drive turbine condenser which lost vacuum and initiated the scram.

3. The failure of ball hopper No. 3 to function and dump its neutron poison balls into the reactor.

In addition to the areas where hardware problems were actually experienced, special teams were also formed to:

4. Independently review the N and KE reactor safety systems from the electrical, mechanical, operational and testing aspects and identify any failure modes that exist which could disable all or significant portions of any safety system. Provide proposed corrective actions to the Working Committee for any problem areas detected.

The following report sections cover the detailed findings, conclusions, and recommendations in each of the areas of investigation.

IV. ROD SAFETY SYSTEM SCRAM CIRCUIT INVESTIGATION

A. Chronology

Startup was initiated on the 12-8 shift on September 30, 1970, at 0129. In the startup sequence, No. 59 rod was the twenty-sixth of the 32 rods assigned to safety and, as such, was in the last group of rods in the safety rod withdrawal sequence.

When the attempt was made to withdraw rod 59, it would not respond to commands from the reactor operator's console. After all safety rods, except No. 59, were withdrawn, the startup procedure was stopped. Attempts to withdraw rod 59 using alternate rod control gang switches 1, 2, 3 and 4 were unsuccessful.

The shift manager, along with Maintenance, Radiation Monitoring, and Operations' personnel went to the left outer rod room to determine the cause of the non-operation of rod 59. A review of the hardware on the rod indicated that the lines from the "out" solenoid and "dual-pilot check valve" were warm, and leak-through of oil could be heard. Leakage through either of these components would permit oil to return to the reservoir rather than to the rod withdrawal cylinder and would account for the rod not withdrawing. (Reference: Drawing H-1-27993, Attachment A.)

Since changeout of either component would require some time, the shift manager decided to withdraw rod 59 from the reactor, assign it to "off", substitute another rod as a safety rod, and proceed with the startup.

Physics' approval was obtained, and the rod was withdrawn from the reactor by shutting the normal and scram hydraulic fluid supply valves, bleeding off the pressure in the rod 59 system, disconnecting all electrical connections to the rod control solenoid valves, opening the bleed valves at each end of the actuating cylinder and then pulling the rod by hand.

A metal blank was placed in the "75% In" limit vane switch to remove rod 59 from the slow rod counting circuit. Then, rod 58 was assigned as the substitute safety rod, and rod 59 was assigned to "off". The log indicates that all safety rods were withdrawn by 0211.

After startup resumed, the dual-pilot check valve was replaced, but the operability of the rod was not tested at that time in order not to conflict with normal startup activities. The rod selector switch was left in the "off" position.

As a result of shutting the down rod 59 hydraulic system, the "Low Scram Accumulator Pressure" annunciator was lighted on the local rod room panel and the "Rod Room Trouble" annunciator was lighted in the reactor control room. An unsuccessful attempt was made to extinguish the annunciators by disconnecting the cannon plug for rod 59, located on the bottom of the scram accumulator meletron pressure switch. When reconnecting the cannon plug, an audible arcing was noticed, and power was lost to the local annunciator panel. Breaker CH 15 (24 VDC) was determined to be tripped. This was about 1.5 hours after the failure of rod 59 to withdraw was first observed. Breaker CH 15 provides only annunciators and other auxiliary systems. This arcing incident was apparently not related in any way to the subsequent failure, but only serves to illustrate that accidental momentary shorts can be created during the connection and disconnection of electrical plugs.

Critical jumper 194 was installed to clear the annunciators, and the system was considered normal; reactor power ascension resumed at 0345.

The reactor power level reached 450 mw_{th} at about 0415 and was held at that level to permit a planned visual inspection of 105 Building

Zone 1. While this inspection was being accomplished, the reactor received an automatic shutdown (scram) signal from the primary coolant flow monitoring system.

When the reactor scrammed at 0525, the following events related to the rod scram circuit occurred at appropriate time intervals:

1. The rod scram annunciator came on.

2. Annunciators indicated slow rods on reactor scram.

3. Annunciators indicated that one or more ball hopper gates had opened.

4. Flux monitoring and power calculating equipment indicated zero power.

The detail chronology of the scram sequence was previously shown in Table II.

Visual checking indicated that all in-service ball hoppers had dumped except hopper No. 3. Detailed information on the hopper No. 3 failure is provided in Section VI. Ball backup to rods column lights showed slow rods in every rod column, indicating that the ball drop was initiated by the ball backup to rod circuitry. A check of the rod positions indicated that 56 rods which were out or partially out of the reactor at the time of the scram were still out. The scram pumps also had not started (which is normal if the rods do not scram and cause the hydraulic pressure to drop low enough to start the pumps). The reactor console operator, after assuring that the reactor was safely shut down, asked if he should run the rods into the reactor. The control room supervisor, after reviewing the information available to him, decided that the reactor was safe and not to run the rods into the reactor. (At this time, rods could have been inserted by use of either the setback circuit, the all-rod insertion pushbutton or the manual gang switches.) This decision was shortly thereafter ratified by the shift manager, the Operations Manager, and the N Plant Section Manager. The reactor physicist confirmed that the ball system was capable of holding the reactor subcritical by a safety factor of approximately two. A brief inspection of the major affected areas of the plant was then made.

1. The left rod room was entered. All rods were verified to be in their pre-scram positions. The rod scram solenoids were warm to the touch, indicating that they were still energized. Rod scram hydraulic pressures were indicated to be normal.

2. Visual position indicators showed that both rod scram breakers, 1K1 and 2K1, were open (e.g., de-energized).

3. The main ball safety system circuit breakers were found to be still energized, further indicating the ball drop occurred due to activation of the ball backup to rods auxiliary circuit.

As previously noted, a special team was formed of senior engineering personnel to investigate the rod circuit problem and determine the actual cause and extent of the problem.

Review of the circuitry design indicated that the most probable cause of the rod scram failure was the continuing energization of the rod scram solenoids by the auxiliary circuit which was used to electrically put control rod 59 in the "off" condition prior to the scram. For this to occur, however, shorting failures would be required of four diodes in series (DUN P/N 59D9, GE P/N AJA420BX8) which are specifically included in the control rod circuits to prevent power from the rod "off" or "withdrawal" circuits from inadvertently energizing the rod scram solenoids. (Reference: Figure 3 and Drawing H-1-32025).

The first checks were aimed at determining whether the theory of the diodes shorting was plausible. The breaker feeding the rod off and withdrawal circuit was visually checked. It was closed and warm to the touch. The four diode quad pack 59D9 was visually checked; it was observed to be cracked along the side and the top label was slightly raised. A bead of solder was extruded through the side crack. (See Figure 4.) The diode pack was not removed at this time in order to permit additional testing.

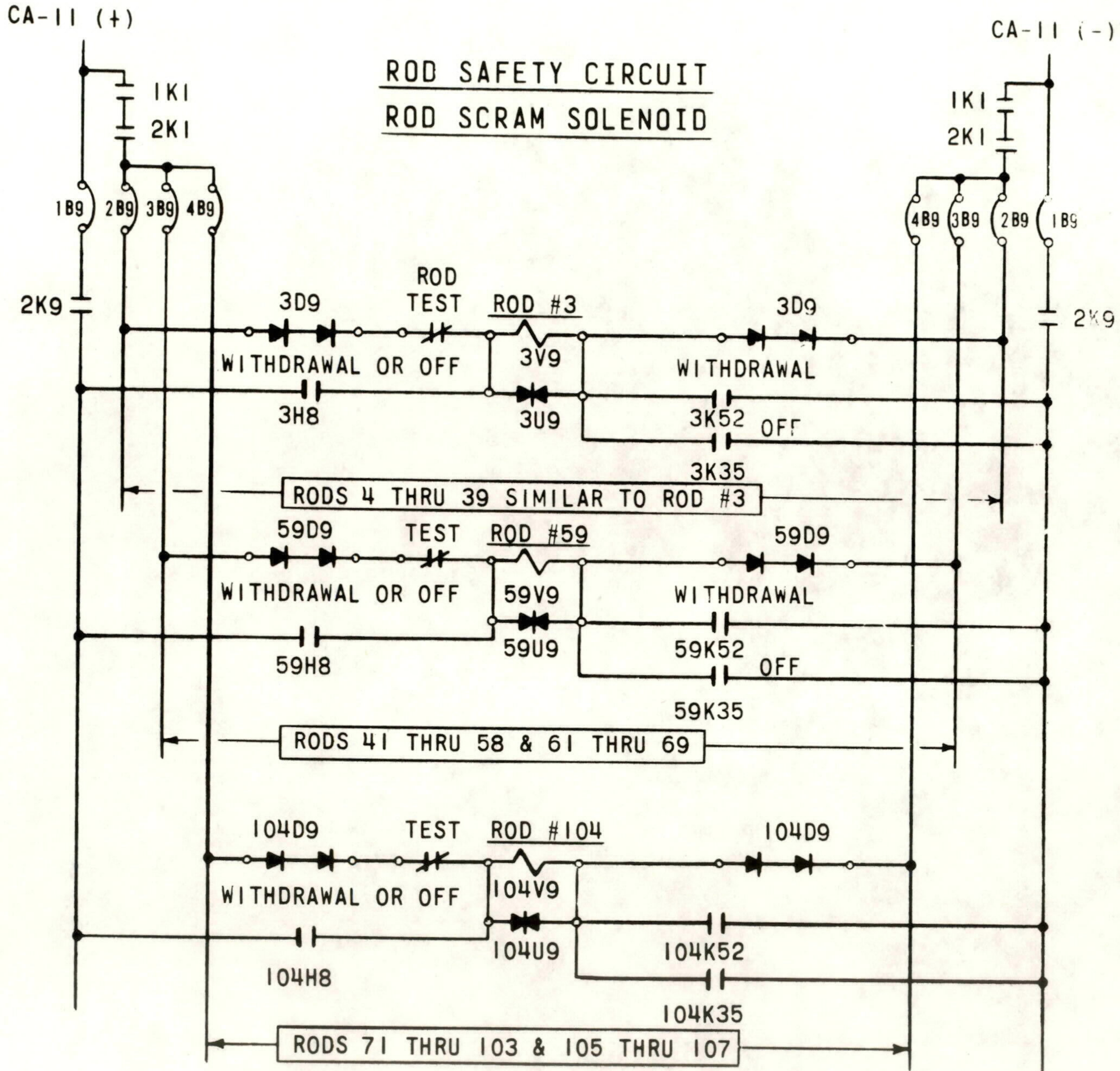
The voltage measurements across various scram solenoid coils indicated that coil 59 was reading the highest at 95 volts, and that the other coil voltages were lower depending upon location. (See Figure 5.)

Voltage measurements were then made at various points in the rod scram circuit, confirming that the main contacts of the rod safety circuit contacts were open and that no wiring bypassed these contacts.

Voltage checks were then made across various diode groups. These voltage measurements indicated that the diode group on rod 59 had approximately the same voltage drops as other diode groups, but was of the opposite polarity. This confirmed that current was flowing backwards through 59D9.

The current being drawn through the 1B9 circuit breaker feeding the rod "off" and "withdrawal" circuit was measured with a clamp-on ammeter at 10 amperes (breaker trip setting 12.5 amps), again indicating that multiple coils were being energized through this circuit.

Additional voltage and amperage measurements were made by the investigating team and further verified that the all-rod scram solenoids were being held energized by the rod "off" or "withdrawal" circuit being fed through the four shorted AJA420BX8 diodes, the 59K35 relay "off" contacts, and the 59H8 selector switch contacts of rod 59. (Reference: Figure 6.)



D9 IS THE FOUR INDIVIDUAL DIODES ENCASED IN ONE PLUG IN DEVICE.
V9 IS THE ROD SCRAM SOLENOID.
U9 ARE THE ROD SCRAM SOLENOID SURGE SUPPRESSORS.
1K1 AND 2K1 CONTACTS CLOSED WHEN SAFETY CIRCUIT IS ENERGIZED.
2K9 CONTACTS CLOSED WHEN EITHER LESS THAT 5 BALL HOPPERS LOCKED OUT OR WHEN LESS THAN 4 RODS WITHDRAWN FROM FULL IN POSITION.
H8 ARE SB-1 SWITCH CONTACTS CLOSED IN OFF OR WITHDRAWAL.
K52 ARE RELAY CONTACTS CLOSED WHEN ROD IS SELECTED FOR WITHDRAWAL.
K35 ARE RELAY CONTACTS CLOSED WHEN ROD IS SELECTED FOR OFF.

Figure 3. Rod Safety Circuit, Rod Scram Solenoid

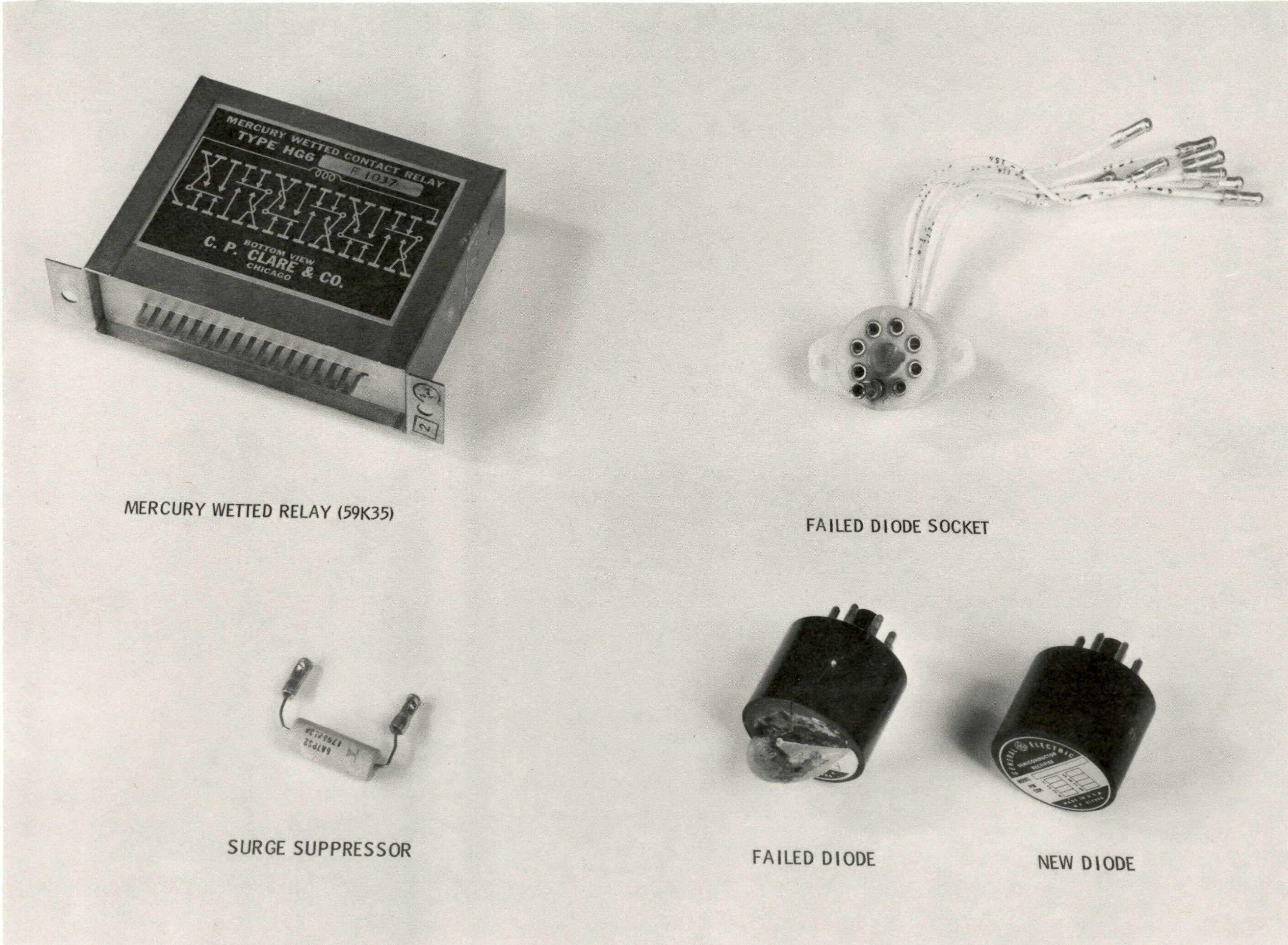


Figure 4. Rod Control Components

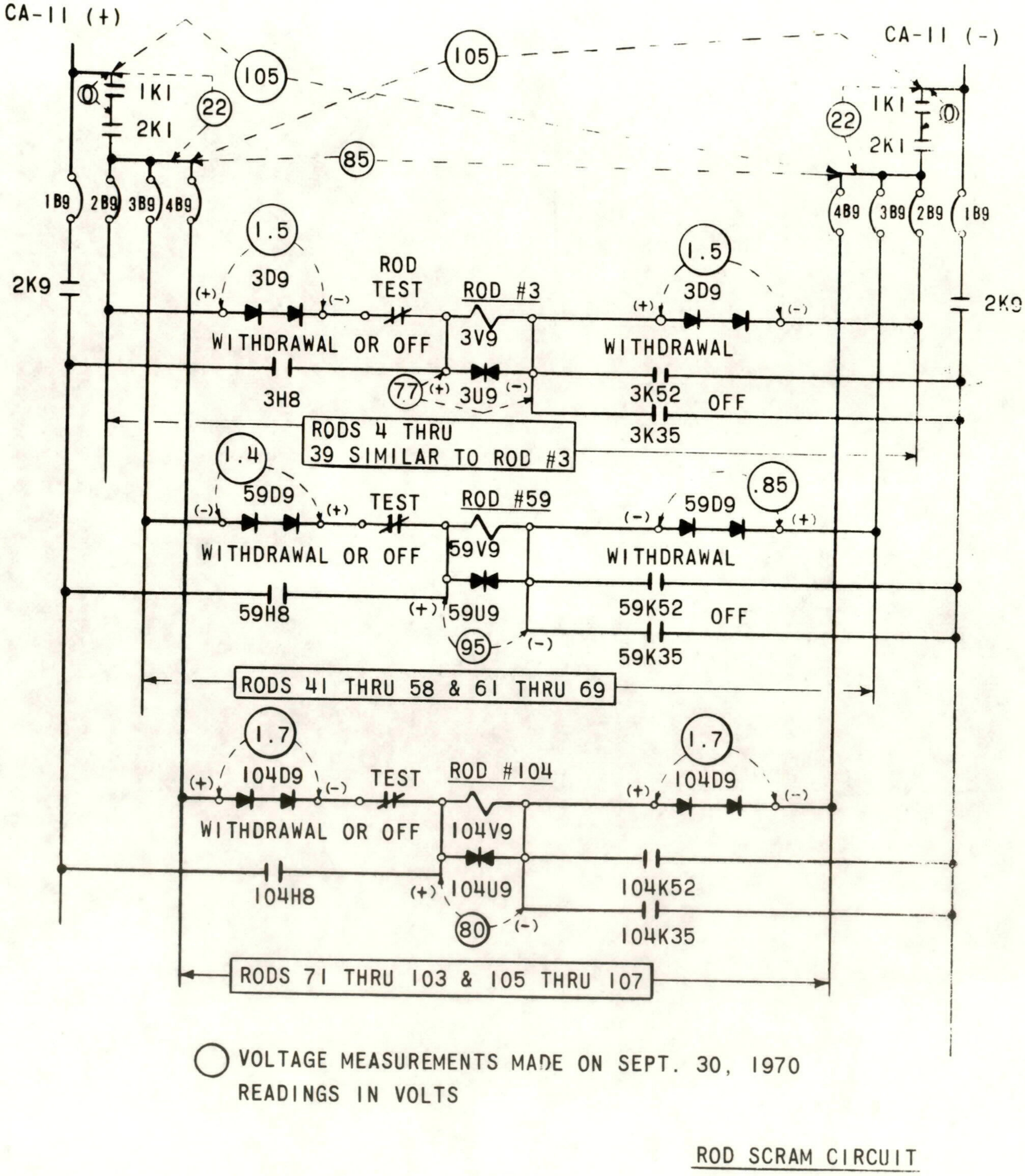


Figure 5. Rod Scram Circuitry
(Showing Voltage Measurements)

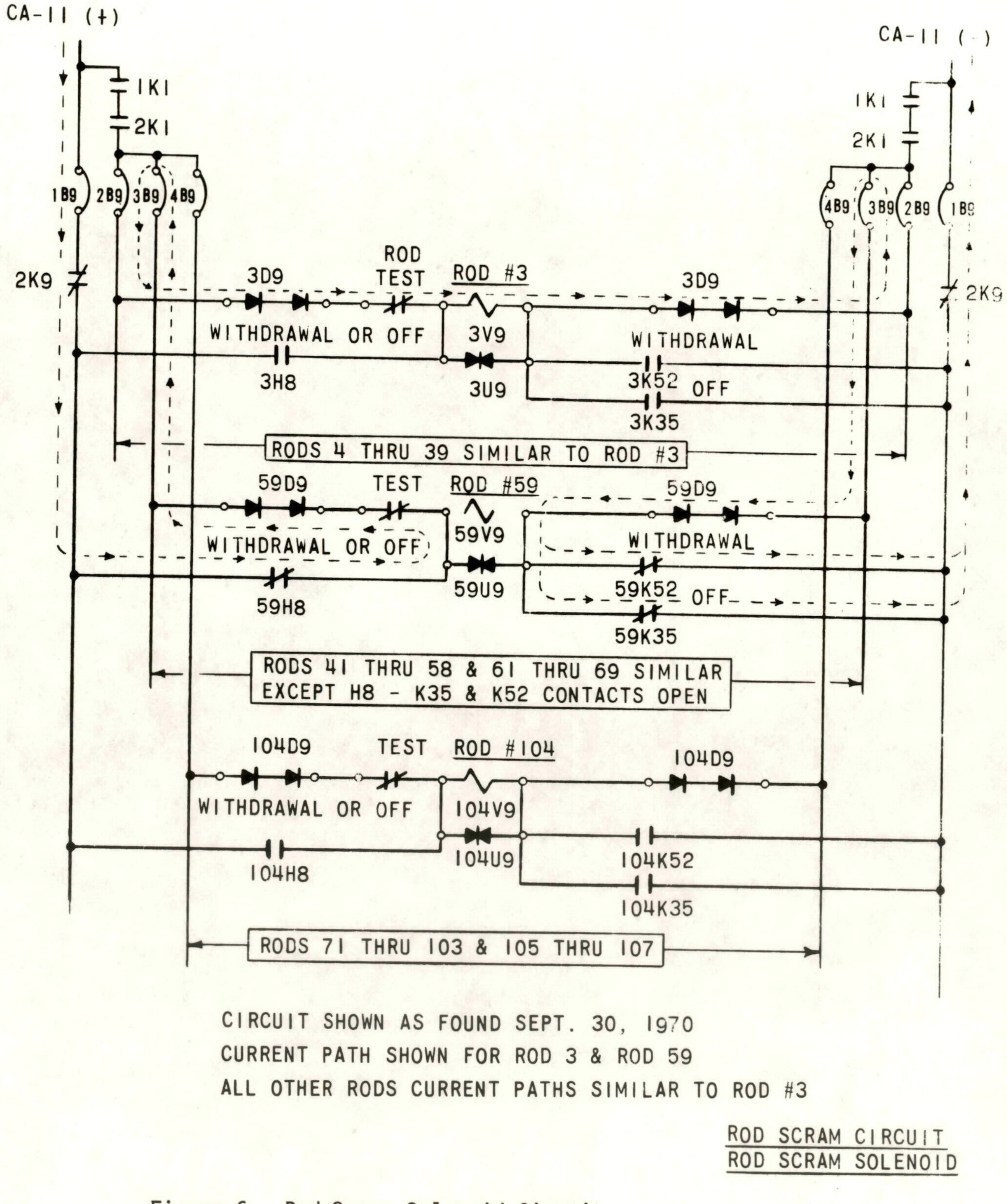


Figure 6. Rod Scram Solenoid Circuit
(Showing "Sneak" Circuit)

With the circuits remaining in the original post-scram condition, tests were run to verify that the capability did exist and had existed all along to insert the control rods using various combinations of the gang switches. All but six control rods were successfully inserted into the reactor using the six-gang switches. The successful insertion of the control rods, using the gang switches, also essentially verified that the capability also existed to insert all rods simultaneously using the "all-rod insertion" pushbutton and to insert rods assigned to "setback" using the "rod setback" switch. However, in order to retain six rods out of the reactor for a scram test, the "all-rod insertion" button and "rod setback" switch were not used.

