U. S. ATOMIC ENERGY COMMISSION
REGION I
DIVISION OF COMPLIANCE

Report of Inspection

CO Report No. 219/70-7

Licensee: JERSEY CENTRAL POWER AND LIGHT COMPANY
Oyster Creek 1
License No. DPR-16
Category C

Dates of Inspection: October 13-16, 1970

Dates of Previous Inspection: September 24-25, 1970

Inspected by: R. J. McDermott 12/1/70
R. J. McDermott, Reactor Inspector Date

Reviewed by : R. T. Carlson 12/2/70
R. T. Carlson, Senior Reactor Inspector Date

Proprietary Information: None

SCOPE

Type of Facility: Boiling Water Reactor

Power Level: 1600 MWt

Location: Forked River, New Jersey

Accompanying Personnel: Mr. R. T. Carlson, Senior Reactor Inspector, CO:I on October 13-16, 1970
Mr. F. J. Nolan, Senior Reactor Inspection Specialist, CO:HQ on October 13-15, 1970
Mr. J. G. Keppler, Senior Reactor Inspection Specialist, CO:HQ on October 13-16, 1970

Scope of Inspection: A routine announced inspection was made to review; (1) the status of outstanding items identified in previous reports, (2) operations for the inspection period and (3) complete the requirements of PI 3000/1. Messrs. Carlson and Keppler reviewed and tested the management systems and controls that are in effect at the site. Mr. Nolan reviewed and tested the administrative system for insuring the adequacy of operating, maintenance and emergency procedures and the adequacy of the surveillance testing program. The findings of Messrs. Carlson, Keppler, and Nolan will be documented in CO Report No. 219/70-8.

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SUMMARY

Safety Items - None

Noncompliance Items -

1. Contrary to the requirements of Technical Specification 4.7.A.3, each diesel generator has not been given a thorough annual inspection. No inspection has been performed since the issuance of the provisional operating license on April 9, 1969. (Section N.6.)

2. Contrary to the requirements of Technical Specification 4.7.A.5, the diesel generator starting battery surveillance checks are not being performed in entirety. Specific required checks that are not being performed include the quarterly temperature and electrolite measurements. (Section N.3.)

3. Contrary to the requirements of Technical Specification 4.7.B.3, the 125 volt station battery surveillance checks were not being performed in entirety. Specific required checks that are not being performed include the quarterly check on electrolite level and temperature reading of every fifth cell. (Section N.5.)

4. Contrary to the requirements of Technical Specification 3.1, Table 3.1.1, item H.2, the trip setting for the "high flow" instrument in the condensate line (input into the isolation condenser isolation circuitry) was set in excess of the ≤ 27 inches ΔP H_2O required. (Section F.3.)

5. Contrary to the requirements of Technical Specification 3.5.A.6, the O_2 level within containment exceeded the specified 5% limit during operation on June 4, 1970, for a period of at least 20 hours. (Section K.1.)

6. Contrary to the requirements of Technical Specifications 4.5.K and 4.5.L, the standby gas treatment charcoal and particulate filters were not tested for removal efficiency within the required six-month interval. (Section K.6.)

Unusual Occurrences -

1. Unidentified Leakage Into Containment - The measured unidentified leakage rate into containment increased from 1 to 4.5 gpm over a period of approximately two weeks. The reactor was shutdown on September 16, 1970, to investigate the source. A recirculation pump discharge valve packing was found to be leaking and was repaired. Measured unidentified leakage returned to approximately 1 gpm following the repairs. (Section K.2.)

2. Diesel Generator Failure to Start - Surveillance testing during the period of February through October, 1970, disclosed that the No. 1 diesel generator failed to start automatically on the first attempt on four occasions. The problem was reported by the licensee to be related to poor alignment between the diesel cranking motor pinion gear and its associated ring gear. The mounting brackets for the two starting motors (that act together to crank the diesel) have been relocated on both the No. 1 and the No. 2 diesel generators. Electrical problems have been experienced which have prevented proper operation on three occasions during this period. (Section N.2.)

3. Loss of Main Circulating Water - On July 11, 1970, and August 2, 1970, sea grass plugged the main circulating water intake screens to the extent that the main circulating water pumps had to be shut down. On both occasions the reactor was operating and following the first occurrence, the reactor scrammed for high reactor pressure. Following the second occurrence, the reactor was manually scrammed. (Sections C and H.)

Status of Previously Reported Problems - A formal enforcement letter* was sent to Mr. R. F. Bovier, President, Jersey Central Power & Light Company listing seven items of noncompliance identified in previous CO reports** and other concerns regarding administrative systems and staffing for the operation of the OC-1 facility. A reply to the letter was received September 29, 1970.***

Other Significant Items -

1. Control Rod Performance - Following the April - May, 1970 rod work outage, all control rod drives were scrammed in the hot pressurized condition. The maximum time observed for any drive for 90% insertion was 3.4 seconds. During the inspection period there have been three scrams for which the scram times for 26 monitor control rods had been recorded and the maximum time observed for any drive for 90% insertion was 3.1 seconds. Totalized stall flows have been taken monthly and have increased from 167 gallons per minute to 218 gallons per minute during the inspection period. No operating difficulties had been experienced with the control rod drives. Followup inspection items previously identified by Mr. D. Pomeroy, TSB, CO:HQ, during his May, 1970 assist inspection of the control rod drives were reviewed. (Section F.1.)