During the testing, it was determined that rods could be withdrawn as well as inserted by using the gang switches. Withdrawal capability should not have been electrically possible with the reactor in a scrammed condition and, therefore, the cause of this discrepancy was traced down to determine its connection, if any, to the rod scram problem. An unauthorized jumper was found across the contacts of relay 56K1 which was the interlock which should have disabled the normal rod withdrawal AC circuits at the receipt of a scram signal. (See Figure 7.) It should be noted that, under normal scram conditions, the scram vent valve opens and bypasses any hydraulic fluid coming from the "out" control solenoid and makes it hydraulically impossible to withdraw a rod during a normal scram. (See Figure 2 or Attachment A.) This fact would also prevent the jumper from being noticed during normal testing.

A review of available records was not able to determine why or when the jumper had been installed. The jumper was later removed, and the system tested normally. It was concluded that the 56K1 jumper was not related to the rod scram problem and had probably been in place since reactor construction.

Following the completion of the previously described rod control tests, the rod 59 diode package (59D9) was manually pulled from the circuit, as a positive demonstration of the fact that the power holding the scram solenoids was coming through the diode package. All six remaining control rods immediately scrammed into the reactor. Circuit measurements also confirmed that all extraneous voltage had been removed from the scram solenoid circuits. Diode 59D9 was found to be very hot to handle and was very difficult to pull. In fact, one of the socket contacts was pulled out during removal. This socket contact contained solder from the diode on its inner surface.

The diode resistance was measured and the reading on each diode was essentially zero in both directions. It was also verified that there was no continuity or short between the four diodes. (They were all electrically separate.) The surge suppressor 59U9 was checked for resistance and found normal.

All of the quad diode packs for the remaining rods were checked at both 28 volts and 125 VDC reverse voltage. All four diodes in rod 38 diode

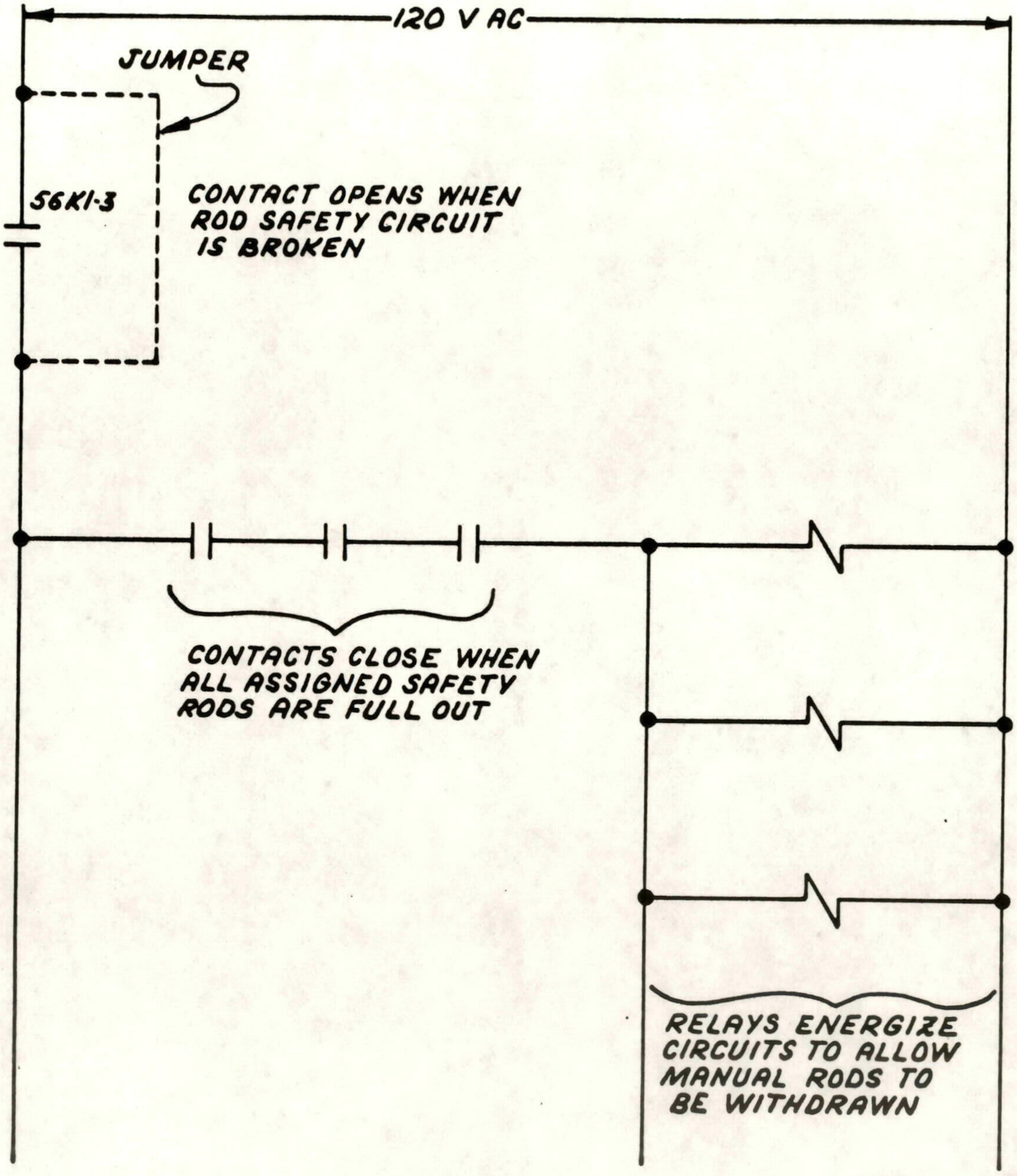


Figure 7.

MANUAL ROD WITHDRAWAL PERMISSIVE

pack 38D9 were also found to be shorted, and two diodes were found shorted in rod 44 diode pack 44D9, one on each side of the rod 44 scram solenoid coil. All other diodes were normal.

The following checks were made in preparation for returning rod 59 to normal.

1. The flexible connector to the rod scram coil was removed and dissected; no problems were found. Insulation resistance was infinite.

   Note: When the connector was being reinstalled following disassembly, the electrician's screwdriver shorted across the two coil wires, tripping breaker 1B9. Later, it was shown that this type of fault could short all four diodes in a quad pack.

2. The rod local control switch box was then installed at rod 59 station. Valving was returned to normal. It was verified that the rod could be run in and out without problem, indicating that the replacement of the dual-pilot check valve had corrected the rod 59 failure.

3. Critical jumper No. 194 was removed so that annunciation could again be restored to the rod 59 scram accumulator.

4. Rod 59 "off" relay (59K35) and "withdrawal" relay (59K52) were removed. A new relay was installed in 59K35 because the current rating of the contacts had been exceeded. 59K52 was inspected and showed no signs of damage. It was then reinstalled.

5. The wires from Room 6 to the scram coil were meggered and found to read 100 megohms (normal). The coil resistance and wire resistance was measured and found normal.

6. A new tested surge suppressor, quad diode pack, and diode pack socket were installed in rod 59 circuitry.

7. The safety circuit was then energized, and rod 59 was operated in the manual position by in, out, fast, and slow commands from the control room. Operation was normal.

8. Rod 59 was then assigned to "off". The safety circuit was broken, and voltage measurements were taken. Rod 59 scram coil measured 125 VDC, and the remainder of the coils read zero volts.

A test was then run on rod 38 with its four shorted diodes still in place. The safety circuit was made up. All rods were assigned to manual. Rod 38 was then assigned to "off" and the safety circuit was tripped. The rods did not "scram" into the reactor, and voltage measurements indicated that all scram solenoids were being held energized through the faulty diodes of rod 38. Diode package 38D9 was pulled, and all scram solenoid voltages (except for rod 38) dropped to zero; thus, the failure through rod 59 had been reconstructed through the shorted diodes of rod 38.

At the completion of all rod circuit testing, the following components had been replaced in the rod scram circuits:

1. Rod 59 scram solenoid pigtail cable (to permit examination and dissection of original cable).

2. Rod 59, 38, and 44 diode packs (P/N AJA420BX8).

3. Rod 59, 38, 44 and 42 surge suppressors (rod 42 surge suppressor was good, but marginal, as a surge suppressor) (ITT P/N 8A7PS2).

4. Rod 59 and 38 "rod off" relays 59K35 and 38K35.

The following components were examined and functionally tested, but not replaced.

1. 1B9 breakers.

2. H8 switch on rods 59 and 38.

3. K52 relays on rods 59 and 38.

4. Rod 59 and 38 scram solenoids (meggered).

B. Failure Analyses

1. Hypotheses

Although the testing verified that the shorting failure of the four rod 59 diodes provided the path for the holding voltage, the cause of the shorting failure of all four of this rod 59 diodes remained to be determined. Several hypotheses existed.

a. Normal lifetime of the diodes. The total of ten diode failures detected in the circuitry of rods 59, 38, and 44 after 21,000,000 diode hours of operation demonstrates a reliability reasonably close to the industrial data on diodes of .75 failures per million unit operating hours. The non-random distribution of the failures, however, would cast doubt on the assumption that these failures resulted from normal wear-out. It is of interest to note that the demonstrated failure rate of the total quad was two failures in five million operating hours. (Reference: Attachment C.)

b. A short-to-building ground on both sides of the rod 59 scram solenoid could cause a current flow which could exceed the peak ampere rating (15 amps) of the diodes long enough to result in a simultaneous shorting failure of all four diodes. This double-ground failure should have been detected by the automatic ground detectors on the rod circuitry. In addition, the circuit breaker feeding the rod circuitry is a 10-amp breaker; however, it is of a time delay type and might not trip if the grounds were of a short enough duration. The wiring to the scram solenoids of rods 59 and 38 were meggered at 500 V in an attempt to detect any insulation breakdowns or other intermittent grounds. None were found.

c. A shorted surge suppressor (59U9) could have permitted sufficient current flow through the diodes to short them. A check of all rod scram solenoid surge suppressors using a Textronix type 575 curve tracer with 400 VDC (+) and 100 VDC (-) detected no shorting type failures. The surge suppressor on rod 42 indicated marginal performance as a surge suppressor and was replaced.

d. A shorted scram solenoid coil could have permitted sufficient current flow through the diodes to fail them. A check of the scram solenoid coils on rod 59 and 38 indicated normal coil resistance. Less than two solenoid coil failures are experienced per year, and these are normally due to bending or failure of the male pins at the cable connection point, and <u>not</u> coil shorting.

e. Accidental shorting of the plus side to the minus side of the normal scram solenoid circuit such as by insulation failure in the scram solenoid pigtail cable, or oil, water, or other contamination in the pigtail cable or scram solenoid connector, or accidental shorting during maintenance or EMS checks could cause the diode failures. The pigtail cable on rod 59 was inspected, meggered, and then physically dissected with no problems detected.

   As previously noted, during the replacement of the rod 59 scram solenoid pigtail cable, a screwdriver was accidentally shorted across the cable terminals, resulting in the opening of the 1B9 breaker. (Reference Figure 3.) The 59D9 diode pack was not installed at the time so the effect on the diodes could not be determined; however, testing was accomplished during the component test program on the diodes which showed that a screwdriver accident such as this could short all four diodes even before the circuit breaker could trip. (See Attachment D.)

f. If the scram solenoid pigtail cable (DC) was accidentally plugged into one of the AC rod control solenoids <u>and</u> if a specific diode installed across the AC control solenoid was shorted, a path could be created which should provide sufficient ampere flow to fail the rod diodes. This would also cause the trip of the 10-ampere breaker. The AC solenoid diodes on rod 59 were tested and were not shorted.

2. Record Search

In an attempt to try to establish the time of the shorting failure of the diode quads, the history of the affected rods was researched with the following results:

a. No record could be found when these rods (38, 44 and 59) were assigned to off or withdrawal in the past.

b. All rods had been scrammed in from the full-out position for a routine three-month EMS check, following the controlled shutdown of the reactor on September 27, 1970. Two slow rods were detected. Those abnormal were corrected by adjusting the scram speed control valve.

c. The normal EMS startup checks were made prior to going critical including tests that the rod scram circuit would scram a rod.

d. No log entry was made to show that any work was done on rod 59 from September 27, 1970 to September 30, 1970.

e. No record could be found of any checks made of the four diode group in each rod's scram solenoid circuit. No EMS required these checks to be made.

3. Lab Experiments

Lab experiments were conducted to examine different failure mode effects. (Reference: Attachment D.)

Those examined were:

a. High reverse voltages.

b. High currents at the greater than rated current of .75 ampere of

1) 6-ampere long duration
2) 8-ampere long duration
3) 50-ampere short duration, and
4) >50-ampere long duration.

c. Simulated circuit without surge suppression.

The results of these experiments were as follows:

Cases 1 and 2 did cause failure of the diode junction. Case 3 did not cause a junction failure after 70,000 trip operations. Physical examination of the diodes failed in the lab were compared with those failed diodes removed from rod 38. Diodes failed at the lower currents and higher currents long duration exhibited a mottled look. Diodes failed at 50-amperes short duration appeared similar to the diode removed from rod 38. These diodes were dissected and had the shiny appearance of a new diode; for example, a diode failed this way could not be separated visually from a good diode.

In summary, all physical evidence examined in these tests indicates that the quad diode failures experienced in the rod scram circuit were the result of momentary shorting of these diodes across 125 VDC with the unlimited current being shut off in 20 to 50 milliseconds by the installed in-line circuit breaker.

4. Independent Review of Rod Safety System

As a result of the previously discussed problem, independent reviews of all critical safety systems were made to insure that no other similar problems existed. The criteria for these reviews and a discussion of the other systems reviewed are presented in Sections V, VI and VII of this report.

Since the components that precipitated the safety systems review were diodes, the rest of the rod control and safety circuits were specially reviewed for evidence of any other interconnecting diode circuits. None were found. In addition, nowhere was a location identified where a single short-to-ground could negate the protective action.

A review of the normal rod control system was made with particular emphasis on the all-rod insertion circuit since this single control function affected all rods. The Review Team was not able to identify any single component failure that could negate any of the rod scram functions. It was also positively established by the Review Team that the manual rod control and all-rod insertion functions would have been operable (except for five seconds following the scram signal) had the control room operator chosen to use them following the incident.

As a result of the review of rod system EMS, the recommendation was made to include an annual check of the integrity of the rod diodes.

Surge suppressors are used throughout the rod control and safety systems in the DC circuits to preclude the possibility of arcing across contacts and potentially welding them closed upon coil deenergization. Since these suppressors could fail open and not perform their protective function, it is conceivable that certain contacts in the system could fail. In the event of such a failure, many of these welded contacts could go undetected until the actual operation of all contacts on these relays is checked annually by Equipment Maintenance Standard 119. Equipment Maintenance Standard 119 was conducted during the recent extended summer outage. Since the timing is so recent, the executing of the EMS was judged by the Review Team to be adequate verification that the relays were functional.

Individual safety circuit trips are routinely checked before each reactor startup with the exception of the main steam header trip and the primary loop extremely low pressure trip, which are not included in the startup check sheet. In addition, the startup check sheet appeared to be deficient from the standpoint that it only relies on indicating lights for indication of rod system actuation. The Review Team recommended that the main steam header pressure and primary loop extremely low pressure trip be included in the future in this startup check sheet and that positive verification be made that the rod safety system circuit breakers actually operate. In addition, it was recommended that a complete safety circuit functional test, including these features, be conducted before startup.

A location in the circuit was identified where two shorts or grounds could develop and prevent the function of that system. These were determined to be a problem since the physical location of the two ends of these postulated shorts are less than 1/4-inch apart in the control panels, and the terminals involved are bare. A simple solution of installing insulated sleeving on the terminals was recommended by the Review Team.

C. Conclusions

The failure of the rod scram circuit to function correctly on September 30, 1970 was due to a sneak circuit caused by a shorting failure of four diodes in the rod 59 scram circuitry. When rod 59 was assigned to off, another source of power was placed into the rod scram circuit, allowing all scram coils to be fed through the sneak circuit (failed diodes) from the rod off and withdrawal power supply.

Failure of the diodes was probably some shorting failure (such as accidentally shorting the scram coil wires together) that caused a high current through all four diodes simultaneously. Such a failure mechanism has been demonstrated by laboratory techniques. The failure of the diodes could have occurred in any period since plant startup in 1963.

D. Corrective Action

A design review team was set up to provide corrective action for the failed diode problem. Personnel were represented from Technology Section, Engineering Section, Manufacturing Engineering Section, Process Section, and N Operations Section. The team came up with the following alternatives:

1. Remove the "off" and "withdrawal" circuits totally. This would eliminate the source of the problem and also permit the removal of the D9 diodes.

   Operations requested that the circuits be retained, but restricted to use only when the reactor was subcritical. It was initially felt that without the "withdrawal" circuit a rod which lost cooling water could not be withdrawn from the reactor following a scram fast enough to avoid rod damage. Without the withdrawal circuit, no rod can be removed following a scram until the safety circuit can be electrically restored to pre-trip conditions, which normally takes from 15 to 30 minutes. In addition, the "off" and "withdrawal" circuits would be of great convenience during outage maintenance and check-out work. It was noted that a rod could be held out of service by closing off appropriate hydraulic system valves.

   A study by Technology Section (reference: RL-GEN-1095, "N Reactor HCR Temperature Transients, 100% and 120% Power Level Operation," JD Agar, July 27, 1966, Confidential) indicates that even if a rod in the highest flux zone of the reactor totally loses its cooling water and reactor scram automatically occurs, as it should, within 60 seconds, no rod damage should result even if the rod is not withdrawn from the reactor.

2. Install two sets of redundant contacts of a key-lock manual switch in both the positive and negative leads supplying power to all the rods "off" and "withdrawal" circuits. Actuation of this switch and/or placing any rod in "off" or "withdrawal" would be prohibited by procedures and Process Standards except when the reactor is subcritical. A control room annunciator will be wired to indicate when the switch is turned on. The key to the switch will be controlled by the control room supervisor.

3. Install a voltmeter to provide a continuous monitoring of the positive and negative lines supplying power to the "off" and "withdrawal" circuits to insure that no power is applied to these lines from any source during reactor operation.

4. Install a push-to-test button to provide a means of assuring the proper functioning of the voltmeter described above.

   The voltmeter test circuit was rejected since it provided another possible source of inadvertent power to the "off" and "withdrawal" circuits.

5. Install automatic interlock circuitry actuated by the safety circuit which would not permit energization of the "off" and

"withdrawal" circuits unless the reactor is in a "scrammed" and subcritical condition. This could be accomplished by placing a set of contacts from 50K1 or 51K1 (five-second time delay after scram) in the power supply lines to the "off" and "withdrawal" circuits so they could be powered no sooner than five seconds after a scram.

6. Replace the 10-amp rated 1B9 circuit breaker supplying the "off" and "withdrawal" circuits with a 1- or 2-amp breaker so that even if this situation were to occur in some other way, the breaker would not be able to carry the load required to hold in all the rod scram solenoids (10 amps) and would trip off. This was rejected due to the desire to have a reliable power source for the rod and withdrawal circuits.

Installation of the key-lock switch and voltmeter, in conjunction with the appropriate procedural controls, was accepted by the Working Committee as adequate to permit a safe startup of N Reactor.

Design Change 3121 was installed, implementing Design Concept No. 2 and No. 3, listed above. (See Figure 8.) Operations procedures, startup checks, and Process Standards were implemented that forbid actuation of either the 1H9 bypass switch or any of the rod off or withdrawal selector switches during reactor operation. The voltmeter will be viewed periodically during each shift to insure that no voltage exists in the "off" and "withdrawal" circuit.

It was the consensus of the Committee that these should be considered only as interim solutions until a detailed engineering study of other alternatives can be completed. It was specified that final recommendations resulting from the engineering study should be presented to the Working Committee in January, 1971.

V. NUMBER 6 PRIMARY PUMP INVESTIGATION

A. Chronology

The investigation into the cause of the loss of the condenser vacuum which initiated the No. 6 drive turbine tripout and caused the September 30 scram revealed that there had been some difficulty in maintaining surface condenser vacuum during startup operations following the extended summer outage. The use of the auxiliary ejector (hogger) was required at times in addition to the normally used air ejectors. During a previous investigation of the problem (September 19, 1970), a tube leak was detected in the inter-after condenser on No. 6 turbine and has since been repaired. As a part of pre-startup procedures, checks were performed satisfactorily on September 29, 1970, with the 6B condensate pump in-service and one set of steam jet air ejectors in-service. One of the checks was a test of the low vacuum trip that is run before startup from any extended outage. After the test, the hogging ejector was secured, and the system appeared normal.

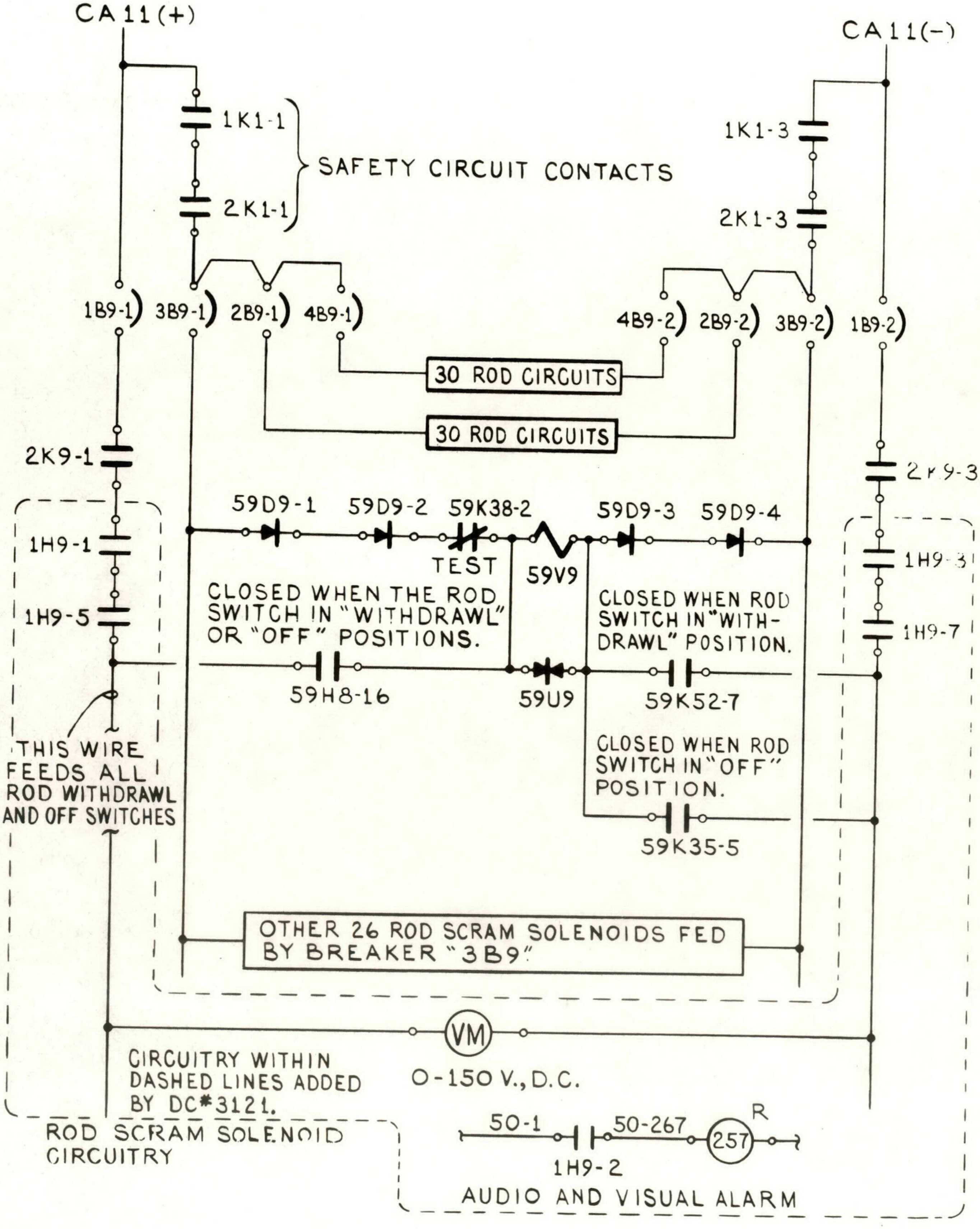


Figure 8. Rod Scram Solenoid Circuitry
(Showing Design Change 3121 Changes)

Just prior to the turbine tripout, the operator working next to the turbine was alerted to a reduced vacuum situation by the "drive turbine low vacuum" annunciator which indicates a vacuum of 7 inches of mercury absolute instead of the normal operating level of 1 to 3-1/2 inches of mercury absolute. The operator checked the mercury manometer and verified the low vacuum indication. He immediately proceeded to the auxiliary ejector station and began opening the steam supply valve, but the condenser vacuum raised to the 15" trip point before the auxiliary ejector was valved to the condenser. The primary pump slowed down and transferred to the pony motor drive at 900 rpm without incident, but the resulting drop in total primary loop flow caused low-flow trips on process tubes 1370, 0571, 0873, 1873 and 2773 which tripped the scram circuit. Only a 10% decrease in nominal process tube flow is required to give a low-flow trip.

Following the turbine tripout on September 30, 1970, a procedure was developed for inspecting and testing the No. 6 surface condenser in an attempt to locate the cause of the loss of vacuum. The following items were included:

1. Inspect the raw water side of the condenser for plugged tubing, and remove any debris found.

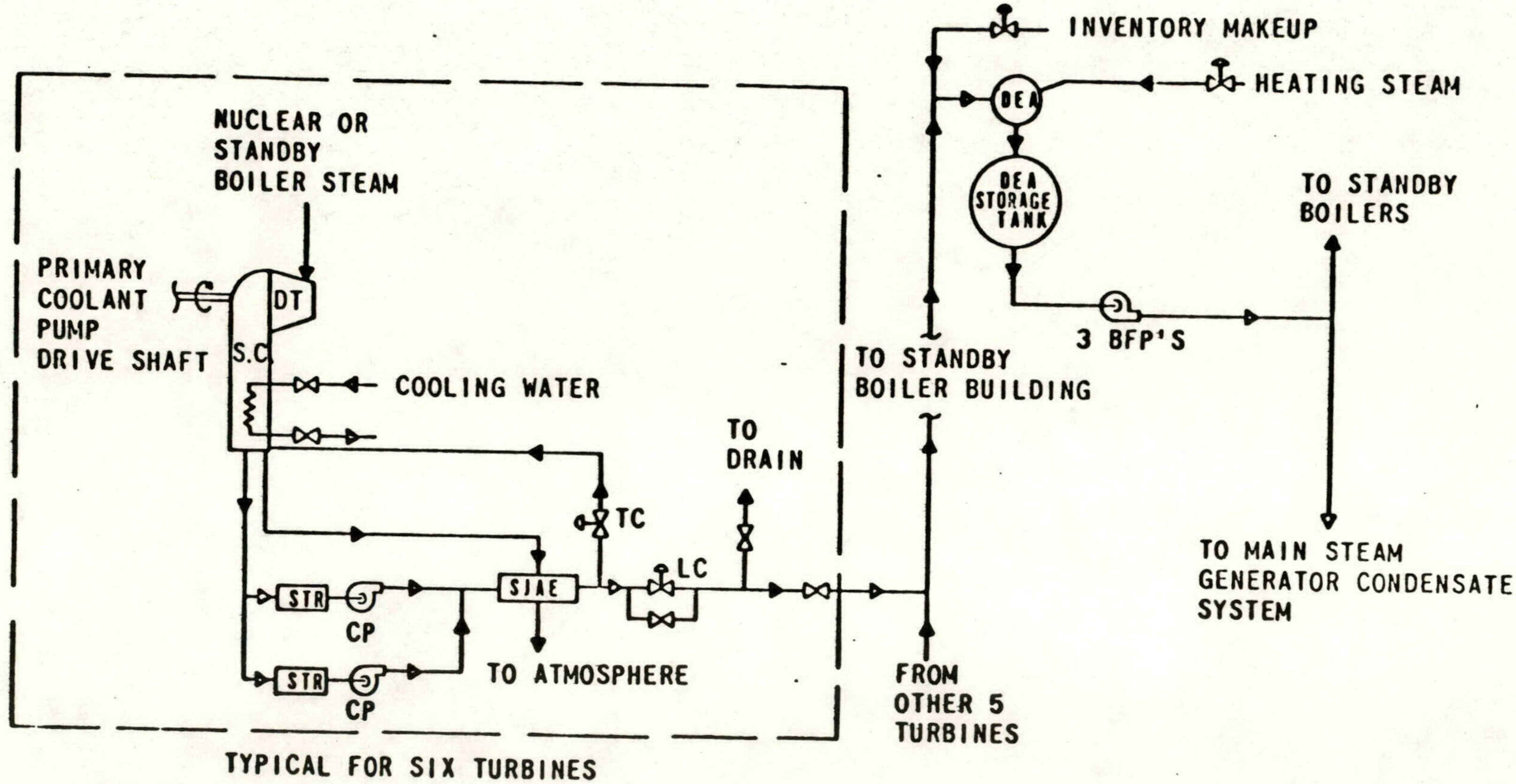
2. Inspect for leakage between the condensate side and the raw water side.

3. Test condensate pump performance.

The inspection revealed little debris on the raw water side of the condenser, and no significant amount of plugging of the condenser tubes (less than 5%). No leaks were detected in the tubing. The functional test of the condensate pumps indicated that the flow was less than normal for two condensate pumps, although the vacuum stayed in the normal operating range expected under the test load conditions.

A second test procedure (reference: DUN-7306, "N Reactor Scram Report September 30, 1970") was prepared to determine the cause for the inability to control hotwell level. This test indicated little or no flow through condensate pump 6A and a significantly high Delta P across the pump's in-line suction strainer. (See Figure 9.) The performance of condensate pump 6B was equal to, or better than, the vendor's rated capacity. The decision was made to inspect the pumps internally.

The insulation was removed from the two surface condenser condensate pump suction headers and the lines were opened for inspection. The in-line strainer to pump 6A was found to be totally plugged with Garlock gasket material (Catalog No. 670). The pump 6B strainer was approximately 60% plugged with the same material. Similar material, along with five 2- to 3-inch sections of welding rod, was also found in a subsequent inspection of the hotwell of the No. 6 turbine condenser.



LEGEND

DT - PRIMARY COOLANT PUMP DRIVE TURBINE
SC - DRIVE TURBINE SURFACE CONDENSER
STR - IN LINE STRAINER
CP - CONDENSATE PUMP
SJAE - STEAM JET AIR EJECTOR
TC - SJAE TEMPERATURE CONTROL
LC - SURFACE CONDENSER HOTWELL LEVEL CONTROL
DEA - DEAERATOR HEATER
BFP - BOILER FEED PUMP

Figure 9.

PRIMARY COOLANT PUMP
DRIVE TURBINE
CONDENSATE SYSTEM

The original purpose of the pump suction strainers was to remove debris left in the condenser during construction. It was intended that the strainers should remain in the system for a few weeks initial break-in period following the initial startup (June, 1966) of the No. 6 condenser system and then be removed. After the removal of construction debris, the pumps should encounter only steam condensate, essentially distilled water, and thus suction strainers should not be necessary. Similar temporary strainers were included in the design of the other five cells. These strainers were, as intended, removed after a rather long break-in period of about fifteen months (March, 1965).

It was concluded that the strainer plugging was sufficient to inhibit condensate pumping capacity to where a low hotwell condensate level could not be maintained and, therefore, vacuum could no longer be maintained. To confirm this, the system was cleaned; the suction strainers re-installed; and No. 6 turbine placed on steam operation for two days with periodic acceleration of the turbine up to NPSH limits for the primary pump. Following this hot run, the strainers were re-examined, and found to be clean. No difficulties were encountered during the hot run. Both A and B pumps operated without difficulty. The turbine was then considered serviceable and testing was terminated.

Following this last test, the suction screens were permanently removed from the No. 6 condensate system, and checks were made which confirmed that all condensate pump screens were removed from all other cells. An analysis was also made which indicated that neither the Garlock gasket material or the small pieces of welding rod found in the No. 6 condensate system should not have been able to significantly affect the operation of the secondary steam system if the screens had not been in place.

Due to the nature of the material found in the strainers and in the hotwell, it has been concluded that it could not have entered through the turbine itself, but rather must have been introduced directly into the hotwell from an external source.

The two primary hypotheses which were proposed to explain the method of entry of the gasket material into the condensate system were:

1. A full sheet of Garlock material, approximately 2-feet square, was used as a gasket for one of the large hatches on the No. 6 condenser shell instead of the normal practice of cutting out the center portion of the sheet. This center portion of the gasket might then weaken after prolonged exposure to steam and eventually enter into the hotwell condensate stream.

2. The gasket material was accidentally introduced into the condenser shell during maintenance activities.

An attempt was made to quantitatively determine the amount and approximate original shape of the recovered portions of the Garlock material to see if that information would help in determining the source. Best estimates were that there was a total of about 90-square inches of material removed from the No. 6 condensate system. The material was so badly deteriorated that its original shape could not be clearly defined; however, one edge of one of the pieces might have been a straight edge.

In an attempt to rapidly and grossly determine the effect which the 100° F steam environment of the condenser might have on the exposed area of a "full face" gasket, a sample of fresh Garlock gasket material No. 670 was boiled in deionized water for three weeks with very little deterioration noted. This would indicate that the material taken from the No. 6 condensate system had probably been in the system for a long time. Both large rectangular hatches of the No. 6 condenser shell were also opened following the September 30 scram. The lower hatch was found to have a Garlock oil pack gasket and the top hatch gasket was graphite-impregnated asbestos. Both gaskets were found to be in good shape and were reused. There was no indication that a "full face" gasket had been installed.

The Garlock gasket material is one of several types of gasket material commonly used in N Plant for steam and hot water service. Other gasket materials are (1) white asbestos G-1, (2) red rubber, and (3) black neoprene. There are no connections inside the shell of the No. 6 condenser where any of these gasket materials could be normally used; however, there are several openings to the condensate system in this area where gasket materials are commonly used and through which materials might be inadvertently introduced during maintenance activities.

- A rectangular 20-inch by 24-inch hatch on the top of the No. 6 condenser shell.
- A rectangular 23-inch by 24-inch hatch on the bottom of the No. 6 condenser shell.
- The No. 6 turbine cover.
- The steam supply lines between the No. 6 turbine and the No. 6 condenser.
- The turbine casing rupture disc port.

Operations' logbooks from May, 1964 were reviewed to determine and summarize the No. 6 drive turbine unit maintenance history. Significant observations made in reviewing the logs were:

1. Recent significant amounts of work on the condenser hotwell are as follows:

May 21, 1969	Cleaned No. 6 surface condenser (raw water side).
October 8, 9, 10, 1969	No. 6 hotwell drained and inspected for evidence of tube leaks (both sides).
March 19, 20, 1970	Removed section of tubing from surface condenser No. 6.

On the May, 1969 and March, 1970 dates, the hotwells (according to Maintenance and Manufacturing Engineering records) were not entered. Consequently, there was no chance of leaving any gasket material in the unit during that work period. Consequently, the only recent chance of leaving material in the unit was in October, 1969.

2. Prior to the outage which began on May 8, 1969, the water quality analysis of No. 6 surface condenser condensate samples rarely exceeded 25 ppb O_2. Following the May 27, 1969 startup, there were consistent indications of high O_2 (300 ppb) in the condensate samples from the No. 6 unit. A number of tests were made from June, 1969 up to the present in an attempt to determine the cause of the continuing high O_2. Results of these tests indicated that the in-leakage was probably occurring at the condensate pump seals.

3. To further troubleshoot the cause of the air in-leakage and determine whether the tubes were being damaged, further checks of the system were made in the October, 1969 outage. The raw water side of the surface condenser was opened and inspected in an attempt to locate any plugging, debris, etc. Nothing significant was found. The two pumps were disassembled, inspected, new sleeves installed, and new packing installed after reassembly. The suction screens were not checked at this time.

4. A section of tubing was cut out and removed from the raw water side of the surface condenser during the March, 1970 outage; however, the condensate side was not opened at that time.

B. Conclusions

The investigation into the cause of the No. 6 drive turbine tripout revealed the following:

1. The turbine trip was caused by a low vacuum condition in the surface condenser.

2. The low vacuum condition in the surface condenser resulted from a high condensate level in the condenser hotwell which, in turn, was the result of reduced pumping capability of the condensate pumps.

3. The surface condenser condensate pump capability was reduced because the in-line strainers in the pump suction lines were plugged with Garlock gasket material.

4. How or when the Garlock gasket material was introduced into the surface condenser has not been determined. The last time the No. 6 condenser secondary side was entered for maintenance activities was November, 1969.

5. The No. 1 through 5 drive turbine units had screens in the surface condenser condensate pump suction lines on original startup but they were removed in March of 1965.

6. The No. 6 drive turbine unit is the only one that had screens in the surface condenser condensate pump suction lines at the time of the September 30 scram.

7. The suction piping to all of the drive turbine condensate pumps were equipped with gauges to read the pressure drop across the screens. However, no periodic procedural requirements existed to monitor or clean the screens.

8. The No. 6 drive turbine condensate system problem did not contribute to the rod scram problem.

VI. BALL SAFETY SYSTEM INVESTIGATION

A. System Description

The function of the ball safety system of N Reactor is to provide a redundant method to achieve safe shutdown of the reactor which is independent of the control rod system. The ball safety system nominally consists of 107 hoppers filled with a neutron absorbing material (Samarium Oxide) in the form of balls approximately 3/8-inch in diameter. The ball hoppers are located above the reactor core and will drop their Samarium Oxide balls into the reactor upon either automatic or manual signals. The ball system is capable of safely shutting the reactor down with a safety factor of two in any credible situation without assistance from the control and safety rod system.

Each ball hopper has an individual ball dump or trip mechanism which is held closed against an opening spring by two solenoids. Each solenoid is powered by a separate electrical circuit. Both solenoids must de-energize to permit the balls to drop into the reactor. (See Figure 10.)

The ball hoppers trip mechanisms can be activated in any one of the following ways:

1. Actuation of a manual pushbutton on the reactor operator's console in the control room will trip all ball hoppers and

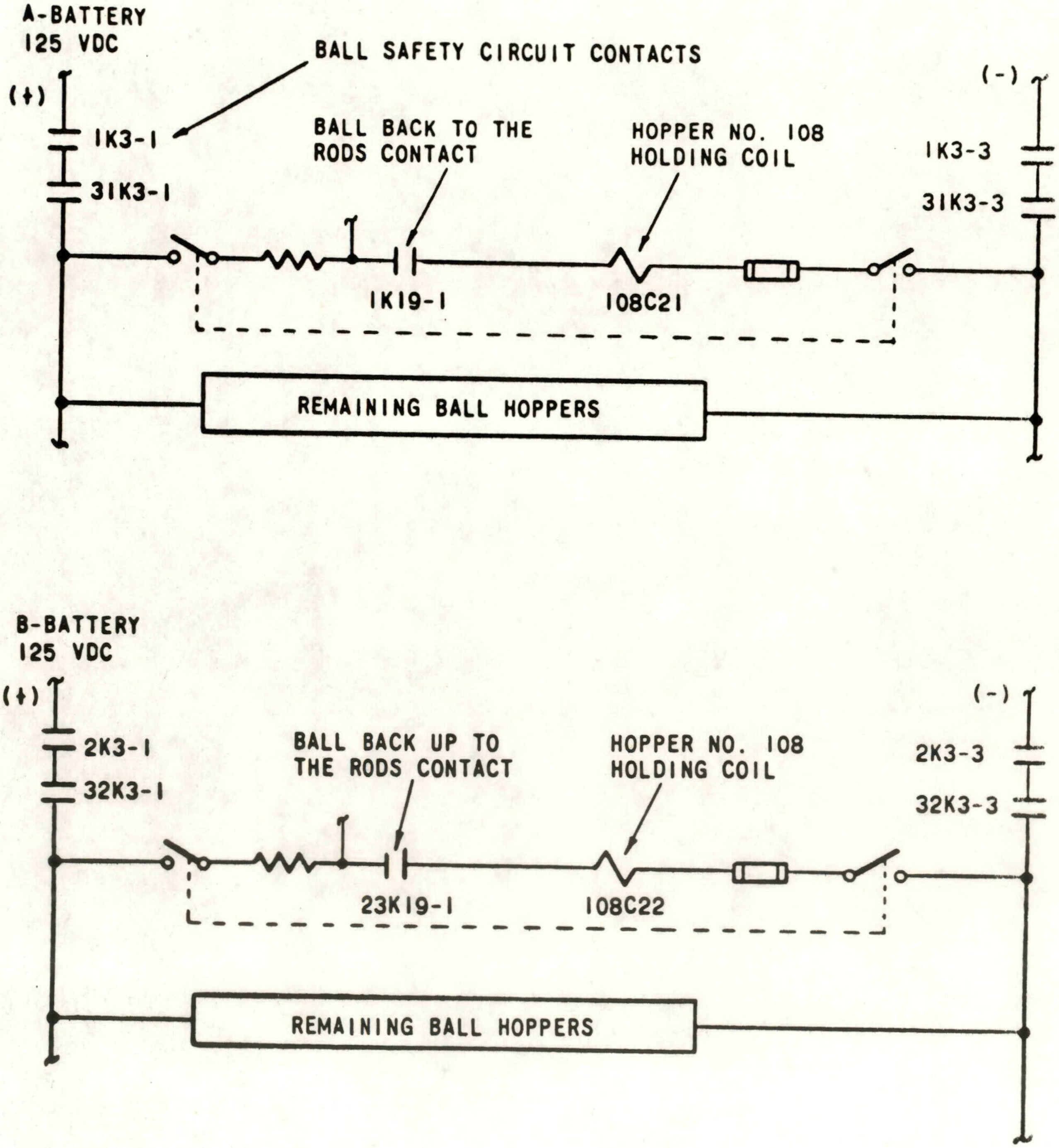


Figure 10. Ball Safety System, Ball Hopper Solenoid Circuits

drop all balls simultaneously. Individual hoppers can also be tripped by actuating two switches, located at the 50-foot level for test purposes; however, this method is not used while the reactor is critical.

2. Dump of the reactor primary loop by initiation of the once-through emergency cooling system for any reason during reactor operation will dump all the ball hoppers into the reactor.

3. Indication of a modified Mercalli intensity-4 earthquake by two out of the three seismoscopes will drop all balls.

4. High flux after scram - If the high-range flux monitors indicate that the reactor is not below 25% power (approximately 1000 mw_{th}) within five seconds after the scram signal is received, all balls will drop.

5. Supercritical after scram - If the intermediate range flux monitors indicate that the reactor power level is not below 4 mw_{th} within 300 seconds after the scram signal is received, all balls will be dropped.

6. If two or more rods in any one vertical column of control rods take more than 1.5 seconds after scram to reach the 75% inserted position, the required ball hoppers will dump to create a "curtain" of balls on both sides of the affected column of rods.

7. If three or more vertical columns of rods have one or more slow rods (more than 1.5 seconds after scram to reach 75% in), the ball hoppers will dump which are necessary to create curtains of balls on both sides of each affected column.

Since no rods responded to the scram signal within 1.5 seconds on September 30, the simultaneous functioning of control circuits 6 and 7 above caused all ball hoppers to dump. Reactor shutdown by the ball safety system was, therefore, safely achieved less than two seconds later than it would have been if the rods had scrammed. (See Figure 11.)

B. Chronology

Immediately following the September 30 scram signal, control room instrumentation on the open position of each ball hopper gate indicated that all ball hoppers had dumped except Nos. 3, 7, 19, 40 and 48. Ball hopper No. 40 was known to have been taken out of service for maintenance and, therefore, should not have dumped. A visual inspection of all ball hoppers immediately after the scram revealed that only ball hopper No. 3 had actually failed to function as expected.

An examination of hopper No. 3 revealed that the hopper gate did not open because the trip mechanism locking collar failed to release. Only

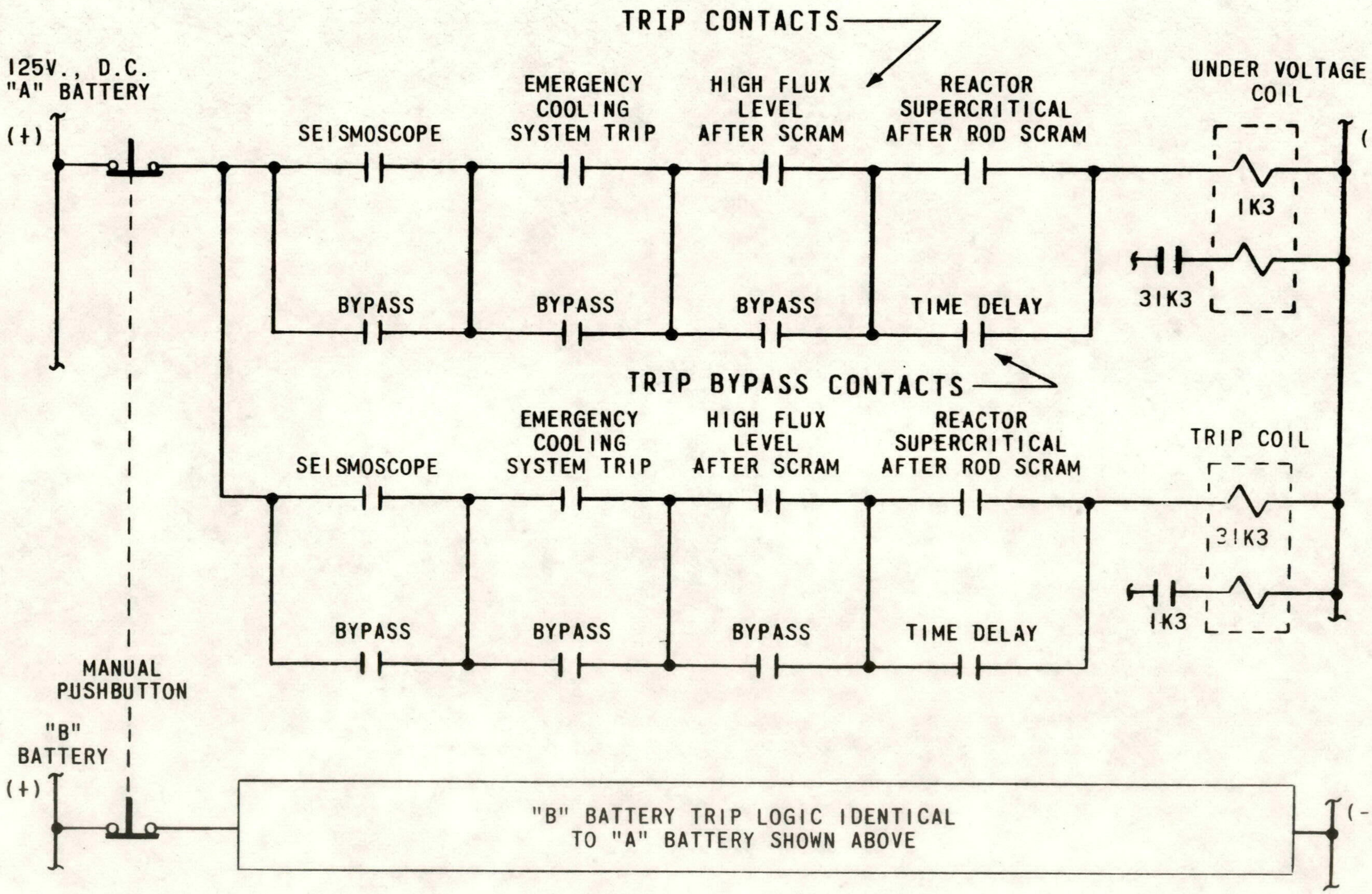


Figure 11. Ball Safety Circuit

one probable cause as to why the collar failed to release could be found when carefully disassembling all of the hopper components; for example, one of the four steel retaining balls in the latching mechanism was missing. (See Figure 12.) The missing retaining ball allowed the hopper gate locking collar to misalign sufficiently to prevent opening of the hopper gate. The missing ball was found near the No. 3 hopper and had apparently dropped out of place during assembly.

The trip mechanism on hopper No. 3 was reworked on June 22, 1970, because it was corroded. The mechanism was reworked again during the summer outage because an inspection after the first reworking revealed that the cocking collar could not be pushed down.

Hopper No. 3 was satisfactorily tested in accordance with the EMS-103 requirements on August 26, 1970. The test consisted primarily of verifying that the hopper trip mechanism could be cocked, locked, and tripped satisfactorily. An accelerometer was attached to the hopper fill spout to determine the trip mechanism response time, and the "hold-in" coils were tested individually to verify that either would keep the mechanism cocked.

Following the investigation on September 30, 1970, the trip mechanism on hopper No. 3 was cleaned and reassembled with new gas seal "O" rings and four steel balls. Testing in accordance with the EMS-103 requirements was performed, and the hopper was returned to service.

The failure of the "gate open" indications on hoppers 17, 19, and 48 was determined to be due to misalignment of the microswitches. All the "open" microswitches on all hopper gates had been successfully tested by EMS-103 on August 26, 1970; however, the configuration of the microswitch holding brackets is such that it is believed that they could have become subsequently misaligned by the tripping action of the hopper gates. It should be noted that the open hopper gate microswitches are not considered critical since they do not provide control inputs to any safety systems. The microswitches on hoppers 17, 19 and 48 were adjusted and tested, and the hoppers returned to service.