2. Control Rod Inadvertent Drop - During operation on September 26, 1970, a control rod inadvertently dropped into the reactor from position 32. The cause for the inadvertent drop was determined to be an improper valving arrangement which occurred during maintenance on the control rod drive on the previous day. The improper valving resulted in removing the air supply to the scram valves which then allowed the air to slowly bleed from the scram valve diaphragms and to eventually open the scram valves. Rod recovery was made within approximately two hours. (Section F.2.)

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*Letter to Mr. R. F. Bovier, President, JC from Mr. L.D. Low, Director, CO:HQ, dated September 9, 1970.

**CO Report Nos. 219/70-1, 219/70-2 and 219/70-5 Noncompliance Items.

***Letter to Mr. L. D. Low, Director, CO:HQ from Mr. R. F. Bovier, President, JC, dated September 24, 1970.

3. Isolation Condenser Initiating Logic Circuitry - Circuitry changes were made during August, 1970 to prevent the closure of a single excess flow check valve from removing the automatic actuation of the isolation condensers on a high reactor pressure. As previously reported in CO Report No. 219/70-5, the closure of an excess flow check valve resulted in a circuit review by JC and the review disclosed the isolation condenser automatic initiating logic circuitry had been defeated. The cause for the loss of function was stated by the licensee to be due to a design error in the initiating logic circuitry. (Section E.4.)

4. Temporary Strainers in Condensate and Feedwater Systems - JC has stated that all temporary strainers have been removed from all of the nuclear systems at the OC-1 facility. The inspector's inquiry was prompted by a recent failure of a temporary strainer in the feedwater system at the Nine Mile Point reactor. (Section E.5.)

5. Turbine Initial Pressure Regulator - As discussed in CO Report No. 219/70-6, during the period of September 17-28, 1970, five steam pressure (flow) disturbances resulted from malfunctions with the initial pressure regulator. Performance of the system since September 28, 1970, has been satisfactory and no additional disturbances from this source have resulted to date. (Section H.2.)

6. Secondary Containment Testing - A review was made of the method of testing the secondary containment and the records of the results. It was disclosed that the test is conducted with both airlock doors at any one penetration closed. The results of the tests indicate the specified leak tightness for secondary containment is within the requirements of the Technical Specifications. (Section K.5.)

7. Facility Staffing - There has been a noticeable improvement since the May, 1970, inspection in the numbers of people at the site who are currently preparing for either the senior reactor operator license or a reactor operator license examination. (Section B.2.)

8. PI 3000/I "Survey of Security Measures for Emergency Power Systems" - The requirements of PI 3000/1 were completed. Station security aspects provide reasonable assurance that an authorized person cannot gain access to the emergency power controls without detection. The review did disclose that some key components within the DC emergency power system could be defeated without control room annunciation. (Section N.1.)

9. Gaseous Release Rate - The current gaseous release rate from the facility was reported to be approximately 7000 uCi/second. (Section Q.3.)

10. Carbon-14 - Per the memo from CO:HQ on this subject,* the substance of the Public Health Service's findings at the Yankee reactor were discussed with the licensee. JC has stated that samples will be obtained to assess the magnitude of Carbon-14 in their effluents. (Section Q.1.)

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*Memorandum, O'Reilly to Senior Reactor Inspectors, dated October 17, 1970.

11. Chemistry - A review of the chemistry records for the period from May through September, 1970, indicated that no Technical Specification limits had been exceeded. Typical values and the ranges of the values for measured variables were obtained. (Section E.2.)

12. Facility Plans - An outage is planned for the week of October 18-25, 1970. Major maintenance items for the outage include:

    a. Containment integrated leak rate testing.

    b. Main steam isolation valve testing.

    c. Turbine initial pressure regulator control system modifications including the replacement of control valve cams, and the replacement of control linkages.

Exit Interview - Messrs. Carlson, Keppler and McDermott conducted the exit interview with Messrs. McCluskey, Ross, Carroll, and Riggle on October 16, 1970. Messrs. Carlson and Keppler discussed their inspection findings and their discussions will be included in a separate report. Mr. McDermott discussed items of apparent noncompliance and other areas of concern as follows:

1. Diesel Generator Annual Inspection (Item of Noncompliance)

   The inspector stated that the annual inspection test of the diesel generators had not been performed since the issuance of the provisional operating license on April 9, 1969. Mr. Ross stated that it was JC's intention to perform this test and that Mr. Riggle had prepared a maintenance procedure that was awaiting the review and approval of PORC. He further stated that the inspections required by the Technical Specifications would be completed in November, 1970. This was identified as an item of noncompliance.