C. Independent Review of Ball Safety System

As a result of the problems experienced during the September 30 scram, a total review of all critical safety systems was made to insure that no other similar problems existed. The criteria for this review and a discussion of the other systems reviewed are provided in Section VII of this report.

Like the rod safety system, the ball safety system is failsafe; for example, if the power to the ball system circuits is lost for any reason, the hoppers will dump. However, unlike the rod system, it requires that two redundant safety circuits de-energize for the total system to operate. Therefore, failure of one of the two redundant circuits, which causes it to remain energized, can result in loss of

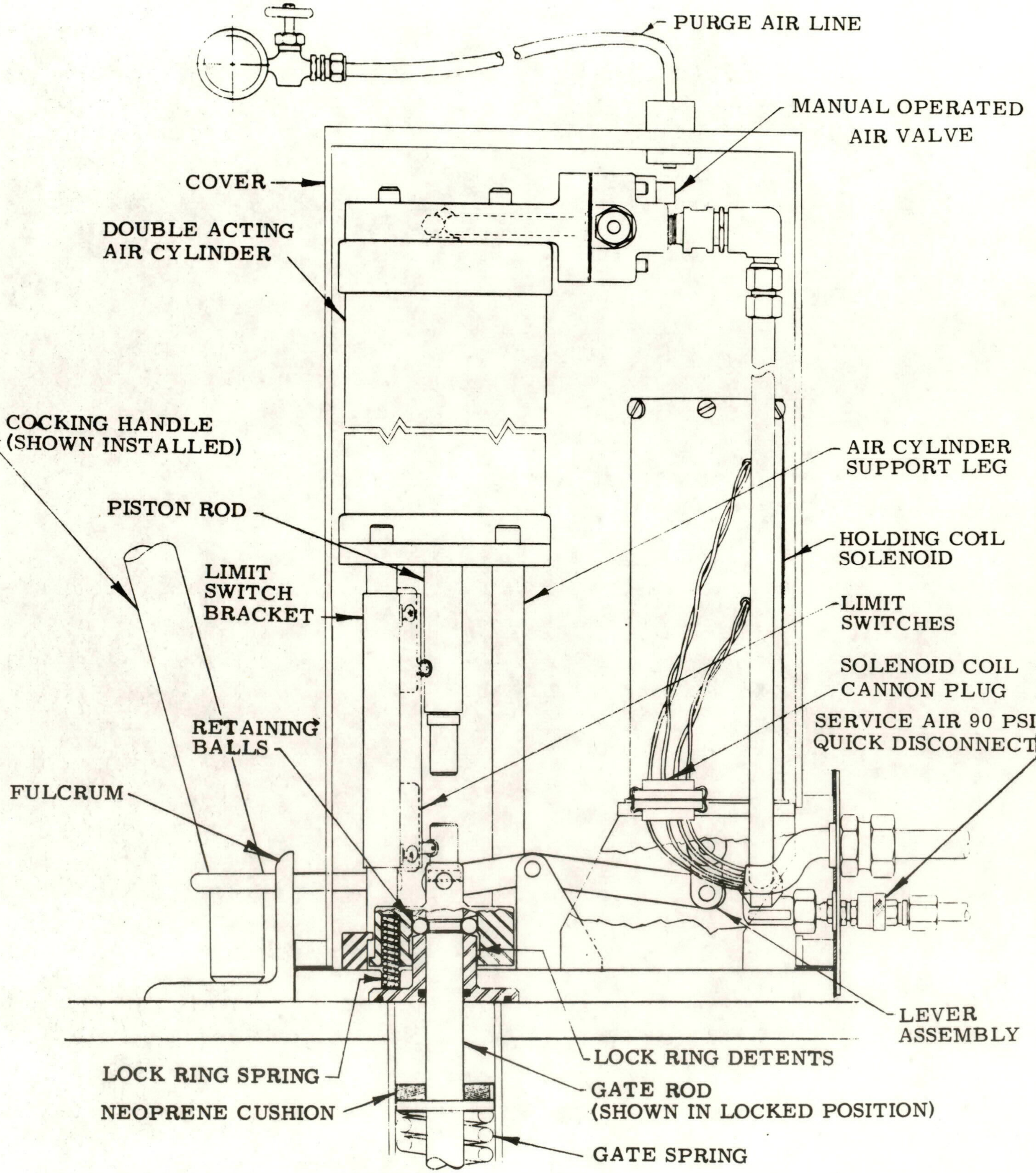


Figure 12. Ball Trip Mechanism

the ball safety function. This fact was considered carefully during the review of the ball system.

No interconnecting diodes were found between the rod system and the ball system, nor were any other physical connections noted.

Like the rod system, two or more shorts or grounds are normally required to negate system operation. Two exceptions are as follows:

1. A short across one of the manual pushbutton contacts will negate that function.

2. A timer circuit was identified which bypasses for five seconds following a rod scram to the high flux after scram matrix input to the ball safety system. This bypass is significant in that failure of either of the two timer contacts to open will negate that function.

Since both of these functions are backed up by the supercritical-after-rod scram and the ball backup-to-rod scram trips, it was not felt necessary by the Review Team that any modifications be required before startup; however, the recommendation was made to develop a design change as soon as possible to rectify the timer situation.

Also, like the rod system, locations were found where bare terminals were less than 1/4-inch apart. A simple solution of installing insulated sleeving over the terminals was recommended by the Review Team.

The ball system employs many surge suppressors in the same fashion as described above for the rod system. The individual relay contacts are inspected and operated every six months by EMS-103 similar to the rod system. Equipment Maintenance Standard 103 was last performed during the 1970 extended summer outage. Hence, it was felt that the action was sufficient to ensure reliability in this time.

The startup procedure for the ball safety system requires that the main ball circuit breaker be tripped only once every three months. In addition, it is not clear in the procedure whether merely one trip input system is to be used at each three-month interval to trip the ball breaker or whether all four are to be tested. This established a possible condition where it could take one year to test the safety circuit trip input system, if only one system were used each three months. Hence, it was recommended by the Review Team that all safety circuit trips in the ball system be functionally tested each startup following an extended outage, and that actual verification be made that the ball safety system circuit breaker actually functioned in response to each trip input.

D. Conclusions

1. The N Reactor ball safety system responded automatically as required to achieve a safe shutdown when the control rods failed to scram into the reactor on September 30, 1970.

2. Out of 106 in-service ball hoppers only No. 3 failed to successfully dump its Samarium Oxide neutron absorbing balls into the reactor. Failure of hopper 3 was determined to be due to a missing steel retaining ball which permitted the hopper gate locking collar to misalign. The missing retaining ball was apparently inadvertently lost out of the ball hopper gate mechanism at assembly following maintenance during the 1970 Summer Outage of N Reactor.

3. The failure to receive "gate open" indications from hoppers 17, 19, and 48 following the successful dump of these hoppers during the scram was found to be due to misalignment of the microswitch brackets. Misalignment apparently resulted from the normal, but violent, action of the trip mechanism during a ball dump.

E. Recommendations

1. Insulate all bare terminals in the ball safety system circuitry where a potential exists for inadvertent shorts which might degrade system reliability.

2. Each startup, following an extended outage, functionally test to insure that the ball system safety breakers (1K3, 2K3, 31K3, 32K3) are actuated by each trip input to the ball safety circuit.

3. Develop a design change to eliminate the possibility of one of the parallel five-second timers in the "high flux after scram" circuit failing and preventing an input to the ball safety trip circuitry.

VII. REVIEW OF OTHER N AND KE REACTOR CRITICAL SAFETY SYSTEMS

A. Introduction

As a result of the problems experienced during the September 30 scram of N Reactor, reviews of all critical safety systems of both N and KE Reactors were made to insure that no other similar problems existed.

B. N Reactor Critical Safety Systems Review

For the purposes of the N Reactor review, a team was established which consisted of representatives from the Technology Section, Engineering Section, Manufacturing Engineering Section, Process Section, and N Plant Section.

The criteria that were used to judge system adequacy during the N Reactor review were selected requirements from the IEEE 279 Criteria document, dated August, 1968. Those selected were:

- 4.2 Single Failure Criterion
- 4.6 Channel Independence
- 4.7 Control and Protection System Interaction
- 4.10 Capability for Test and Calibration
- 4.11 Channel Bypass or Removal from Operation
- 4.12 Operating Bypasses
- 4.13 Indication of Bypasses
- 4.14 Access to Means for Bypassing

In addition to the Single Failure Criterion, another similar criterion was invoked regarding redundant channels. In redundant channels, if all but one order of redundancy failed leaving one channel for operation, and if this fact could occur without notification, then that remaining channel was considered a single failure problem. Since the rod scram problem was the apparent result of diode failures, particular attention was given to diode installations in all safety systems circuitry.

The systems studied were the rod control and safety system, the ball safety system, the emergency cooling system, the confinement system, and appropriate individual instrumentation systems which provide control inputs into safety systems.

The recommendations of the Review Team in each system are provided in the following text except for the recommendations concerning the rod safety system and ball safety system which have already been presented in Sections IV and VI of this report. All Review Team pre-start recommendations were accomplished prior to renewed N Reactor operation. Longer range recommendations and their action schedule are summarized in Section IX of this report.

1. Emergency Cooling System

The Emergency Cooling System (ECS) is a two-channel redundant system, either channel of which will initiate the emergency cooling function. However, the systems must energize in order to operate (not failsafe). A single failure potential exists in many locations in these two redundant systems in that wiring, controls, and other components for both are located in single relay panels and wireways. An inspection was made to review housekeeping, fire hazards, etc., in these common location areas. As a result of this inspection, the Review Team recommended several specific items of housekeeping be accomplished.

The function of each redundant channel is tested each month by completely disarming one system then energizing the other through a manual trip and vice versa. This was last done in September, 1970. This test operates each system all the way, from a pushbutton at the head end of the circuit clear through the final outputs of the system. Therefore, the functional adequacy of each system was considered to be established. However, since this procedure is not a standard operating procedure, it was recommended by the Review Team that it be made a part of mandatory startup check lists.

One test was recommended by the Review Team which involved functionally operating the individual ECS valves from their individual accumulators (nitrogen cylinders). This test was recommended even though the system had been functionally tested very recently. The one shortcoming that the normal functional test had was that valve operation was not tested using only the individual accumulators; rather, the two supply manifolds to the individual valves were valved off, one at a time, and the valves were operated using the other manifold for air pressure. The team recommended that the accumulator test be made as a part of a mandatory EMS.

2. Confinement System Annunciation

A review of the control circuits of the confinement bypass annunciation systems revealed that three bypass functions have no direct annunciation in the 105-N control room. These are:

a. 105-N Steam Vent and Confinement Bypass - This switch is not a key interlock switch and has only panel light indication of circuit bypassing.

b. 109-N Steam Vent Closure Bypass - CS/49 - This switch is not a key interlock switch and has only panel light indication of circuit bypassing.

c. 109-N Supply Fan Bypass - 2H502 - This switch is a key-locked switch with panel light indication of circuit bypassing. During normal reactor operation, the key is left in the switch.

The recommendation was made to provide annunciation in the 105-N control room when any of these functions are bypassed.

Review of the 105-N steam vent timer circuit showed that the timers necessary for proper confinement sequencing are supplied with an automatic DC bus transfer circuit. Should the normal DC bus powering the timers fail, the circuit automatically transfers to a backup DC bus. Once the transfer has been made, return to the primary DC supply requires manual action in 105-N, Room 6. The recommendation was made to provide annunciation in the 105-N control room when the backup DC bus is supplying the timer circuit.

These recommendations were incorporated into Design Change 3125, "Confinement System Annunciator Changes." This design change was completed on October 9, 1970.

3. Confinement and Fog Spray Pressure Switches

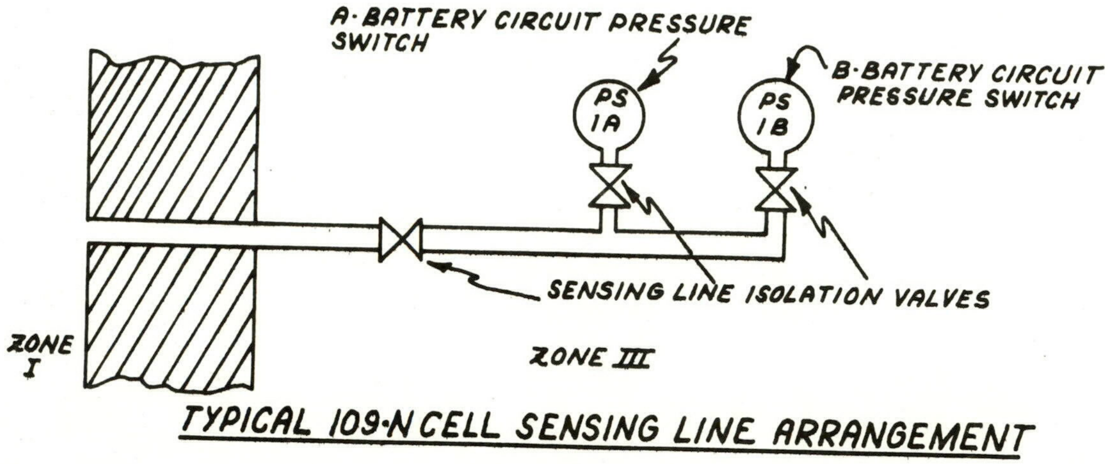
Zone 1 atmosphere sensing pressure switches in the 105-N and 109-N Buildings are used to detect reactor incidents requiring activation of the fog spray and confinement systems. In the 109-N Building, there are two ambient pressure sensing lines in each of the six cells and in the pipe gallery with two pressure switches on each line. The two pressure switches on each line are powered from different DC power sources for redundancy. The two switches on one line actuate at 2 inches of water pressure and provide inputs to redundant two out of seven trip logic circuits which initiate the Zone 1 confinement sequence. The two pressure switches on the other sensing line actuate at 10 inches of water pressure and provide inputs to redundant two out of seven trip logic circuits which activate the fog spray system. (Reference: Figure 13.)

During the confiner leak tightness test on September 9, 1970, two similar ambient pressure sensing lines were found capped in the 105-N left and right front pipe spaces. During subsequent testing, three of the pressure switches in the 105-N pipe spaces were found to have been inadvertently left valved off. These disabled pressure switches had apparently not been detected during the integrated confinement system test of August 18, 1970.

It was felt essential to verify that no other ambient pressure sensing lines were capped or disabled. The recommendation was made to pressurize all such pressure switches from the sensing line openings in Zone 1 and verify proper pressure switch operation on indicator lights in the 105-N control room.

As a result of the special test, all eight sensing lines for the fog spray and confinement pressure switches in cells 1, 2, 4, and 5 were inspected and found sealed off. (Reference: DUN-7395, dated October, 1970.) One line in cell 1 was sealed off with self-tapping screws and rubber gaskets in each of the 21 holes in the specially made probe of the sensing line. (See Figure 14.) The other seven lines were sealed by wrapping the sensor heads with black plastic electrical tape. The sensing lines in cells 3 and 6 and the pipe gallery were unobstructed. However, tape marks were visible on the cell 3 sensing lines.

The evidence of cell wall elastomer coating on the outside of the tape indicates that the taping was done prior to the coating of the cell walls prior to initial reactor startup in 1964. Initial cell leak-tightness tests or pressure or vacuum tests during initial startup may have been the type of activity for which sealing of the pressure sensing lines was necessary.



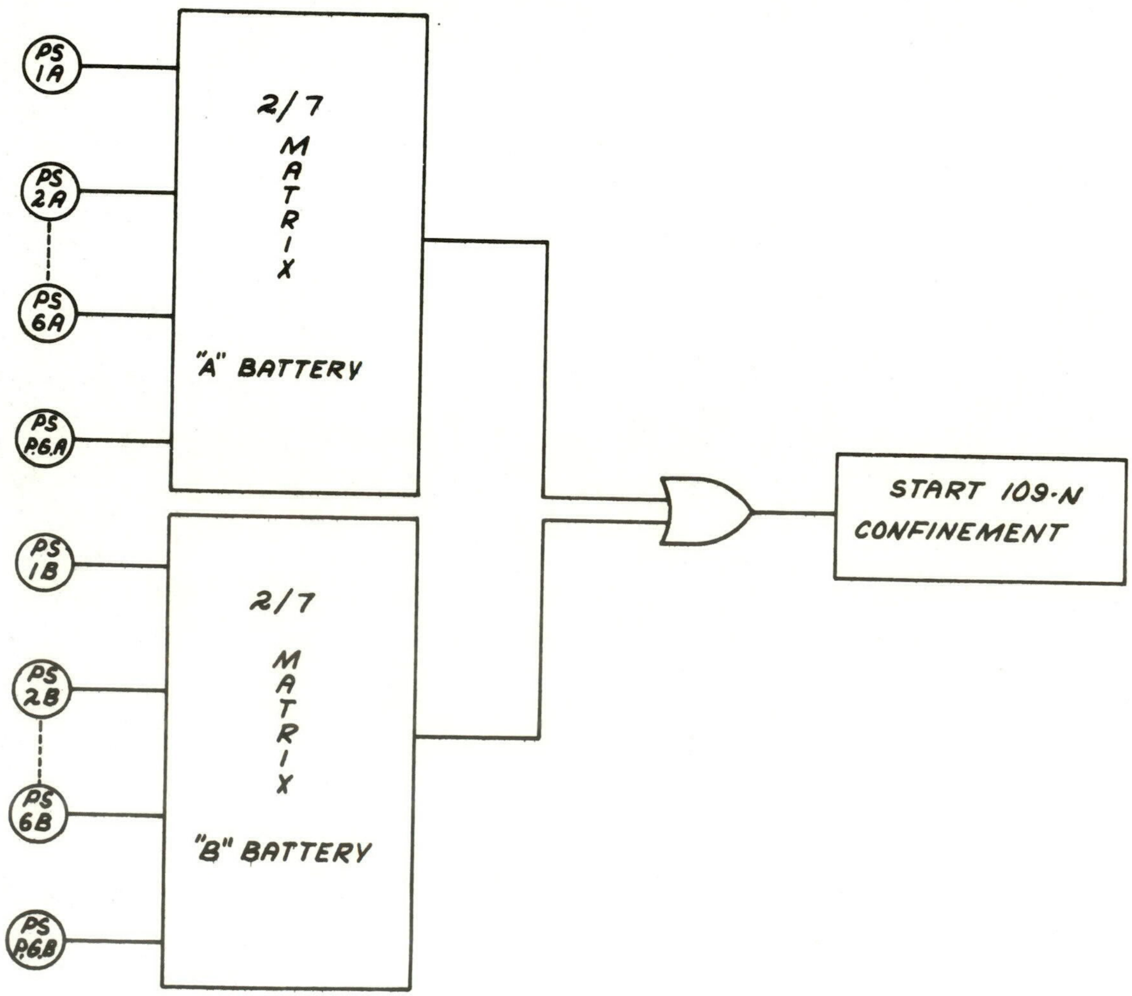


Figure 13. 109 Confinement Trip Circuit

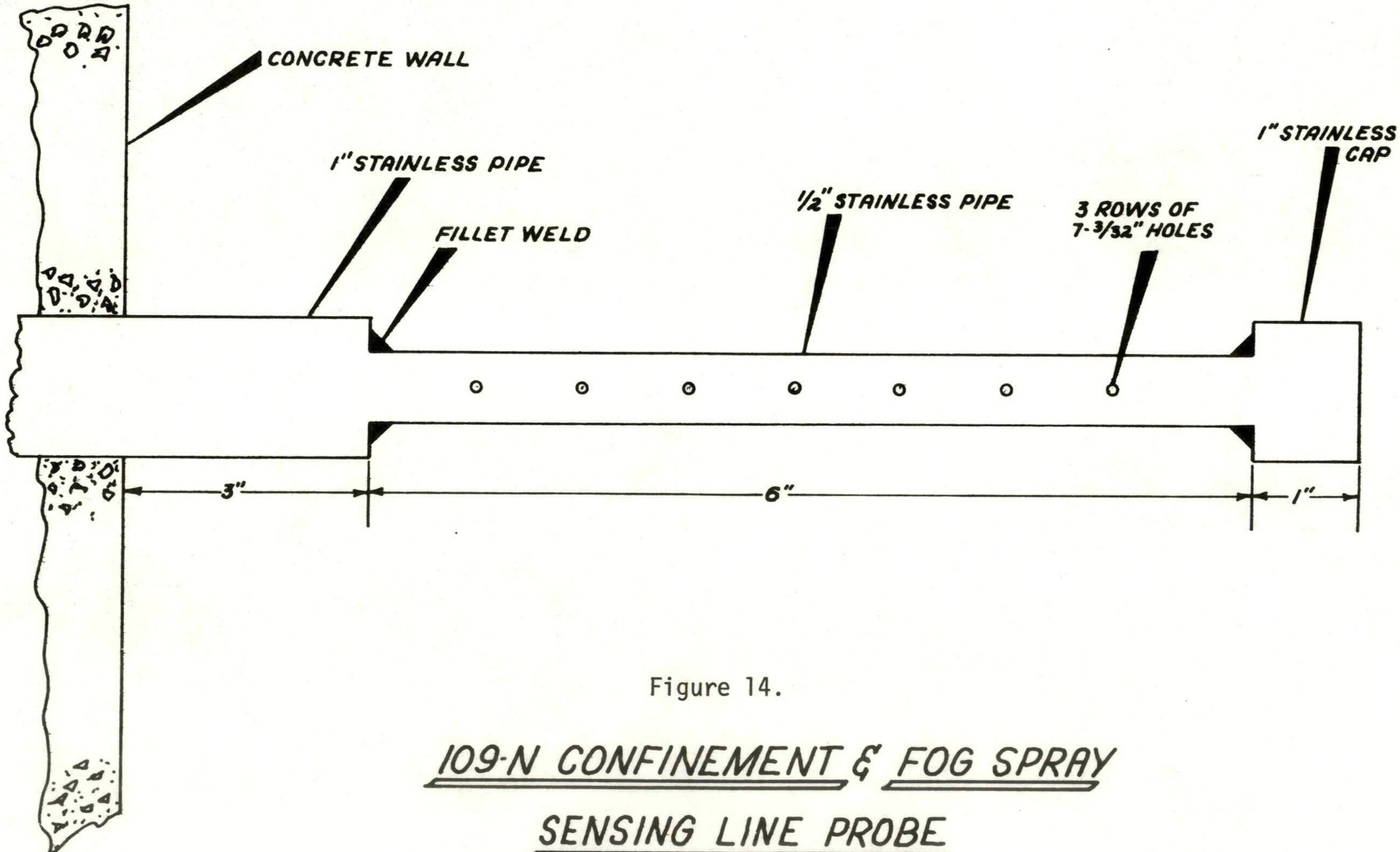


Figure 14.

109-N CONFINEMENT & FOG SPRAY

SENSING LINE PROBE

These inadvertent deactivations of the pressure switches in four out of the seven locations reduced the fog spray and confinement actuation logic to a 2 out of 3 matrix, rather than 2 out of 7. During the time when cell 3 was isolated for retubing, only a 2 out of 2 matrix probably existed. In both cases, however, there were two redundant logic circuits available with a 2 out of 2 matrix. The communication between cells also should have permitted a problem in any of the areas with deactivated pressure switches to be detected by the other active pressure switches.

All plugging was removed and all 109-N pressure switches were then tested individually for actuation, but not in the sets of 2 out of 7 required to achieve fog spray or confinement trips.

Following these last tests a process engineering check of the 109 Building pressure switch isolation valves was made, and three pressure switches were again found valved off.

As a result of the difficulties encountered with the 105 and 109-N ambient pressure switches, the following recommendations were made:

1. Testing of these pressure switches should be covered in the future by proposed EMS-603, which in its present form should detect any capped sensing lines. It is recommended that EMS-603 be modified to also detect closed isolation valves.

2. Signs should be placed at all the sensing lines and isolation valves to warn against inadvertent closing or plugging.

3. Isolation valves for pressure switches which provide control inputs to safety systems should be included in the "N Plant Operations Subsection Equipment Status Control System," DUN-7201, DL Howard, dated August 31, 1970.

4. <u>Confinement Backup Devices</u>

Backup to the 105-N Zone 1 exhaust confinement valves is supplied by vertically sliding, piston-operated gates. (Reference: Figure 15.) These gates are held open by the application of pressurized gas against the piston operators. Normal supply of gas is supplied by on-line nitrogen cylinders. The recommendation was made to remove the normal supply of instrument air from the gates and verify that the backup nitrogen system develops sufficient pressure to maintain the gates in the raised positions. Observations of gate slippage have been made when the normal air supply to the gates was removed. If these gates were to close (which would be their failsafe mode), they would prevent establishing a filtered fission product release to the confinement filters.

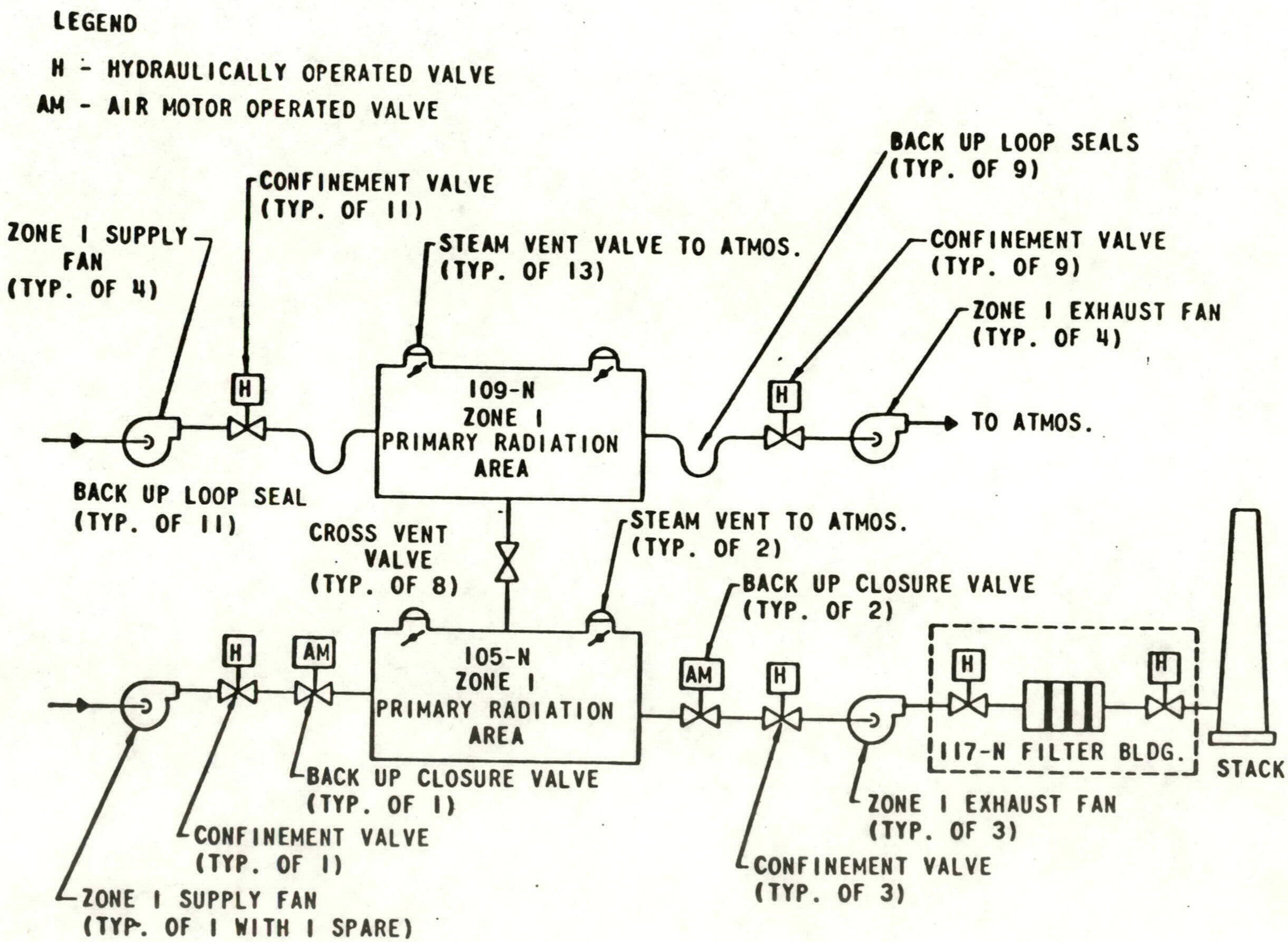


Figure 15. Confinement Ventilation System - 105 - 109-N

The test was conducted by valving off the normal air supply with the following results:

> One nitrogen cylinder was valved into the backup gas supply header (two were available -- one was valved off); the pressure was 1400 psi at the beginning of the test. Three minutes after shutoff of the instrument air supply, the larger of the two backup gates had fallen completely closed; the smaller gate remained fully open. After thirty minutes had transpired, the smaller gate remained fully raised. The test was terminated that point when it was noted that the nitrogen cylinder pressure dropped to 500 psi, a 900 psi pressure loss over a thirty-minute period. It was felt that the smaller gate would have probably fallen if the test were run to its scheduled end (one hour).

A study previously made by Technology Section had recommended that the Zone 1 exhaust backup gates be fastened to stay in the raised or open position. This recommendation was accomplished on October 15, 1970, by Design Change 3129, "105-N Zone I Exhaust Fan Confinement Backup Gates."

Backup to the 105-N Zone I supply confinement valve is supplied by a vertically sliding, piston-operated gate. This gate is identical to those used at the exhaust valves. The recommendation was made to cycle this valve, verify proper device closure, and verify that the gate has a seal. No records could be found of a previous test performed on this gate. Also, since the last confiner leak-tightness disclosed that one of the exhaust backup gates had no seal, the same possibility existed for this backup gate. This gate will be periodically tested on proposed EMS-603. Results of the test indicated that the gate worked freely. A small gap was found on the mating surface. The gate will be adjusted to provide complete seating.

5. <u>Fog Spray Strainer Bypass</u>

The butterfly bypass valve around the 109-N fog spray Y-strainer serves to bypass the Y-strainer should strainer plugging from debris accumulated in the fog spray header occur on a fog spray trip. Equipment Maintenance Standard 605 (Fog Spray System) does not properly test this valve and its controls. The recommendation was made to simulate a 10 psi pressure differential across the Y-strainer and verify that the bypass valve fully opens. Results of the test were successful except for a pressure switch calibration shift. This was corrected, and the system returned to normal.

6. <u>Fog Spray Diesel Pumps</u>

The fog spray diesels are designed to automatically start when the water level in the RWS-2 system (high pressure raw water) accumulator drops below either of two level switches or when the pressure in the accumulator drops below 160 psi. These functions have been found not to operate correctly in the past, and were not tested in the recent confinement system test. The recommendation was made to functionally test both the level switches and the pressure switch by verifying that the trip of any will start both diesel engines.

Both diesel engines successfully started from each of the level switches. The accumulator low-pressure switch was found to be set below the 160 psi recommended setting of Equipment Modification Procedure (EMP) 64; the diesel engines started at 133 psi. Several diesel engine local starting air accumulator check valves were observed to be leaking during the test as the fog spray water accumulator level was raised and lowered during the test. Specifically, low starting air pressure annunciators alarmed on ECS high-lift diesel engines Nos. 1 and 3, and on ECS low-lift diesel engine No. 2. The fog spray accumulator check valves and 182-N starting air receiver check valves were tested and repairs made to those valves which failed on the decay tests.

The method of testing the fog spray accumulator low level and low pressure switches (50-710-MSFT-13) will be revised to provide better test results.

6. Radioactive Drain Valving System

The 109-N RDR system drains liquid effluents from the 109-N cell sump pumps. Normal effluent goes directly to the river; under confinement conditions, the water is diverted to the crib by motor-operated valves. These valves failed to function on the confiner leak-tightness test of September 7, 1970, due to electrical control wiring failure. A recommended test to cycle the RDR valves (RDRV-802-1 and 802-2) from the 105 control room and verify proper operation by limit switch operation was successfully accomplished. Testing of these valves will periodically be performed on proposed EMS-603.

7. Appropriate Individual Instrumentation Systems

Through the vehicle of the Regulatory Program, the individual instrumentation system that supply inputs to safety systems have been given detailed review. This knowledge was capitalized on to rapidly focus attention on the following areas:

a. Power Range Flux Monitor

It was discovered that a single connector can be disengaged from the rear of the high-average electrical chassis which averages the output of the 12 high-range flux monitors and provides inputs to several trip circuits. Disconnection of this single connector would result in:

- Loss of two of three inputs to the power rate-of-rise scram circuit. Since this circuit requires an input on two out of the three channels to provide a trip, it would be disabled.

- Loss of all three of the three input channels to the high-average power scram logic. This would disable this scram circuit.

- Loss of all three of the three inputs to the "reactor critical after scram" logic which would disable one of the methods to initiate a ball drop.

Since this could occur and no annunciation would result, it was recommended by the Review Team that the connector be tagged and labeled. The recommendation was successfully accomplished prior to startup. More drastic action was not recommended by the team because the connector is of the type that has a thread lock ring and cannot be disconnected simply. In addition, it should be noted that the outputs of the 12 individual high-range flux monitors to the two out of twelve scram logic is not affected by the disconnection of the connector.

b. Intermediate Range Flux Monitor

The self-check system in the intermediate range flux monitor is isolated from the main system through six diodes. A shorting failure of any one of the six diodes will lower the signal on other channels. This is in the non-failsafe direction and could occur unannunciated. These isolation diodes have no voltage transients and are not load carrying; therefore, redundancy does not seem to be of any particular advantage. Removing the circuit adversely affects safety as it is a failure detector. It was recommended by the Review Team that EMS-107 be expanded in the future to include testing of these diodes on an annual basis. However, the system appears to be in good functional condition at this time since the six-month checks of EMS-107 were performed in September, 1970. The annual checks of EMS-107 are due to be performed in December, 1970. In addition, pre-startup EMS-107 was performed on the 28th of September, 1970. Through these EMSs the calibration was checked and it is, therefore, known indirectly that the diodes have not failed.

c. Flow Monitor (Reference: Drawing H-1-27531, Attachment E)

(1) Potential Shorts

A potential exists for a short to occur between terminals of individual channels on the back of the main flow monitor panel. This could produce loss of the channels involved without annunciation of the event occurring. The fact that this event has actually occurred by a drop of solder falling between terminals in the Spring of 1970 makes it rather real. This problem was detected during startup trip checks of each channel. Since there are literally thousands of these terminals, which are very small and compact, they would be difficult to insulate. It is considered that a potting technique may be the most effective. No recommendations were made by the team for action before startup; however, it was recommended that the insulation be accomplished as a longer range item.

(2) Failure of Voltage Bias Resistor

Failure of a 392,000 ohm voltage bias resistor on any channel high or low trip circuit will negate that channel from being able to produce a trip signal. This failure can occur without annunciation. However, since these components have an excellent (no failures) reliability history, it was recommended by the team to accept their integrity based on a thirty-day (or each shutdown, whichever is longer) functional test of the system. This functional test would involve actually inserting a trip signal at the transducer input to the controller (upstream of the bias resistor) and observing actuation of the trip timers for each channel. The flow transducers and the circuits from the transducers to the controller input points are functionally checked by flow maps which are run during each startup and compared with a "master" flow map from the previous operating period.

(3) Cold Solder Joints or Broken Connections

Cold solder joints and broken connections can be a problem any where and can disarm single channels, rows, or even the whole system without annunciation. It would depend upon where they occurred. However, these kinds of failures are usually associated with new installations or following recent maintenance. Hence, it was concluded that the channel-by-channel trip test of the entire system, which was described earlier, would also verify that the system was free of cold solder joints and broken connections.

(4) Non-Redundant Power Supply Wiring

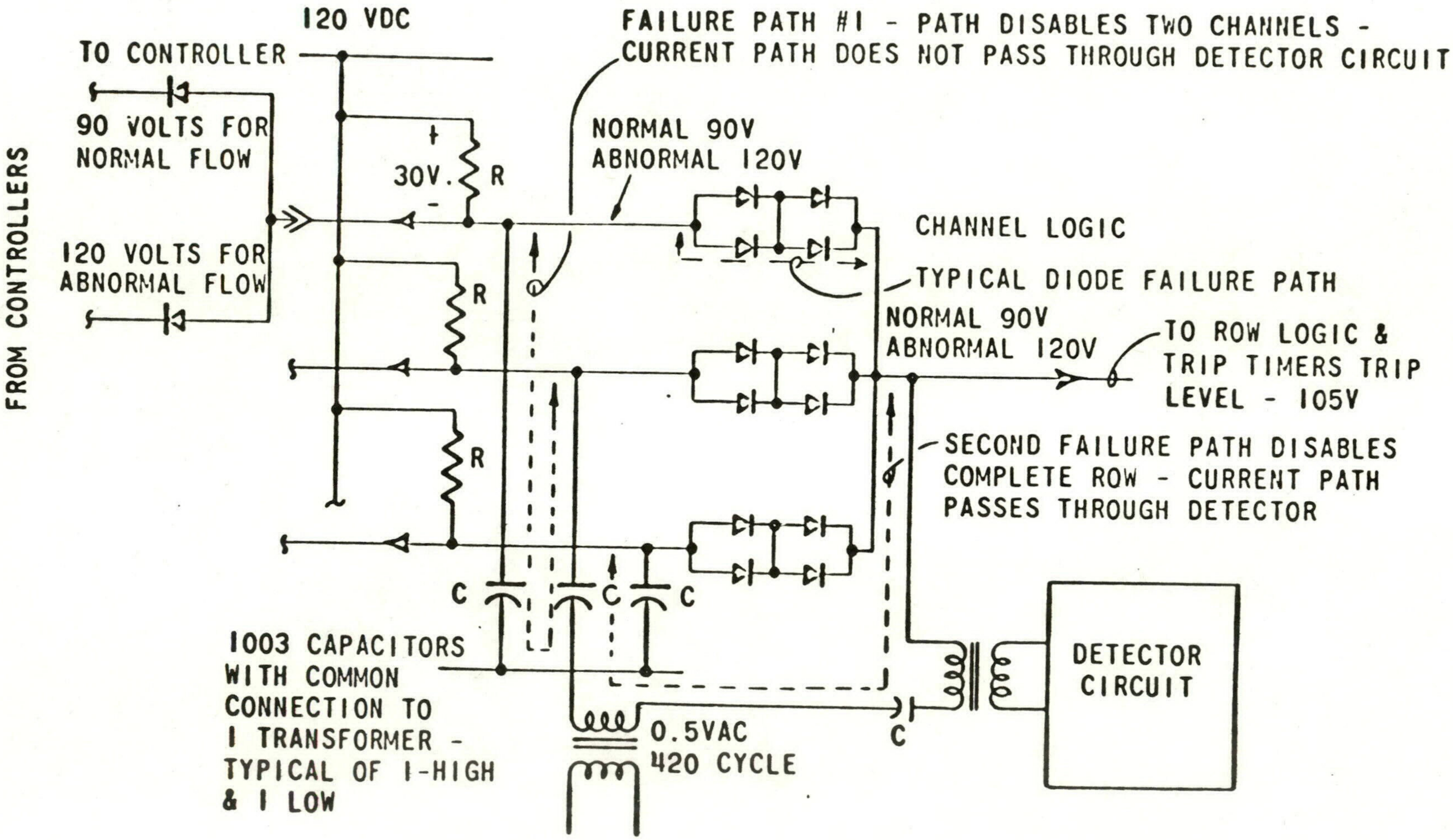
Failure of a single wire supplying 120 Volt AC power to the system could negate the trip function for the entire flow monitor system. In addition, this event is not monitored and alarmed for all locations where the event can occur. It was, therefore, recommended by the team that jumpers be placed between the downstream circuit terminals of the 120-Volt AC supply of all rows of channels to effect a loop bus arrangement, thereby providing greater reliability. In addition, the monitor that is provided on the system was recommended to be relocated to the extreme downstream circuit position so that in the event of loss of power to the system the monitor would function to scram the reactor. These recommendations were implemented prior to startup by Design Change 3123, "Flow Monitor Power Supply Redundant Buss."

(5) Inductrol Failure

Failure of a single variable transmitter unit (inductrol) will result in loss of the low en masse trip capability for the entire flow monitor system. If the low setpoint reference voltage drops to zero, the controller voltage cannot drop below the zero reference voltage and, therefore, no low flow trip can occur on any channel. The loss of output from the inductrol could occur unnoticed as it is not annunciated. A recommendation was made to install, on an interim basis, a voltmeter on the output of this transformer for surveillance until a sensor could be purchased or developed to monitor this function and initiate an annunciator in the 105-N control room. This recommendation was accomplished by Design Change 3122, "Flow Monitor Low En Masse Signal Failure Detector."

(6) Shorted Diode Detector

The flow monitor system has a circuit which continuously checks for shorted diodes by inserting a 400 cps sine wave upstream of the diodes and monitoring downstream to insure that the sine wave is rectified. The shorted diode detector has capacitors installed in it that have been calculated to be about 200 times more likely to fail then the diodes which the detector is installed to monitor. (See Figure 16.) A complete shorting failure of the diode quads would be detected by the shorted diode detector. However, since a combination of two capacitors shorting will produce the same event [negation of trip function for that row of process tubes by bleeding off trip level voltages (120 volts) back into the 90-Volt sink created by the controller whose capacitor is also shorted], it was questioned whether the shorted diode detector should be removed from the system or modified to be made more reliable. The Working Committee hesitated to remove the shorted diode detector



NOTE: 3 CHANNELS OF LOGIC SHOWN TYPICAL OF 1003 HIGH & 1003 LOW.

Figure 16. PROCESS TUBE FLOW MONITOR SHORTED DIODE DETECTOR

until further tests or checks to insure diode reliability could be proven and implemented. However, before startup, it was recommended that all capacitors in the shorted diode detector be tested for shorts. This test for shorted capacitors was accomplished by applying 120 VDC in the shorted diode detection circuit in such a way that, if any of the capacitors were shorted, a flow monitor trip would be received. No shorted capacitors were detected. Since two shorted capacitors are required to negate the flow monitor function, a safety factor of at least two has thus been demonstrated. Leaving the 120 VDC on the shorted diode detector circuit was considered so that in the event of a capacitor shorting failure, a flow monitor trip would be produced, and the reactor would shut down. It was concluded that this modification was not required.

It was also recommended that the individual diodes be tested annually to insure no single failure. Since all diodes were tested for shorts in June, 1970, it was not recommended that they be tested again prior to startup. It was further recommended that until a resolution as to whether to remove the shorted diode detector on a permanent basis or not was determined that the channel-by-channel functional trip test of the complete flow monitor system and the shorted capacitor test, described above, both be conducted before each startup or thirty days, whichever is longer.

No loss of a flow monitor trip function has yet been experienced in seven years of operation, due to a diode or capacitor failure. In fact, no failure of a capacitor or diode has yet been experienced in seven years of operation, due to a diode or capacitor failure. In fact, no failure of a capacitor or diode has yet been experienced in the flow monitor system. (Reference: Attachment C.)

The flow monitor system also incorporates an open diode detector circuit which inserts 100 ms pulses of 120 VDC upstream of the diode quads and monitors downstream to insure the pulses come through. The 100 ms duration of the test pulses is below the 0.5-second actuation response time of the flow monitor trip circuitry and, thus, no trip occurs due to the test pulses.

The open diode detector circuitry was also reviewed, but no problems were detected.

C. <u>KE Reactor Critical Safety Systems Review</u>

The systems reviewed for KE Reactor were the vertical safety rod system, the Ball 3X system, the last ditch and secondary coolant systems, the confinement system, and the No. 1 safety system and its primary inputs.

The Review Team consisted of representatives of the Technology Section, the Manufacturing Engineering Section, and the K Plant Section.

Each recommendation of the Review Team was assigned an action priority according to the following criteria:

<u>Priority 1</u> - Immediate reactor shutdown.

<u>Criterion</u> - Any single component non-failsafe failure that would disable the protective function.

<u>Priority 2</u> - Action taken next outage.

<u>Criterion</u> - The non-failsafe failure of a component in a redundant matrix when the failure is not annunciated and will reduce the system to a Priority 1 condition.

<u>Priority 3</u> - Action taken within four months (before March 1, 1971).

<u>Criterion</u> - Judgment - Inspection frequency considerations.

<u>Priority 4</u> - Action taken within the year (before November 1, 1971).

<u>Criterion</u> - Judgment - Inspection frequency considerations.

<u>Priority 5</u> - Further study required.

<u>Criterion</u> - Problem or action not adequately defined.

No problems were found which warranted Priority 1 action. All Priority 2 action items were completed during the following KE outage as requested. Longer range recommendations and their action priorities are summarized in Section VIII of this report.

1. <u>Vertical Safety Rods System</u>

The KE Reactor vertical safety rod system consists of 41 neutron poison rods divided into two separate groups which are located on the top of the reactor core. The VSRs are held out of the reactor during normal operation by energized latching solenoids. When a VSR system trip occurs, the latching solenoids are de-energized by the opening of redundant RDX relays which permits the VSRs to fall into the reactor core.

During the review of the VSR system, no single failure of components was found that could prevent the VSR system from accomplishing its protective function. Instances were found where a single component failure could prevent a single rod from dropping into the reactor.

There are normally open contacts of two RDX relays in the control circuit for each of the two banks of VSRs. The normally open contacts of one of the two relays in each circuit is in the positive leg of the DC circuit to the rod latching solenoids with the contacts of the other relay in the negative leg of the circuit. Opening of the contacts of either of the two RDX relays will result in the release of all rods in that bank.

The Review Team determined that the RDX relays were not given periodic EMS inspections as were other safety circuit relays. In addition, checks were not made to assure that all four RDX relays drop out on demand from the IXX relays during normal start-up VSR trip checks.

It was recommended that RDX relays be inspected and individually checked for proper operation during the next normal outage (Priority 2) and periodically thereafter by EMS (Priority 3). The relays were checked out as requested during the November outage of KE Reactor.

Each of the VSR junction boxes at the top of the reactor containing the latching solenoid circuitry also contains a positive DC power circuit supplying other functions which is not broken by the RDX relays. It is possible for an unannounced short circuit to exist in the junction box between the positive leg of the latching solenoid circuitry and the positive leg of the second DC supply circuit so that all latching solenoids in that bank of VSRs would be supplied with a positive DC power source which could not be broken by the RDX relays. Since the RDX relays open both the positive and negative legs of the VSR circuits when they de-energize, the single-short described could not alone prevent the functioning of a bank of VSRs. However, it would set up a condition where only the failure of the relay in the negative leg of the VSR circuit would be necessary.

It was recommended that a megger test to insure against connection between the 5D12, 5D13, and 5D14 circuits be accomplished during the next outage (Priority 2) and periodically thereafter by EMS (Priority 3).

The recommended megger tests of the VSR circuits were successfully accomplished during the November outage of KE Reactor.

2. 125-Volt DC Power System

Multiple grounds on either the "A" or "B" battery power supplies could create sneak circuits which effectively bypass control switches or relay functions. The circuits are presently automatically monitored by ground detection circuits and controlled shutdown procedures are implemented upon ground detection. The ground detectors, however, are not failsafe from the standpoint of sensing circuit failure. It was recommended that a ground detector circuit failure relay be provided at both "A" and "B" battery ground detectors to annunciate on the loss of detector circuit power (Priority 3). The addition of the circuit failure relay will assure that the ground detector is functioning at all times and that procedures are promptly implemented.

The Review Team also endorsed the present practice of testing the ground detectors once each shift.

3. Ball 3X System

a. Seismoscope Trip System

The Ball 3X system is designed to be activated by inputs from the reactor seismoscope system of sufficient intensity to de-energize the seismoscope time delay relays (STDRA and STDRB). The contacts of both seismoscope time delay relays (STDRA and STDRB) must open to initiate a ball drop. Slow flux decay in the relay core is controlled by a neutralizing coil which must continue to be energized if relay drop-out is to occur. Failure of either the neutralizing coil or its associated circuit for one relay will negate relay drop-out and prevent release of the balls on system demand.

It was recommended that a redesign of the circuit which considers relays of an improved type should be provided to upgrade the circuit. The circuit should be failsafe and redundant. An engineering request has already been initiated to provide an upgraded design (Priority 4).

b. Coolant Supply Riser Very Low Pressure Bypass Relays

The very low pressure bypass relays (LPRA and LPRB) are non-failsafe in design. Failure of one of these relays so that it de-energizes prior to a riser very low pressure demand trip will bypass the VLP circuit and prevent a release of balls.

It was recommended that a redesign of the circuit which considers relays of an improved type should be provided to upgrade the circuit. The circuit should be failsafe (Priority 4).

c. Riser Pressure Switches

Due to the way the circuit logic is designed, a single failure of one of two riser VLP pressure switches to open on demand can negate a ball release. In this circuit, both pressure switches, one in each of two redundant circuits, are required to drop-out to release the balls. Failure of a single switch in either circuit will negate the ball release.

In any condition when the VLP pressure switches are bypassed, a failure of one extremely low riser pressure switch to open during a riser break or loss of flow to a single riser can also negate a ball drop. Redundancy of extremely low pressure switches can only be relied upon when the rate of pressure decay on two or more risers is nearly the same.

It was recommended that the application of pressure switches for protection against a single riser failure be reviewed using the philosophy that a single component failure must not prevent a ball drop (Priority 5).

d. Arc Suppression

No arc suppressors are installed on the Ball 3X summation relays (3XA, 3XAA, 3XB, 3XBB). This creates the possibility of voltage transients and sustained arcs at the pressure switch contacts and eventual breakdown or welding of switch contacts.

It was recommended that a review be made of pressure switch current and voltage ratings versus summation relay operating characteristics to determine the compatibility of all components and arc suppression or other corrective measures be initiated as indicated (Priority 5).

e. Control Circuit Testing

Short circuits between the Ball 3X control circuit wiring which could bypass control switches or contacts can occur and be undetected for extended periods of time. The only assurance of freedom from shorted control circuits is through insulation testing and system functional tests.

More comprehensive functional testing was recommended to be included in the EMS testing program. These tests would include operation of the pressure switches in their control circuit by controlling pressure of the sensing line as opposed to the current technique of substituting a test switch for a pressure switch.

It was also recommended that a megger test be made of the insulation between common lines of the two halves of the Ball 3X circuits (A3 to B3 and AN3 to BN3) during the next outage (Priority 2) and periodically thereafter by EMS (Priority 4). Currently, megger tests are made from conductor to ground in each circuit but not between the two circuits.

The megger tests of circuits A3, B3, AN3 and BN3 were successfully accomplished during the November outage of KE Reactor.

4. Last-Ditch Cooling System - Control Systems

a. Ground Detector

The last-ditch cooling system diesel pumps starting circuits are ungrounded and share the 182-K Building battery power supply with other reactor and diesel pump control systems which are also ungrounded. Since the 182-K Building battery power supply system does not have a ground detector installed, the situation exists where multiple grounds in the various control circuits could create sneak circuits which would effectively bypass control switches or contacts and prevent one or more diesels starting on command.

The Review Team recommended that a full-time, failsafe ground detector should be installed in the 182-K Building battery system of sufficient sensitivity to detect a ground with a resistance three times that required to provide a current path adequate to maintain energization of the diesel starting air supply solenoids (Priority 3).

Testing of the diesel starting circuits for grounds was also recommended for accomplishment during the following outage (Priority 2). During the actual accomplishment of this testing, grounds were found in the starting air solenoid control circuits of diesel pumps 1 and 3. If one more appropriately located ground had existed, both diesels would have been prevented from starting on demand.

b. Control System Short Circuits

Short circuits between control circuit wires running in the same cable or terminating at a single switching device could occur and negate diesel starting by bypassing control switches or contacts.

It was recommended that EMS procedures be modified to include functional testing of each pressure switch while actually connected in the control circuit (Priority 4). The current mode of testing does not permit switching of the actual circuit load by the pressure switches. Summation relays are now held energized with a bypass switch and the pressure switches only operate an indicator lamp.

Testing to assure independence of common control power lines of the diesel start control circuits (5D6-P and 5D6-N at the KE Reactor; CB-45 and CB-55 at 182-K) was recommended to be accomplished during the next outage (Priority 2) and periodically thereafter by EMS (Priority 4).

The independence test of the diesel starting circuits was successfully accomplished during the November KE outage.

c. Arc Suppressors (LPSRA and LPSRB Relays)

No arc suppressors are installed on the relays operated by pressure switches. Failure of switch contacts may result if the inductive rating of pressure switches is inadequate for the impressed voltage.

An engineering review of pressure switch current and voltage ratings versus summation relay (LPSRA and LPSRB) characteristics was recommended to insure the compatibility of devices and to initiate designs for arc suppression or other corrective measures as indicated (Priority 5).

5. Last-Ditch Emergency Cooling System - Mechanical

Single events affecting the diesel engines, crosstie check valves, or the V-73 valve could disrupt the emergency cooling water system to the KE Reactor.

Other single component failures could cause the loss of a diesel-driven pumping unit; however, there are three pumping units, only two of which are required in an emergency situation.

a. Fire Protection of Diesel Engines - 182-K

The review indicated that a fire starting at one diesel engine could spread to the other two engines and cause the loss of all three pumping units.

The Review Team recommended that the fire protective system be upgraded as shown on SK-1-71925 to include a dry pipe sprinkling system and fire walls separating the diesel engines and the starting air accumulators. These improvements are documented in DUN-4683 and DUN-5453 and are considered necessary to provide the assurance that a fire starting at one diesel engine will not spread to the other two before the reactor could be placed in a "safe" (non-operating) condition.

b. V-73 Valve

Though somewhat improbable, accidental closing of the manually-operated V-73 valve would totally invalidate the emergency cooling water system. Since the valve closure is not annunciated, it could go unnoticed in the control room.

The recommendation was made that the closing of the V-73 valve be controlled and the valve tagged to provide further assurance that inadvertent valve closure will not occur and block emergency cooling water to the reactor. Though the operation of the valve is normally done under strict procedural control, having a warning tag on the valve would alert an operator that having the valve in a closed position could have serious consequences.

The manual operator of the V-73 valve has been disabled and tagged out of service.

c. Crosstie Check Valves

Failure of either the A riser crosstie check valve or the B riser crosstie check valve to open would prevent cooling water reaching half the reactor in an emergency situation.

Although the check valves are currently functionally checked by EMS every six months, uncertainty that the check valve will open tends to grow as the time since the last test increases. Decreasing the testing interval reduces both the degree and the duration of that uncertainty.

A further reason for increasing the frequency of the functional check is to "exercise" the valve. This exercise breaks up any corrosion buildup that could cause the flapper hinges to bind or cause the flapper to stick in the seat.

The Review Team recommended that the frequency of the functional tests of the crosstie check valves be increased from the present six-month EMS requirement interval to a one-month interval (Priority 3).

6. Confinement System - Instruments

a. Power Loss Annunciation

The confinement annunciator panel is an essential link between the control room and the confinement system. Loss of power to the confinement annunciator panel prevents actuation of alarms which result from control instrument power loss or equipment malfunction. Without annunciation, a fog spray trip demand could exist undetected during the interval between daily EMS checks.

The Review Team recommended that annunciation be provided to warn of power loss to the confinement annunciator panel. It was suggested that a relay be installed to monitor the power at the annunciator cabinet and actuate a 105 control room annunciator in the event of a power loss to the panel (Priority 3).

b. Contact Meter Power Loss

Loss of power to the contact meter will disable the instrument and prevent automatic operation of the fog spray system. Manual control of the fog spray system could be initiated provided annunciation of a trip demand occurs; however, annunciation of a fog spray trip demand also relies upon normal functioning of the meter.

The Review Team recommended that annunciation be provided for loss of power in the contact meter (Priority 3). It was also recommended that the plug-in power cord for this instrument be modified to prevent its inadvertent removal from the power receptacle (Priority 3).

c. Iodine Monitor

Loss of sample flow to the iodine monitor prevents this instrument from performing its primary function. An iodine monitor sample pump alarm (ΔP switch) has been installed to annunciate loss of pumping action. The sample flow rate is checked daily per EMS requirements. The existing EMS does not, however, provide for testing or calibrating the protective device.

It was recommended that the EMS include testing and calibrating requirements for the iodine sample pump alarm (Priority 3).

7. Confinement System - Other

a. Ventilation Tunnel Flooding

Plugging of the No. 2 Building drain system could result in undetected flooding at the -24 foot level of the main ventilation tunnel. Water from the storage basin can also cause flooding at the -24 foot level if the floor drain in the sump becomes plugged. A partial or complete blockage of the confinement exhaust air system could result from the uncontrolled buildup of water.

The recommendation was made to install a liquid level monitoring system in the sump at the -24 foot level of the main building exhaust tunnel which will provide annunciation of abnormal water level on the main annunciator panel in the 105-KE control room (Priority 3).

b. Seal Pit Overflow

Undetected buildup of water in the inlet seal pits to the confinement system filter banks could cause wetting of the absolute filters and result in filter damage and loss of efficiency. An undetected buildup of water in the filter bank outlet seal pits could result in a partial or complete blockage of the confinement exhaust system.

Annunciation of the liquid level in the seal pits is now located in the 117 Building. It was recommended that this instrumentation be modified to independently indicate abnormal water level in all four seal pits and provide annunciation of the abnormal water level in the 105-KE control room (Priority 4).

c. Confinement System Solenoid Valves

The KE Reactor confinement system was designed with single solenoid valves to operate the confinement system equipment. Among the single components most vulnerable to failure are the single solenoid valves which operate the fog spray groves valves, shut off the water to the storage basin, close the damper in the air supply line to the rear face, and operate the supply and exhaust fan dampers. Failure of any one of these components could negate that part of the confinement system.

The recommendation was made to investigate the feasibility of installing redundant solenoid valves in the system (Priority 5).

d. Service Water Pressure Switch

The effectiveness of the rear face fog spray could be negated if the service water pressure dropped too low and was not detected. The pressure switch now being used to monitor and annunciate low service water pressure is sluggish and has too wide a reset band.

The Review Team recommended that the procurement of a different type pressure switch could be investigated. In addition, an EMS revision was recommended which would require the periodic calibration of the service water pressure switch. The present EMS only requires a functional test of the switch. The EMS calibration procedures should also include the frequency of calibration as well as specify the pressure setting for low pressure annunciation (Priority 3).

8. Pressure Monitor System

The primary coolant pressure monitor system is designed to be fail-safe in its operation. Frequency EMS checks and functional tests ensure high reliability for the system and its components. Single component failures that can prevent a pressure monitor system trip for a single process tube can result from sensing line plugging or gauge failure. Routine EMS testing of sample lines and gauge response minimizes the failure potential. The temperature monitor system provides some measure of backup control action in the event process channel flow reduction is not detected by the pressure monitor.

Other potential single component failures involve the wiring of the Panelit gauge switches. The wires connecting ends of each row or column of switches appear on adjacent terminals of barrier type terminal boards in the coincidence trip relay cabinet. Likewise, the summation relay contact wires (5D1-15 and 5D1-17) appear side by side. Short circuits between these adjacent points could negate trips from a total row or column of Panelits. Since the cabinet is normally closed during reactor operation and EMS functional tests are made on these circuits before each startup, the potential for inadvertent bridging of adjacent terminals is considered to be remote.

No recommendation for corrective action was made.

9. Flux Monitor System

No non-failsafe circuits were found in the investigation.

The method and frequency of checking the safety circuit with the flux monitors, as covered by the Equipment Maintenance Standard, were questioned for adequacy.

The Review Team recommended that consideration be given to revising EMS-KE-105-I to either (1) check all combinations of safety circuit action by the four channels on a definite schedule, or (2) cover a larger percentage of the total when randomly selected on the pre-startup schedule (Priority 5).

10. High-Speed Scanning System

No non-failsafe circuits were found in the investigation of the safety circuit. It should be noted that a detailed review of the logic system was not attempted.

The EMS pre-startup safety circuit check, using the various scram modes, does not specify the status of the safety circuit and HSSS bypass switch during the test. A revision should be made to EMS-KE-108-I to require that the No. 1 safety circuit be made up and that the bypass switch be in the OFF position during the performance of that part of the pre-startup checks covering safety circuit trips (Priority 3).

11. Seismoscope System

The seismic protective system is highly unlikely to be disabled by a single component failure. Component redundancy, system design, and the EMS inspections provide assurance that the system will function on demand.

12. Pump Under Power System

No single failure of components was found which could disable the protective function of the underpower safety circuit.

13. No. 1 Safety Circuit (1XX String)

No situation was found to exist where failure of a single component would disable the protective function of the No. 1 safety circuit.

14. Secondary Cooling System

The adequacy of the K Reactor secondary cooling system was reviewed in-depth in 1968, using detailed fault-tree analysis techniques. This study was documented by DUN-4461, "Comparative Reliability Analysis - K Reactor Secondary Cooling System," and was updated by

DUN-4461, Supplement 1, dated January 12, 1970. The equipment changes recommended by this document have been made.

Because of these recent reliability analyses, and because the functional adequacy of the secondary cooling system's major components are tested monthly by trip of the normal BPA power, this system was not reviewed, in detail, by the Review Team.

VIII. N REACTOR PRE-START TESTING PROGRAM

As a result of the conclusions, recommendations, and modifications resulting from the varied investigations conducted as consequences of the September 30 scram, a special team of Technical and Operations Divisions personnel was formed to prepare a special "Pre-Start Test Procedure" which would verify the proper integrated functional performance of all channels and all components of N Reactor safety systems.

A. Pre-Startup Test Procedures

The special pre-startup test procedures were designed to test each safety subsystem in its entirety from the primary sensors through all combinations of trip logic possible through to the final trip contacts. [An exception is the zone temperature monitor system which due to the large number of combinations possible (any three trips out of one-hundred and eight sensors) had a special test performed at low power levels after startup.]

The systems and subsystems checked in this manner are given in Table IV.

B. Safety System Input Circuit Isolation

Each safety circuit DC power supply was verified to be electrically separate from any other power supply by isolating the individual circuit to a battery cart and imposing a ground on the system and verifying that the ground did not show up on the rest of the battery systems. (The battery system is an ungrounded system.) Periodically the battery cart and battery system were tied together and grounded to verify the validity of the test.

C. Other System Checks

The following systems were also checked prior to startup and have been reported previously in other sections:

Flow Monitor System. Diode Capacitor Check . . . . Section VII
Rod Safety System. Diode Check . . . . Section IV
Confinement System . . . . Section VII
Flow Monitor System. Channel Continuity . . . . Section VII
Fog Spray & Emergency Cooling System Diesels . . . . Section VII
Condensate Pump Test. Cell No. 6 . . . . Section V
ECS V-3 and V-4 Valves Accumulator Capacity . . . . Section VII

TABLE IV

PRE-STARTUP TEST PROCEDURE DETAILS

ROD SAFETY SYSTEM

Manual Scram. Nuclear Console

Manual Scram. AA Console

Fast Period Intermediate Power Level

Excessive Neutron Flux. Intermediate Power Level

Excessive Neutron Flux. High Power Level

Excessive Power Rate of Rise

Very High Average Power Level

Outlet Boiling

Zone Temperature Monitor

Rod Cooling Water Low Flow

Electrical Power Failure

Primary Loop Pressure

Extremely Low Pressurizer Level

Main Steam Header High Pressure

Circulating Raw Water Low Flow

Extremely Low Surge Tank Water Level

BALL SAFETY SYSTEM

Seismoscope

High Flux After Scram

Reactor Critical After Scram

Ball System Pushbutton

ECS Actuation

BALL BACKUP TO SLOW RODS

Single Rod Response

Two Rods in Column

Any Three Slow Rods

TABLE IV (Cont'd)

FAST ROD WITHDRAWAL INTERLOCK

Period Trip Test
Low Count Rate Test
Reactor Critical IRFM Test

POST SCRAM ACTIONS

Pump slowdown
Steam Generator Bypass Valves
Afterheat Removal Pumps
Boiler Feed Pump Start
High Pressure Injection Pump Start
Pressurizer Heater Cutout

EMERGENCY COOLING SYSTEM

Ball and Rod Safety Systems Trips
V-3 and V-4 Actuation
Pump Shaft Rotation
Pump Low Speed Trip Combinations
ECS Low Flow Trip Combinations
ECS High Temperature Trip
High and Low Pressure Trips
Confiner High Pressure Trip
ECS Master Actuation Functional Tests

FINAL INSPECTION PRIMARY SENSORS

D. Design Change Acceptance Tests

Each of the design changes performed has a section defining the acceptance test required to verify the design. All acceptance tests were completed prior to startup.

E. Equipment Maintenance Standards

All startup and other applicable (e.g., monthly) Equipment Maintenance Standards were performed prior to startup.

F. Startup Checklist From Extended Outage

All mandatory startup requirements outlined in Operating Procedure No. 48-145-000-389-101 were completed prior to startup.

IX. POST-STARTUP ACTION ITEMS

The recommendations in each of the areas investigated as a result of the September 30 scram were noted in the appropriate previous sections of this report. The action taken on each of the pre-start recommendations was also provided.

Many of the recommendations require continuing action and these actions are summarized below. The Working Committee of the President's Review Council will be responsible for following the progress on the remaining recommendations and to assure that action is appropriate and timely.

A. N Reactor Action Items

Action By		Action	Due
Manager, N Plant	1.	Add primary loop extremely high pressure trip and main steam header pressure trip to extended outage check list.	
Manager, N Plant	2.	Change extended startup procedure to require check of ball breaker trip from <u>each</u> input source during each extended outage including this one.	
Manager, Manuf. Eng.	3.	Periodically check ability of ECS valve air accumulators to actuate valves alone.	
Manager, Manuf. Eng.	4.	Trip-check each flow monitor channel using signal at transducer input to controller every thirty days or each extended outage following sustained operating periods of more than thirty days.	

Action By		Action	Due
Manager, N Plant	5.	Re-do Design Changes 3122 and 3123 to improve workmanship.	
Manager, Process Section	6.	Review document HW-84402, dated 10/7/64, "Audit of N Reactor Control Systems."	December, 1970
Manager, Manuf. Eng.	7.	125 VDC system ground spikes. Is there a mechanism where a short-to-ground could defeat the safety systems fed by the 125 VDC system? Study and report.	
Chairman, Working Committee	8.	Revision of procedures to require(?) manual insertion of HCRs.	
Manager, Manuf. Eng.	9.	Review EMS and other periodic checks to insure adequate checkout to avoid future problems; for example, periodic check of rod system diodes, relays, surge suppressors, etc. a. Rods b. Balls c. ECS d. Flow Monitor e. Confinement f. Instrumentation systems providing trip inputs to the above systems.	
Manager, Eng.	10.	Further testing to determine most probable diode failure mechanism. What can fail the diodes? Will failure of one diode in the four diode cluster tend to increase the likelihood of failure of others?	February, 1971
Manager, Eng.	11.	Study alternative solutions to rod scram failure problem and recommend long-range solution to Working Committee.	January, 1971
Manager, Eng.	12.	Investigate problem of parallel 5-second timer contacts in "high flux after scram" input to ball system hanging up and preventing input.	DC being routed for approvals.
Manager, Manuf. Eng.	13.	Modify EMS on int. range flux monitors to specifically check the six isolation diodes integrity in the self-check circuit per vendor requirements.	

Action By		Action	Due
Manager, Process Section	14.	Review advisability of having a flow monitor master bypass switch controlled by procedure only.	January, 1971
Manager, Manuf. Eng.	15.	Install potting on bare flow monitor terminals connecting each individual channel.	
Manager, Manuf. Eng.; Manager, Technology Section	16.	Complete DC to remove or correct shorted diode detector in the flow monitor system.	
Manager, Technology	17.	IEEE 279 compliance analysis of all safety circuits.	
Manager, Process Section	18.	Complete Garlock gasket material investigation.	November, 1970
Manager, Technology	19.	Review desirability of electrically allowing safety rods to be extracted from reactor when safety circuit is not made up?	
Manager, Eng.	20.	Should provision for monitoring the power on the scram solenoids exist?	Investigated and no action required.
Manager, Manuf. Eng.	21.	Test all flow monitor capacitors in shorted diodes detector every thirty days or during each extended outage following sustained operating periods of more than thirty days.	
Manager, Manuf. Eng.	22.	Should the power calculator circuitry and/or procedures be revised to account for inaccuracies at low power levels? (450 mw_{th} indicated - 782 mw_{th} actual.)	
Manager, Manuf. Eng.	23.	Annual EMS test of diodes in flow monitor circuit using thermopile device to detect open diodes.	
Manager, Eng.	24.	Correct blueprint showing bypass of scram test contacts by H8 contacts.	Completed.

Action By		Action	Due
Manager, N Plant	25.	Made end-to-end functional check of redundant ECS circuits a mandatory check list and EMS item.	
Manager, Eng.	26.	Why is RWS-2 nominal operating pressure now 165 when it was previously 190 psig?	Investigated & report prepared.
Manager, Eng.	27.	Recheck all diesel starting air accum-umulator check valves? Why are they leaking?	Will investigate after DC 3112 has been installed.
Manager, Eng.	28.	Check out the piping between the receivers and the diesel air accumu-lators and the RWS-2 accumulator.	Complete.
Manager, N Plant	29.	Revise operating procedure (50-710-MSFT-13) to insure adequate check-out of check valves.	
Manager, Technology	30.	Resolve need to repair air system or prop up 105 Zone 1 exhaust backup gate. a. Clean and lubricate guides for smooth operation. b. Isolate the two nitrogen backups from each other so that loss of one will not take out the other.	Complete. Design Change 3129.
Manager, N Plant	31.	Place signs on all 105-N and 109-N sensing line openings into Zone 1 reading "DO NOT PLUG - CONFINEMENT AND FOG SPRAY SENSING LINE INLETS."	
Manager, N Plant	32.	Place tags on all 105-N and 109-N sensing line isolation valves reading "DO NOT CLOSE VALVE - CONFINEMENT (FOG SPRAY) PRESSURE SWITCHES."	
Manager, N Plant	33.	Review the adequacy of protective systems to prevent the introduction of foreign objects into reactor systems.	
Manager, Manuf. Eng.	34.	EMS 603 should incorporate tests to periodically verify operation of RDRV-802-1 and 2.	

Action By		Action	Due
Manager, Manuf. Eng.	35.	Develop a permanent design for replacing the voltmeter on the flow monitor inductrol failure alarm.	

B. <u>KE Reactor Action Items</u>

Action By		Action	Due
Manager, Manuf. Eng.	1.	Revise EMS to periodically visually inspect KE Reactor VSR summation relays 1RDX, 2RDX, 3RDX and 4RDX for proper operation.	March 1, 1971
Manager, Manuf. Eng.	2.	Revise EMS to periodically megger test to insure against connections between VSR circuits 5D12, 5D13, 5D14, and 5D15.	March 1, 1971
Manager, KE Reactor	3.	Revise 125 VDC power system ground detectors on both A and B battery circuits to annunciate loss of ground detector circuit power.	March 1, 1971
Manager, Manuf. Eng.	4.	Redesign the Ball 3X seismoscope trip circuit to provide a redundant and failsafe circuit incorporating improved relays.	November 1, 1971
Manager, Manuf. Eng.	5.	Redesign the Ball 3X VLP circuit to provide a failsafe circuit incorporating improved relays.	November 1, 1971
Manager, Manuf. Eng.	6.	Review the application of pressure switches in the Ball 3X system circuitry for protection against a single riser failure using the philosophy that a single component must not prevent a ball drop.	
Manager, Manuf. Eng.	7.	Review Ball 3X circuitry pressure switches current and voltage ratings versus summation relay operating characteristics to determine the compatibility of all components. Initiate arc suppression or other corrective measures as indicated necessary.	

Action By	Action	Due
Manager, Manuf. Eng.	8. Revise EMS to periodically megger the insulation between Ball 3X circuits A3 and B3 and circuits AN3 to BN3.	November 1, 1971
Manager, Manuf. Eng.	9. Revise EMS to functionally test Ball 3X pressure switches while connected in the circuitry by actually varying sensing line pressure.	November 1, 1971
Manager, KE Reactor	10. Install a full-time, failsafe ground detector in the 182 Building battery system.	March 1, 1971
Manager, Manuf. Eng.	11. Review last-ditch cooling system diesels starting circuitry pressure switches current and voltage ratings versus summation relay characteristics to determine the compatibility of components. Initiate arc suppression or other corrective measures as indicated necessary.	
Manager, KE Reactor	12. Upgrade fire protection of last-ditch cooling system diesels to include a dry pipe sprinkling system and fire walls between diesels as shown on Sketch SK-1-71925.	
Manager, Manuf. Eng.	13. Revise EMS to functionally test last-ditch cooling system pressure switches while connected in the circuitry by actually varying sensing line pressure.	November 1, 1971
Manager, Manuf. Eng.	14. Revise EMS to periodically test to insure the independence of diesel starting circuits 5D6-P, 5D6-N, CB-45, and CB-55.	November 1, 1971
Manager, Manuf. Eng.	15. Revise EMS to increase functional testing frequency of last-ditch cooling system crosstie check valves from once each six months to once each month.	March 1, 1971

Action By	Action	Due
Manager, KE Reactor	16. Provide control room annunciator to warn of power loss to the confinement system annunciation panel.	March 1, 1971
Manager, KE Reactor	17. Provide annunciator to warn of power loss to the confinement system contact meter.	March 1, 1971
Manager, KE Reactor	18. Modify plug-in power cord to contact meter to prevent inadvertent removal from the power receptacle.	March 1, 1971
Manager, Manuf. Eng.	19. Revise EMS to include periodic testing and calibration requirements for the iodine monitor sample pump alarm pressure switch.	March 1, 1971
Manager, KE Reactor	20. Install a liquid level monitoring system in the sump at the -24 foot level of the main building confinement system exhaust tunnel which will provide KE control room annunciation of abnormal water level.	March 1, 1971
Manager, KE Reactor	21. Modify liquid level monitoring systems for the confinement filter banks inlet and outlet seal pits to provide KE control room annunciation of abnormal water level in <u>each</u> seal pit independently.	November 1, 1971
Manager, Manuf. Eng.	22. Investigate the feasibility of installing redundant solenoid valves in the KE confinement system.	
Manager, Manuf. Eng.	23. Investigate the procurement and installation of a different type pressure to monitor service water pressure.	March 1, 1971
Manager, Manuf. Eng.	24. Revise EMS to require periodic calibration of the service water pressure switch.	March 1, 1971

Action By		Action	Due
Manager, Manuf. Eng.	25.	Revise EMS KE-105-I to either (a) check all combinations of flux monitor safety circuit by the four channels on a definite schedule or (b) cover a larger percentage of the total when randomly selected on the pre-startup schedule.	
Manager, Manuf. Eng.	26.	Revise EMS KE-108-I to require that the No. 1 safety circuit be made up and the high-speed scanning system bypass switch be in the off position during the performance of that part of the pre-startup safety circuit trip checks.	March 1, 1971

X. SUMMARY CONCLUSIONS AND RECOMMENDATIONS

As a result of the studies and investigations discussed in previous sections of this report, the following summary conclusions have been reached.

Rod Scram Failure

1. Failure of the rods to be inserted into the reactor in response to the September 30 scram signal was due to the continuing and undesired energization of all rod scram solenoids by power from the control rod "off" and "withdrawal" circuit. Power was supplied through a sneak circuit created by the shorting failure of two sets of in-series diodes in the scram solenoid circuitry of control rod No. 59 when that rod was electrically assigned to "off".

2. Failure of the four rod 59 in-series diodes was probably due to an accidental and momentary shorting during maintenance activities which created a high current through all diodes simultaneously. Existence of such a mechanism was demonstrated by laboratory techniques.

3. Even with the noted failure of the rod scram circuitry, the reactor operator always had the capability of inserting rods into the reactor by using the manually-operated rod-gang switches, the "all-rod insertion" pushbutton, or the "rod setback" switch.

Number 6 Drive Turbine

4. The scram of N Reactor at 5:25 a.m., September 30, 1970, was initiated by momentary low flow indications on several process tubes as a result of normal pressure and flow transients created by the automatic trip-off of the No. 6 primary pump drive turbine. At no time did the flow in any process tube fall below that required to insure nuclear safety.

This would also have been true even if the drive turbine had tripped off with the reactor at the nominal full-power level of 4000 mw_{th}.

5. The automatic trip-off of the No. 6 drive turbine occurred in response to an indication of inadequate vacuum in the No. 6 turbine exhaust condenser which was caused by an excessively high level of condensate in the condenser hotwell.

6. Blockage of the inlet screens of the No. 6 condenser condensate pumps by pieces of Garlock brand gasket material reduced the condensate pumping capacity below that required to maintain a proper hotwell condensate level.

7. The exact sequence of events which permitted the introduction of the Garlock gasket material into the No. 6 turbine condensate system cannot be positively established. Traces of the Garlock material in the shell of the No. 6 condenser indicate that it was probably inadvertently introduced during maintenance activities inside the condenser shell.

Ball Hopper No. 3

8. The actuation failure of the No. 3 ball hopper was due to the undetected loss of one of four ball bearings during assembly of the hopper following the June maintenance work; the missing ball prevented the actuation during the September 30 scram.

Testing

9. During the investigation of the September 30 scram, instances were found where system problems existed and not been detected by existing test procedures. This resulted from a lack of a system test procedure which simultaneously checks out the total functioning of a system from the sensing element input to the final system output.

XI. FINAL REVIEWS

A. DUN President's Council Review

On October 9, 1970, the Working Committee presented to the President's Review Council a summary of the results of the investigation and the corrective actions which were being implemented. The Committee also reviewed its efforts in supporting the AEC-RL Investigating Team.

At the conclusion of the presentations, it was recommended by the Committee that upon the successful completion of the specified prestart action items and with the understanding of the appropriate completion of the longer range action items, the N Reactor should be permitted to resume normal operation. It was estimated that in view of present progress, if startup permission was granted immediately, the remaining startup checks should permit reactor criticality to be achieved as early as October 11, 1970.

The General Manager and his Review Council staff members concurred with the Committee recommendations, but noted, however, that due to the nature of the problem formal approval of the AEC would be required prior to returning N Reactor to normal operation.

B. AEC Reviews

To achieve the AEC approval, a presentation was made to the AEC-RL management on October 12, 1970, which described the problem, the investigation, and the DUN action plan. Although it was indicated that some additional long-range recommendations would probably be forthcoming in the areas of maintenance control and systems testing, AEC-RL generally concurred with the DUN conclusions and recommendations. AEC-RL pointed out, however, that prior to N startup, an endorsement by AEC headquarters would be required.

On October 14 and 15, personnel from the DUN and AEC-RL investigating teams traveled to Washington D.C. and provided presentations to Commissioner W. E. Johnson, the Division of Production, the Office of Operational Safety, the ACRS, and the DRL. As a result of these presentations, formal authorization from the AEC was received on October 16, 1970 to resume normal operation of N Reactor.

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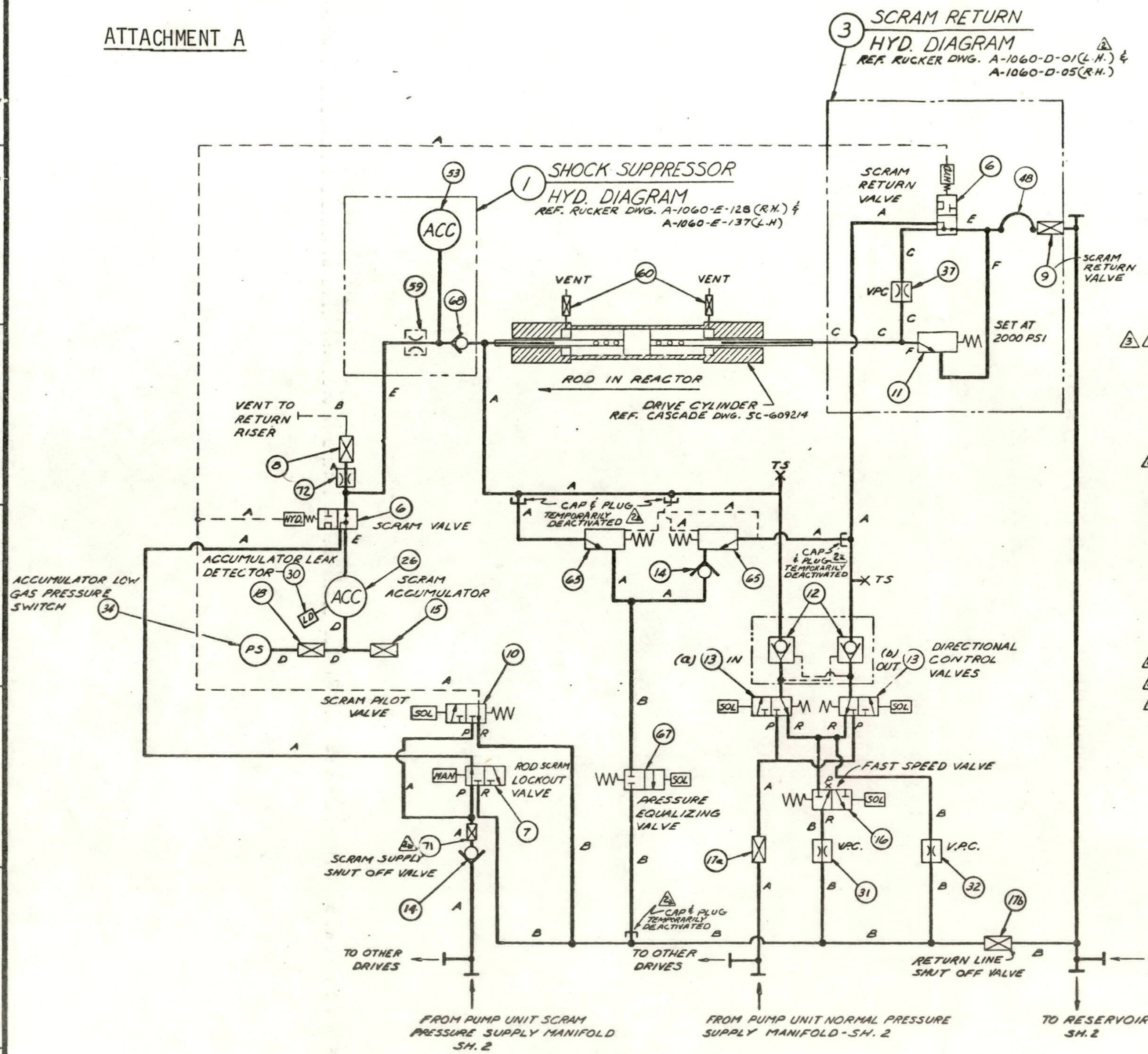
DUN-7342

ATTACHMENT A



2 CONTROL UNIT HYD. DIAGRAM
REF. RUCKER VALVE RACK DWG. A-1060-F-06(R.H.) & A-1060-F-07(L.H.)

LINE	SIZE	MATERIAL
A	3/8 X.049	STL. TUBE
B	3/8 X.049	SST TUBE
C	1 1/2 X.120	STL. TUBE
D	1/4 X.049	STL. TUBE
E	1" SCH80	STL. PIPE
F	1/2 X.049	STL. TUBE

TABLE APPLIES TO THIS SHEET ONLY.

QTY.				PT. NO.	DESCRIPTION
				40	
				41	
1				42	PISTON PUMP, MODEL 80LAL, RACINE - SECO
2				43	PISTON PUMP, MODEL 80LAM, RACINE - SECO
1				44	PISTON PUMP, MODEL C-5, RACINE - SECO
1				45	MOTOR, G.E. 7 1/2 HP. 1800 RPM, FRAME 254, 440V. 3Φ 60 CYCLE, OPEN DRIPROOF, NEMA-F1 ASSY.
2				46	MOTOR, G.E. 7 1/2 HP. 1800 RPM, FRAME 254, 440V. 3Φ 60 CYCLE, OPEN DRIPROOF, NEMA-F1 ASSY. W/ENDBELL FOR RACINE-SECO MODEL 80LAM
1				47	MOTOR, G.E. 15HP. 1200 RPM, FRAME 324, 440V 3Φ 60 CYCLE, OPEN DRIPROOF - NEMA F1 ASSY.
	1			48	HOSE - 1" - 150807-16-27 - AEROQUIP CORP.
				49	
1				50	DIAL THERM & SWITCH, 4 1/2 DIAL, FIG. 3056 - U.S. GAUGE
2				51	SUCTION STAINER, B 1 1/4 -20, 100 MESH - MARVEL
1				52	LOW LEVEL CONTROL, NO. TF-52 SPST, MAGNETROL
			1	53	ACCUMULATOR - MODEL ACH-300315 - AMER. BOSCH
2				54	VALVE, CHECK 3/4" KEPNER PROD. CO., NO 1312C-1
1				55	PRESSURE GAUGE, TYPE 9 FMSU, 4 1/2 DIAL - MARSH
1				56	HEAT EXCHANGER (S.S.) SSCF-501-2 - ROSS
1				57	WATER REGULATOR, 246T06, 85° TO 125°F - PENN
				58	
			1	59	ORIFICE PLATE, A-1060-B-132 - RUCKER CO.
		2		60	VENT VALVE, CASCADE DWG. 1C-609627
1				61	MOTOR - G.E. 1/2 H.P. 1800 RPM 440V. 3PH. 60CYC.
1				62	COUPLING - SIER BATH - SIZE 7/8 STD.
1				63	HOSE, AEROQUIP - J.I.C. SWIVEL #255604-12-84
5				64	OIL SAMPLE VALVE - 1924 - MFA-05 - RUCKER
		2		65	RELIEF VALVE, #1028-6-SAE - FLUID CONTROLS
				66	
		1		67	2 WAY VALVE, #3-C5-2-110/130 - FLUID PWR. ACCES.
			1	68	CHECK VALVE, NO. 2216 - KEPNER
1				69	PUMP, GREASEN WITH FOOT MOUNT #TB-3-75-50K-CW
4				70	VALVE, GLOBE, HOKE 316 S/S #1190 - 3/8"
		1		71	VALVE, SWAGELOK ENDS, WHITEY 1VS6-CS
		1		72	ORIFICE, (CONNECTOR REWORK) DWG. H-1-37471

NOTES:

1. THIS DRAWING SHOWS SCHEMATIC HYDRAULIC PIPING & EQUIPMENT FOR ONE DRIVE UNIT. THERE ARE FORTY-SIX (46) DRIVES ON THE RIGHT SIDE, AND FORTY-ONE DRIVES ON THE LEFT SIDE, EACH DRIVE IS CONNECTED TO THE PRESSURE AND RETURN HEADERS AS NOTED.
2. ACCUMULATOR SHALL BE DESIGNED AND STAMPED TO CONFORM TO THE ASME CODE.
3. SHEET TWO OF THIS DRAWING SHOWS SCHEMATIC HYDRAULIC PIPING & EQUIPMENT FOR ONE POWER UNIT. THERE ARE TWO POWER UNITS, ONE ON THE RIGHT SIDE AND ONE ON THE LEFT SIDE.
4. HYDRAULIC CONTROL EQUIPMENT AND PIPING ASSEMBLIES ARE SHOWN ON RUCKER COMPANY DRAWINGS - SEE BPF-11200.
5. PUMP AND FILTER UNIT FOR FILLING RESERVOIR AND FOR CLEANING OIL BY RECIRCULATING THROUGH THE FILTER.
6. FILL RESERVOIR PIPING AS REQUIRED WITH HYDRAULIC OIL. STANDARD OIL CO. CHEVRON NO. 11-OC.
7. EQUIPMENT SHOWN ON THIS DRAWING SHALL BE PROCURED OR FABRICATED IN ACCORDANCE WITH SPECIFICATIONS HWS 6596.
8. PART NO. 20A IS PART 20 MODIFIED. THE CONTROL SEAT SHALL BE REAMED TO $.209^{+.000}_{-.001}$ IN. DIA. (REF DENISON SERVICE BULLETIN SVR2-3C)(BPF 11200)

QTY.				PT. NO.	DESCRIPTION
			X	1	SHOCK SUPPRESSOR HYDRAULIC DIAGRAM (SCRAM END)
		X		2	CONTROL UNIT HYDAULIC DIAGRAM
	X			3	SCRAM RETURN HYDRAULIC DIAGRAM
X				4	POWER UNIT HYDRAULIC DIAGRAM
1				5	PLUG 1 1/2 I.P.S - STL.
	1	1		6	2 WAY VALVE, P/OPR - MODEL PV-33D - PART NO. 219433-1 - MAROTTA VALVE CORP.
		1		7	3 WAY VALVE, MAN - MODEL LV74MR - PART NO. 219793-04 - MAROTTA VALVE CORP.
		1		8	NEEDLE VALVE, MARSH 1/4 NPT - 1924 FFG
	1			9	BALL VALVE - #455YB(SE) 1" - WORCESTER VALVE CO.
		1		10	3 WAY PILOT VALVE - MODEL MV-74-MS - PART NO. 219813-04 - MAROTTA VALVE CORP.
	1			11	RELIEF VALVE #625-10-8-2 REPUPLIC MFG.
		1		12	DUAL CONTROL CHECK VALVE #221253-04-2 MAROTTA
		2		13	3 WAY VALVE, MV-74-MR - PT. NO. 220783-04 MAROTTA VALVE CORP.
		2		14	CHECK VALVE #488-6SS-2 REPUBLIC
		1		15	GAS CHARGING VALVE #VA 200 932 - AMER. BOSCH
		1		16	3 WAY VALVE - MODEL MV-74-MR - PART NO. 220783-04 MAROTTA VALVE CORP.
4				17	NEEDLE VALVE, MARSH 3/4 NPT - 1924 FFG
		1		17a	NEEDLE VALVE, MARSH 1/4 NPT - 1924 FFG
		1		17b	NEEDLE VALVE, MARSH 1/4 NPT - 1924 FFG
2		1		18	NEEDLE VALVE, MARSH 1/4 NPT - 1924 FFG
4				19	RELIEF VALVE, DENISON 3/4 NPT - RV061303A
4				20	UNLOADING VALVE, DENISON 3/4 NPT - RU061303A (SEE NOTE #8)
1				21	GATE VALVE, CRANE 1 1/2, 200 PSI #438 - BRONZE
2				22	CHECK VALVE 'AA' - D6-175 - 3/4"
6				23	GATE VALVE - 3/4" 200 PSI #438 - CRANE
4				24	CHECK VALVE 'AA' - D6-175 - 3/4"
1				24a	CHECK VALVE 'AA' - D6-175-P - 3/4"
4				25	NEEDLE VALVE, MARSH - 1924 FFG - 3/4" NPT
		1		26	ACCUMULATOR - MODEL ACD 300289 - AMER. BOSCH
1				27	LIQUID LEVEL GAUGE - 29-RW-20 - JERGUSON
2				28	GAUGE SNUBBER - 1/4" NPT #1112 MD - ASHCROFT
1				29	BREATHER - W/BASE - AF 701, BASE #56836 - PUROLATOR
		1		30	LEAK DETECTOR - NO. 12427-A, GEMS CO.
		1		31	FLOW CONTROL VALVE, MOD. 320-2-1.50 WATERMAN
		1		32	FLOW CONTROL VALVE - MODEL 320 - NO. 320-2-0.270 WATERMAN ENGINEERING CO.
2				33	ACCUMULATOR - TYPE - ACD, 4 GAL. - NOTE 2 AMER. BOSCH
2		1		34	PRESSURE SWITCH, MELETRON #312-135
1				35	PRESSURE GAUGE - 4 1/2 DIAL, TYPE 9, FMSU 0 TO 3000 PSI - MARSH
~~1~~				36	~~FILTER-BREATHER-STRAINER, CAT. #58 RACINE HYD.~~
	1		•	37	FLOW CONTROL VALVE, A-61145 - VINSON MFG.
3				38	FILTER #5-S-13460-16-A-1 - BENDIX
1				39	RESERVOIR - DWG. A-1060-D-73 OR 74 RUCKER CO. - NOTE 6

H-1-27993 1/2 3

REF. RUCKER SCHEMATIC DWG. A-1060-D-70

NO.	DESCRIPTION	REV. BY DATE	APP'D BY	FOR	DATE
3	E-3, E-10 ON SHT. 2, CK'D FOR AS BUILT. ADD: ITEM 54 THIS SHT. & ZONES	F.E 11-26-68			
2	CHECKED FOR AS BUILT (EMP #210) CHG. SETTINGS ON ITEMS 20 & 20A CHANGED FILTRATION FROM 10 TO 5 MICRONS. ADDED - ITEM 71, & 72 (EMP #152) (EMP #169) PLUGGED LINES, ADDED ITEM 70. & NOTE #8 & RUCKER DWG. REF. ITEM NO. 54 REMOVED WAS: VALVE, BAILEY CO. NO. 20	F.E 11-1-67			12-1-67
1	52 & 36 (C) ADDED NOTE 3 THRU 7 (B) DELETED PTS. 40, 41, 49. (A) REDRAWN - ADDED SH. 2	5/29/61		IPD PDO	7/6/62

REVISIONS

EPN 105N-1900

DWG. NO.	DRAWING TITLE
BPF-11200	HORIZONTAL ROD SYSTEM
HWS 6596	ROD DRIVE & HYD CONT. SPECS.
H-1-28000	HOR. ROD DRIVE
H-1-31613	ROD POSITION INDICATION EQUIP. ASSY.
H-1-28580	ROD DRIVE INSTRUMENT ENG. DIAGRAM

REFERENCE DRAWINGS

NEXT USED ON H-1-28033

CHECK PRINT | COMMENT PRINT

SCALE

DRAWN CARTER
CHECKED JEB
ISSUED T. OEUSNER
DES. ENG. G. A. NEWELL
PROJECT CA1816 INST. NO. 8679

CLASSIFIED BY T. Dlumer DATE May 23, 62

APPROVED FOR CONST.
APPROVED FOR PURCHASE
APPROVED FOR DESIGN
G. A. NEWELL PDO
W. J. MORRIS IPD

CLASSIFICATION NONE

U. S. ATOMIC ENERGY COMMISSION
HANFORD ATOMIC PRODUCTS OPERATION
GENERAL ELECTRIC

HORIZONTAL RODS
HYDRAULIC SCHEMATIC

BLDG. NO. 105N INDEX NO. 1900

DWG. NO. H-1-27993 SHT. NO. 1 SHEETS 2

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ATTACHMENT B

ROD DRIVE CONTROL SYSTEM

Reference: HW-69000, Vol. II, Page 21.13.4-1

The rod drive control system provides the means for operating personnel to select the type of rod service and to manually control rod movement. This is accomplished by the operation of three groups of control switches: the service selector switches, the rod control selection pushbuttons, and the rod control switches. Incorporated in the control circuitry are various interlocks to provide reactor safety. These controls and their functions are detailed below:

A. The Service Selector Switches - There are 86 key-locked, five-position, SB type control switches located on the rod service selector switch panel. These switches are used to assign each rod to one of the following five different types of service:

1. Assigned Safety Circuit - Rods assigned as safety rods:

   a. Cannot be withdrawn until the safety circuit is made up, which in turn energizes the scram solenoid valves. With the scram solenoid valves de-energized, the normal rod hydraulic controls will not function to move the rods, as neither side of the rod hydraulic cylinder can be pressurized.

   b. Must be full-out before rods assigned to manual and setback can be withdrawn. When all safety rods reach their full-out proximity switches, rod-out control power is available to those rods assigned to manual and power setback.

   c. Cannot be run in with the control circuit unless reassigned to manual, and can then be run in even though all safety rods are not full-out. An exception to this is the all-rod insertion pushbutton, which will insert all rods at either the slow or fast control speed with the exception of rods assigned to OFF or to rod withdrawal.

   d. Are bypassed from the setback circuit. This is really a second bypass in that safety rods are, in addition, NOT assigned to the setback circuit.

   e. Automatically cancel their rod control switch (operating switch) selection by removing power from the selection pushbuttons for each rod as the full-out proximity switch is reached.

   f. Light individual blue indicating lights in the selector pushbuttons for the corresponding rods to indicate their safety rod status.

   g. Scram into the reactor, as do all rods not assigned to OFF, at the rod scram speed upon a safety circuit trip.

h. Actuate a red indicating light on the selector pushbutton panel to indicate that all safety rods are not full-out; and then turn this red light off and an amber light on when all safety rods are full-out.

2. Assigned Power Setback - Rods assigned to power setback:

   a. Are automatically inserted stepwise into the reactor at the slow rod speed upon a setback circuit trip. This is accomplished by periodically energizing the normal rod in slow control circuit to the in slow solenoid valve.

   b. Are used for reactor control in the same manner as rods selected for manual service.

   c. Cannot be withdrawn until the safety circuit is made up and all safety rods are full-out.

   d. Scram when the safety circuit is tripped.

   e. Light individual blue indicating lights in the selector pushbuttons for the corresponding rods to indicate their setback status.

3. Manual - Rods assigned to manual:

   a. Cannot be withdrawn until the safety circuit is made up and all safety rods are full-out.

   b. Are manually positioned with the rod control system for reactor power level control.

   c. Scram when the safety circuit is tripped.

   d. Are placed in the power setback circuit when assigned to one of the six control switches. A blue indicating lamp lights in the selector pushbuttons to indicate that the rod is in a status of controlled setback. This function can be bypassed with another of the backlighted selector pushbuttons which is labeled, "Controlled Setback Bypass."

4. Rod Withdrawal - To assign a rod to withdrawal, the service selector switch is first turned to the rod withdrawal position. This lights an individual red indicating light in the selector pushbuttons for the corresponding rod to indicate the withdrawal status (not yet attained). An annunciator also drops which warns of any rod in the withdrawal or off status. Next, to actuate the interlocks associated with withdrawal, the selector switch must be pulled out momentarily; five seconds later, amber lights (one on the service selector switch panel and one on the selection pushbutton panel) blink on for half a second and then off, signifying that the rod withdrawal and off energizing circuit has timed out. At this time, a red light (one light for all 86 rods) on each of the above panels lights to indicate that the withdrawal interlocks for the rod are in effect; namely,

that the rod scram solenoid valve is being held energized through a bypass circuit and that power to the rod in slow solenoid valve is removed.

Assigning a rod to withdrawal is independent of the safety circuit status except for five seconds after a safety circuit trip, during which time power is off to the rod withdrawal and off energizing circuit.

Thus, rods assigned to withdrawal:

a. Can be withdrawn (with the safety circuit either made up or tripped) but cannot be inserted with either the normal rod control or the all-rod insertion pushbutton.

b. Will scram if four or more rods are withdrawn from the full-in limit at the same time that five or more ball hoppers are locked out.

c. Will not scram provided items in Condition 2 are met. However, rod cannot be moved during five seconds following scram.

d. Will have their withdrawal interlocks de-energized upon a BPA electrical outage (loss of current from the CX panel).

5. Off - To assign a rod to off, the selector switch must be turned to off and pulled out. The same energizing circuit is used as for rod withdrawal; in fact, the rod withdrawal interlocks are energized. The same annunciator drops and the same amber lights time out with actuation of the same red lights signifying pickup of the relay which places in effect the withdrawal interlocks. In addition, the red light which lights in the corresponding rod selector pushbuttons indicates that the rod off interlock has locked-in to hold the rod scram solenoid energized through a bypass circuit and prevent the rod from scramming.

Thus, rods assigned to off:

a. Will not scram when the 1x safety circuit is tripped. However, this interlock is bypassed if four or more rods are out at the same time five or more ball hoppers are locked out.

b. Will be counted as rods that fail to scram to 75% in for the ball column backup to rods circuit.

c. Are inoperable electrically.

ATTACHMENT C

SELECTED COMPONENT FAILURE RATE DATA

Failure rate information was compiled for specified N Reactor components in terms of (1) actual demonstrated N Reactor data, and (2) the best available industrial data. This information is summarized in tabular form below followed by a detailed discussion of each component on how the failure information was obtained.

	Failure Rate in Failures Per Million Hours	
Component	Demonstrated[1] by N Experience	Industrial by MIL-HDBK-217A[2]
1. Rod Diodes		
a. 4-diode unit	1.2	--
b. Single diode	0.8	.75
2. Flow Monitor Diode	0.01	0.27
3. Flow Monitor Capacitor	0.025	0.23
4. Inductrol	25[3]	6.0

[1]Demonstrated to have at least this low of a failure rate at 95 percent confidence level.

[2]Calculated by MIL-HDBK-217A methods at a best-estimate confidence level.

[3]This is not considered to be a valid failure rate because of limited unit hours of operating experience.

Discussion

1. Rod System Diodes - These are the diodes that are in series with the rod scram solenoids. These are GE IN538 silicone diodes not purchased to specific quality control requirements but were selected by sorting for high peak inverse voltage rating. These diodes are mounted and sealed four to a unit with 87 units or 348 individual diodes.

   a. Operating Unit Hours(4)

      4 unit diodes - 87 x 60,000 = 5,220,000 hours
      single diode - 348 x 60,000 = 20,880,000 hours

   b. Failures

      4 unit diodes - 2
      single diode - 10

   The first reported tests of these diodes were October, 1970, indicating that these failures could have been present since installation.

[4]Based on seven years of operation or about 60,000 hours.

c. Demonstrated Failure Rate

4 unit diode - 1.2×10^{-6} failures/hour
single diode - 0.8×10^{-6} failures/hour

d. Industrial Data - taken from curves provided by MIL-HDBK-217A which accounts for these diodes being operated at about 1/4 rated current with a K factor of 1.5 for ground (versus vehicular or airborne) application.

failure rate = 0.75×10^{-6} failures/hour.

2. Flow Monitor Diode - These are special quality Pacific Semiconductors IN484A type silicone diodes and are individually open mounted. There are 8,000 of these diodes in this application.

a. Operating Unit Hours

8,000 x 60,000 = 480,000,000 hours

b. Failures

There have been no reported failures; however, one was replaced as a suspect. One failure was assumed. Last tested early 1970.

c. Demonstrated Failure Rate

IN484A diode - $.01 \times 10^{-6}$ failures/hour.

d. Industrial Data - Same as Item 1(d); however, normally there is no current through these diodes so stress is quite low. A K application factor of 1.5 was used.

IN484A diode - 0.27×10^{-6} failures/hour.

3. Flow Monitor Shorted Diode Capacitors - Two capacitors are used, both are Sprague type 118P, one a .1 ufd (2000) and the other .01 ufd (64). These were combined for failure rate information for a total of 2064 capacitors. These capacitors were purchased to quality control requirements as were the flow monitor diodes.

a. Operating Unit Hours

2064 x 60,000 = 123,840,000

b. Failures

None reported - tested October, 1970 for the first time.

c. Demonstrated Failure Rate

Capacitor - 0.025×10^{-6} failures/hour

d. Industrial Data - Same source as Items 1 and 2. Stress factor F_s was calculated as follows:

$$F_s = \frac{\text{applied voltage}}{\text{rated}} = \frac{90}{200} = .45$$

This factor and a 25° C ambient were used to obtain a failure rate of

capacitor - 0.23×10^{-6} failures/hour.

4. Inductrol Variable Transformer - These units (2), used for low and high en masse voltage supply adjustments, were made by GE and rated at about 40 amps. We could not find failure data on Inductrols per se; however, they are considered similar to a cross between a normal transformer and an autotransformer as far as failure data is concerned. These units are considered to be lightly stressed for our application and were purchased to a specification requiring a failure rate of less than $.01 \times 10^{-6}$ failures per hour.

a. Operating Unit Hours

$2 \times 60{,}000 = 120{,}000$

b. Failures

None reported.

c. Demonstrated Failure Rate

Inductrol - 25×10^{-6} failures/hour at 95 percent confidence (limited by unit test hours)

Inductrol - 6×10^{-6} failures/hour by MIL-HDBK-217A best estimate method.

d. Industrial Failure Rate - Could not find data on Inductrols; therefore, an average value for medium-size transformers was used.

Inductrol - 6×10^{-6} failures/hour

ATTACHMENT D - DUN-7321

QUAD DIODE FAILURE MODES EXAMINATION

INTRODUCTION

It was the purpose of this investigation to duplicate the potential failure modes and subsequent overload effects to 4JA420BX8 diode quad packages. This study consists of the following discrete tests:

I. Reverse Voltage for Diode Junction Failure

II. Forward Surge Current for Diode Junction Failure

- a) Continuous 8 amp Test
- b) 2-6 Amp Test
- c) Unlimited Current Test
- d) Circuit Breaker Test

III. Physical Effects with Steady State High Reverse Currents

IV. Physical Examination of Failed Diode Quad Packages

- a) Rod 38 Diode Quad Package
- b) Circuit Breaker Limited Current Failure
- c) 8 Amp Forward Current (30 sec. breakdown)
- d) 2-6 Amp Forward Current Breakdown
- e) Reverse HV Breakdown
- f) "Unlimited" Forward Current Breakdown

V. Diode Junction Reverse Voltage Breakdown Characteristics as Determined by Curve Tracer

VI. Diode Junction Failure in Simulated Circuit Without Surge Suppression

SUMMARY

In summary, all physical evidence examined in these tests indicates that the recent quad diode failures experience at N Reactor are a result of momentary shorting of these diodes across 125 VDC with the unlimited current being shut off in 20 to 50 milliseconds by the installed in-line circuit breaker.

TEST RESULTS

General:

Diode quad packages, Type 4JA420BX8 and other system test component parts used in the following tests were obtained from shop spare parts stock. The tests were conducted by DUN engineering personnel in an established development facility utilizing established test procedures and nearly ideal test conditions.

Test I:

Each junction in diode quad package (designated "A") was subjected to increasing reverse voltage pulses until the junction failed. The reverse voltage required to break down the junctions ranged from 750 to 1000 VDC. The current was limited to 30 MA maximum by an electronic circuit breaker in the exciting power supply. The resulting shorted diode forward and reverse resistance was nominally 0.1 ohms on two junctions, 0.28 ohms in a third junction and 0.5 ohms in the fourth junction. This diode quad package was then set aside for use in Test III.

Test II:

a) Continuous 8 Amp Test:

Each junction in test diode quad package (designated "B") was subjected to a forward current of 8 amperes until each junction shorted. It required between 20 and 30 seconds to obtain a junction short with a 8 ampere forward current. All four of these shorted junctions displayed a forward and reverse resistance of nominally 0.01 ohms. This diode quad package was then set aside for use in Test III.

b) 2-6 Amp Test:

The following test was conducted to determine what minimum average forward diode current would cause a shorted junction. A single diode junction of a diode quad package (designated "E") was connected to a DC current supply to supply the following currents as a function of time. The current was increased in one ampere steps until the junction failed as shown in the following table:

Time	Current in Amps.	Junction Voltage Drop
Start	2	1.138 volts
10 min.	3	1.214
15 min.	4	1.344
20 min.	5	1.521
26 min.	6	No data
26 min + 30 sec.	6	Junction shorted

Within one minute after the junction shorted and current was removed, the forward and reverse junction resistance was measured at 0.01 ohms.

c) Unlimited Current Test:

The following test was conducted to determine the effect of unlimited forward current (escept by circuit resistance) on the four diodes connected in series.

In this test all four diodes in a diode quad package were connected in series and momentarily placed across a 24 volt DC battery source (100 ampere capacity). Three diode quad packages were tested in this manner with the following results:

Each of the three diode quad packages had three shorted junctions and one open circuit junction. The open circuit junction was not necessarily associated with any particular location of the diodes in the series circuit. During the shorting process (with a screwdriver) a very small spark was observed in two cases and none in the third.

d) Circuit Breaker Test:

The following test was conducted to determine if a Heineman 5 amp circuit breaker would prevent total junction destruction and the open circuit condition observed in Test (b).

A Heineman breaker of the same type (10 amp Curve 3) and characteristics as the 1B9 breaker could not be obtained in time available so a 5 ampere unit with a curve 1 response was substituted. The curve 1 delay time to break is approximately the same as a curve 3 for surge currents 10 times breaker rating. Heineman breaker data indicates the breaker opened in approximately 20 to 50 milliseconds for the estimated 50 amperes peak load.

The test breaker and a single diode from the diode quad package were wired in series with a 24 volt battery. The circuit was then completed by tapping the connecting clip leads together. Before the clip leads could bounce apart the circuit breaker operated. A check of the diode junction resistance (.01 ohms) showed it to be shorted. The diode was then set aside for Physical Examination in Section IV.

Test III:

The two diode quad packages (designated "A" & "B") which were failed in Tests I and II-a were connected in series across a DC current supply capable of producing a reverse current through all diode junctions of 8 amperes.

After the diode quad packages had been subjected to an 8 amp reverse current, the following results were noted as a function of the elapsed time: At 13 minutes solder melted and dripped out of octal base connectors in package "A" which had been previously shorted by high reverse voltages. At 14 minutes solder melted and dripped out of octal base connectors in package "B". At 1 hour elapsed time both packages were very hot to touch, were giving off a lot of gas, and showed slight damage to the top of the case. The cases, however, were not deformed.

After five hours of continuous operation with a reverse current of 8 amperes both diode quad packages were hot to touch and conducting current without excessive voltage drop (2.7V, package "A"; 1.6V, package "B"). The cases were slightly discolored but were not deformed or cracked. At this point the test was discontinued.

Test IV:

General:

The following sectioning procedure was used to examine all diodes. The diode quad package was first put in a lathe chuck and the outer bakelite and inner potting epoxy machined away until the outer metal shells of the diodes were visible. The remaining assembly was then sectioned into pie-shaped pieces with a tungsten carbide saw. Each pie section was then identified with the two corresponding octal pin numbers, that is, 1-2, 3-4, 5-6 and 7-8, representing the diode locations in the package. The surrounding epoxy mass was then removed by squeezing it in the jaws of a bench vise. The residual epoxy was then broken off in small pieces until the plastic enclosed diode was free of all debris. The removed diode was then rechecked with an ohmmeter on the low resistance scale for an open or shorted diode junction. The flattened anode tube was cut off with a jeweler's hacksaw between the flat section and the glass seal to free the anode wire from the tube.

The compression weld holding the two halves of the diode case together was then sawed away and the top half containing the glass seal, anode tube, and internal ceramic insulating sleeve were lifted away revealing the lower half of the diode case. The diode junction was then examined under a 30 power stereo microscope for any evidence of damage.

Color microphotographs were taken of all the examined junctions. The examined diode junctions were then placed individually in a transparent package which was coded to match the microphotographs.

The general appearance of a normal sectioned diode is as follows: The lower half (cathode terminal) has a slight depression with a .050" x .050" silicon chip bonded to it with an unidentified metallic alloy. The 0.01 inch diameter anode wire is fused to the top center of the chip.

a) Rod 38 Diode Quad Package

1. Junctions 1-2, 3-4, 5-6 and 7-8 were shown to be shorted before disassembly.
2. After sectioning, junctions 1-2, 3-4 and 7-8 appeared normal with no evidence of any heating or debris inside the diode case.
3. Junction 5-6 showed smoke residue across the glass seal on the outside of the case and glass particles around 1/4 of the anode tube.

The junction broke apart during disassembly, leaving the lower half of the silicon attached to the lower case and the upper half of the silicon chip attached to the anode wire.

The silicon chip had a surface appearance where the anode wire attaches that looked identical to the other three diode junctions in the diode quad package.

The general appearance of the Rod 38 shorted diode junctions was identical to a normal control diode junction.

b) "Circuit Breaker Limited" Current Failure (Junction 1-2 package E)

1. The junction was shown to be shorted before disassembly.
2. The junction and anode lead appear normal.

c) 8 Amp Forward Current 30 Sec. Breakdown (Junction 3-4, package E)

1. The junction was shown to be open before disassembly.
2. When the flattened anode tube was sawed off the anode lead inside was found to be loose.
3. The chip had a cavity in the center that matched the piece of silicon on the end of the anode lead.
4. One-fourth of the top half of the silicon chip was missing.

d) 2-6 Amp Forward Current Breakdown (Junction 5-6, package E)

1. The junction was shown to be shorted before disassembly.
2. The top surface of the chip, about .040" of the anode lead, starting at the chip and the bottom case were covered with fine white particles.

e) Reverse HV Breakdown (Junction 7-8, package E)

1. The junction was shown to be open before disassembly.
2. The chip had a cavity in the center that matched the piece of silicon on the end of the anode lead. One quarter of the top half of the chip was vaporized.
3. The inside of the ceramic insulator showed metal particles and discoloration of a spot about .050" in diameter.

f) "Unlimited" Forward Current Breakdown

One of the diode quad packages which failed by passing unlimited current through the series diodes, as per Test II-c, was examined.

1. The (1-2) junction was tested and shown open before disassembly.
2. When the flattened anode tube was sawed off the anode lead inside was found to be loose.
3. Examination of the anode lead showed it had melted in two, a portion of the anode lead was still attached to the silicon chip. Parts of the anode lead were spattered across the case, silicon chip, and a 0.010" diameter spot on the inside of the ceramic insulating sleeve. Pieces of silicon were also found adhering to the ceramic sleeve.

Junction 3-4 of the same diode quad was inspected:

1. The junction was tested and shown shorted before disassembly.
2. The anode lead (.010 diameter) was attached to the exploded top of the silicon chip. The area of attachment was about .003" in diameter.
3. Silicon debris was all over the inside of the case.

The remaining junctions 5-6 and 7-8 of the above quad-diode package have been retained.

Test V:

The reverse diode breakdown characteristics of diode quad package "C" were examined with a modified Textronik curve tracer. The results of this test are as follows:

Junction	Max. Reverse Voltage With no Breakdown	Reverse Current for Applied Breakdown Voltage
1-2	900V	1.6 MA @ 1000V
3-4	900V	1.2 MA @ 1000V
5-6	800V	1.5 MA @ 900V
7-8	1300V	1.5 MA @ 1400V

A 5K resistor was placed in series with the diodes to protect them from permanent failure. As this test was non-destructive, the diode quad package (designated "C") was used in Test II.

Test VI:

A diode quad package (designated "D") was placed in a circuit simulating the rod control circuits in which the diode quad package had failed. The hydraulic valve's coil surge suppressor was removed. The contacts of a mechanical breaker were actuated to operate the solenoid valve 72,000 times. After this exercise all diode junctions were tested with a curve tracer as per Test V and found normal.

To insure that a diode failure under the above simulated conditions was not prevented by the high actual reverse voltage characteristics of diode quad package "D" (750-1000 volts) a second test was conducted using a lower quality diode in place of the diode quad package "D".

In this second test a single 1N91 germanium diode with a measured reverse voltage characteristic of 75 volts was placed in series with the rod scram solenoid. During this test the normal 125 VDC was applied on and off (approximately 40,000 times) with the surge suppressor removed from across the solenoid coil. The 1N91 diode was then examined and shown to have the same voltage vs. current characteristics as before the test was started.

CONCLUSION

Laboratory testing and physical examination of normal diode quads, diode quads failed on-reactor and diode quads failed in all conceivable ways has resulted in the following conclusions:

1. Spare replacement diode quad package demonstrate 4 to 5 times manufacturer's reverse voltage rating (200 volts).

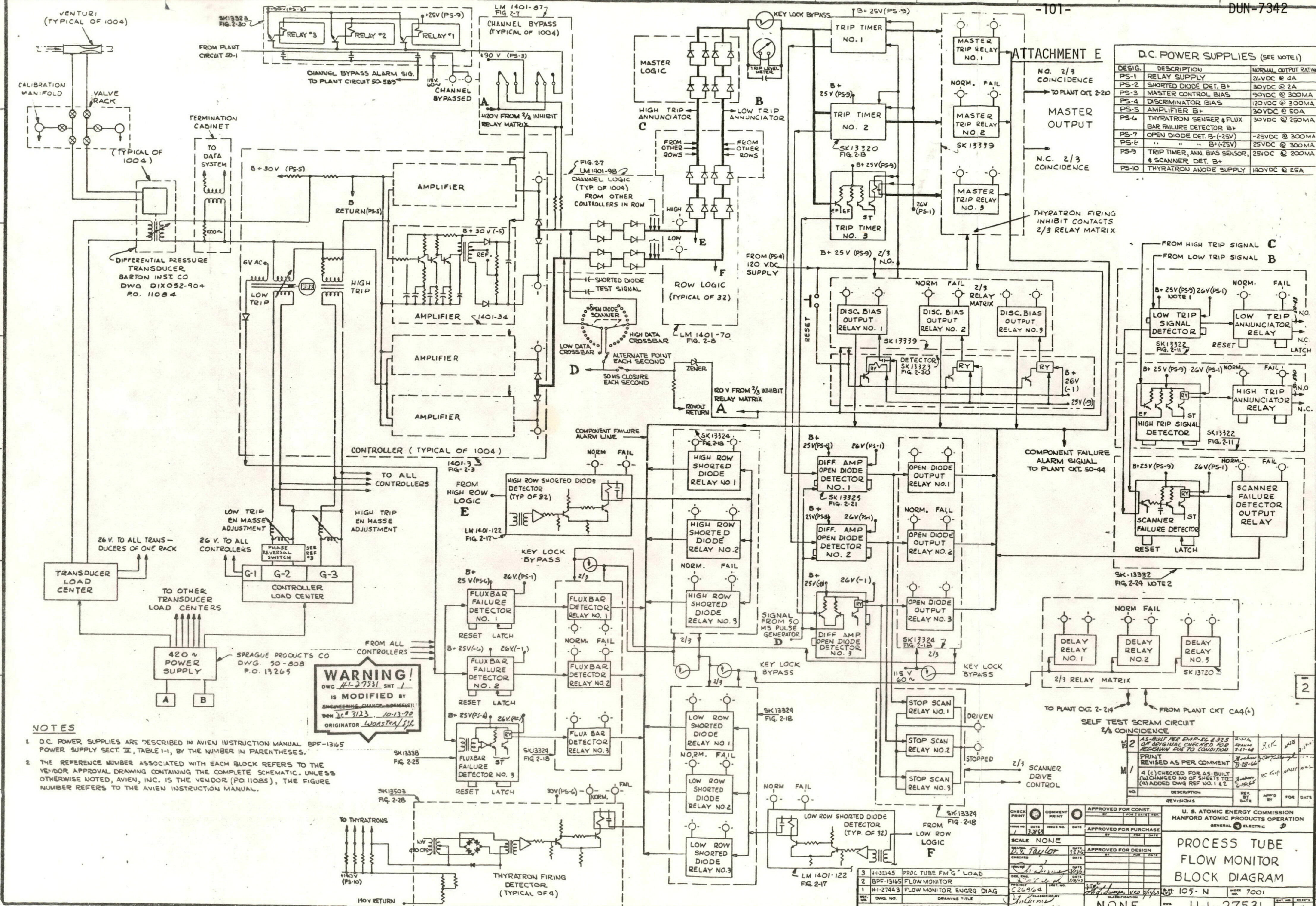
2. Replacement diode quad packages have approximately 4 times the forward current capacity specified by the manufacturer (.75 amps) at ambient conditions. This is approximately 20 times the normal (0.2 amps) rod circuit current requirements.

3. Diode quads failed by excess forward currents (approximately 50 amps) but limited to a time duration of 20 to 50 milliseconds, duplicates on-reactor failed diodes in electrical characteristics and physical appearance.

4. Diode quads failed by high inverse voltage and unlimited forward currents do not have the same electrical characteristics or physical appearance as "incident" diodes.

5. Diode quads do not fail with the surge suppressor removed from across the solenoid coil with 72,000 simulated circuit actuations.

ATTACHMENT E



D.C. POWER SUPPLIES (SEE NOTE 1)		
DESIG.	DESCRIPTION	NORMAL OUTPUT RATING
PS-1	RELAY SUPPLY	26VDC @ 4A
PS-2	SHORTED DIODE DET. B+	30VDC @ 2A
PS-3	MASTER CONTROL BIAS	90VDC @ 300MA
PS-4	DISCRIMINATOR BIAS	120VDC @ 300MA
PS-5	AMPLIFIER B+	30VDC @ 50A
PS-6	THYRATRON SENSER & FLUX BAR FAILURE DETECTOR B+	30VDC @ 250MA
PS-7	OPEN DIODE DET. B-(-25V)	-25VDC @ 300MA
PS-8	" " " B+(+25V)	25VDC @ 300MA
PS-9	TRIP TIMER, ANN. BIAS SENSOR, & SCANNER DET. B+	25VDC @ 200MA
PS-10	THYRATRON ANODE SUPPLY	140VDC @ 25A

NOTES

1. D.C. POWER SUPPLIES ARE DESCRIBED IN AVIEN INSTRUCTION MANUAL BPF-13165 POWER SUPPLY SECT. IV, TABLE 1-1, BY THE NUMBER IN PARENTHESES.
2. THE REFERENCE NUMBER ASSOCIATED WITH EACH BLOCK REFERS TO THE VENDOR APPROVAL DRAWING CONTAINING THE COMPLETE SCHEMATIC. UNLESS OTHERWISE NOTED, AVIEN, INC. IS THE VENDOR (PO 11085), THE FIGURE NUMBER REFERS TO THE AVIEN INSTRUCTION MANUAL.

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NO.	DESCRIPTION	REV. BY DATE
2	AS-BUILT PER EMP-56 & 325 OF ORIGINAL CHECKED FOR REDRAWN DUE TO CONDITION	RWA 2-27-68
1	PRINT REVISED AS PER COMMENT	8-28-66
	4 (c) CHECKED FOR AS-BUILT (b) CHANGED NO OF SHEETS TO (a) ADDED DWG REF NO. 1 & 2	6-15-65

REVISIONS

REF. NO.	DWG. NO.	DRAWING TITLE
3	H-1-32145	PROC TUBE FM "G" LOAD
2	BPF-13165	FLOW MONITOR
1	H-1-27443	FLOW MONITOR ENGRG DIAG

REFERENCE DRAWINGS

SCALE NONE
DRAWN D. F. Taylor
PROJECT C26964
DATE 3-27-63
CLASSIFICATION NONE

U. S. ATOMIC ENERGY COMMISSION
HANFORD ATOMIC PRODUCTS OPERATION
GENERAL ELECTRIC

PROCESS TUBE FLOW MONITOR BLOCK DIAGRAM

105-N INDEX NO. 7001
DWG. NO. H-1-27531 SHT. NO. 1 SHEETS 4