2. Diesel Generator Starting Battery Surveillance Checks (Item of Noncompliance)

   The inspector stated that the surveillance checks required by Technical Specification 4.7.A.5 were not being done in entirety. Specific required checks that were not being performed were stated to include the quarterly temperature measurements and the electrolyte measurements. Mr. Ross stated that it was JC's intention to perform the temperature and electrolyte measurement tests.

3. Station Battery Surveillance Checks (Item of Noncompliance)

   The inspector stated that the surveillance tests required by Technical Specification 4.7.B.3. on the 125 volt station batteries were not being performed in accordance with the Technical Specifications. Specific required checks that were not being performed were identified to include the quarterly check on electrolyte level and the temperature reading of the fifth cell. Mr. Ross stated that it was JC's intention to perform these tests. This was identified as an item of noncompliance.

4. Instrument Trip Setting Isolation Condenser Isolation Circuitry (Item of Noncompliance)

The inspector stated that the trip setting for the high flow instrument in the condensate line had exceeded the Technical Specification limit between the period of December 9, 1969 and July 1, 1970. This was identified as an item of noncompliance.

5. Containment Inerting (Item of Noncompliance)

The inspector stated that a review of the records had disclosed that the O_2 content within containment had exceeded the Technical Specification (3.5.A.6.) limit of 5% during operation on June 4, 1970. The inspector further stated that there was a strong indication that the 5% limit may have been exceeded for the six previous days as proper operation of the O_2 sampling instrument was suspect during this period. Mr. Ross was requested to provide the basis for the PORC committee's review of this occurrence and their findings that this event did not violate the Technical Specification limit. Mr. Ross' justification for this finding was that the Technical Specifications allow a period of 24 hours following a startup and 24 hours before scheduled shutdowns during which periods the specified 5% limit for O_2 is invalidated and that based on this, it was JC's understanding or interpretation of the Technical Specifications that the O_2 limit could exceed the specified 5% for up to a period of 24 hours during operation. The inspector stated that he did not concur with this interpretation and identified this as an item of noncompliance.

6. Diesel Generator's Performance

The inspector stated that his review of the records had disclosed that on five occasions during the period of February through October 1970, the No. 1 diesel generator had either failed to start on the first attempt or had tripped out from an electrical fault. The inspector also stated that his review of the work requests had disclosed that on only two of the five occasions had any followup action been initiated. The inspector stated that it was apparent to him that an adequate review was not being made of surveillance testing records as all of these faults had been indicated in the surveillance test records. Mr. Ross informed the inspector that following Mr. Don Reeves' successful completion of the October, 1970 senior operating licensing examination, he will be assigned overall responsibility for the surveillance testing program and that this should result in an improvement in this area.

7. 125 Volt Station Battery Test

The inspector stated that the review of the semi-annual discharge testing of the A and B batteries had indicated that on five occasions, one or the other of the battery banks had failed to meet the minimum acceptance criteria for the test. Mr. Ross was also informed that the maintenance supervisor was only aware of the most recent failure which occurred on October 3, 1970. These test results were stated to indicate that an inadequate review of the surveillance testing records was being made. Mr. Ross again stated that Mr. Don Reeves will be assigned overall responsibility for the surveillance testing program and it is expected that improvement in the coordination and review of tests will result.

DETAILS

A. Persons Contacted:

Mr. T. McCluskey, Station Superintendent, OC-1
Mr. D. Ross, Technical Supervisor, OC-1
Mr. J. Carroll, Operations Supervisor, OC-1
Mr. W. Riggle, Maintenance Supervisor, OC-1
Mr. J. Sullivan, Technical Engineer, OC-1
Mr. R. Toole, Technical Engineer, OC-1
Mr. T. Johnson, Electrical Foreman, OC-1
Mr. F. Kossatz, Mechanical Foreman, OC-1
Mr. D. Kaulback, Radiation Protection Supervisor, OC-1
Mr. N. Goodenough, QA Engineer (Radiography), GPU

B. Administration and Organization

1. Management - Administrative Controls

Mr. R. T. Carlson, Senior Reactor Inspector, CO:I and Mr. J. Keppler, Senior Reactor Inspection Specialist, CO:HQ were at the site during the period of October 13-16, 1970, to review and test the management systems and controls at the site. Mr. F. Nolan, Senior Reactor Inspection Specialist, CO:HQ was at the site during the period of October 13-15, 1970, to review and test the system for insuring the adequacy of operating, maintenance, and emergency procedures and to review the controls used in implementing and reviewing the results of surveillance testing. Their inspection findings will be discussed in a separate report.

2. Operations Organization

Mr. J. Carroll informed the inspector that the present operating organization consists of the following:

a. Shift Foremen - These positions are currently being filled with four senior licensed operators. In addition, there are two unlicensed operating foremen, one of which is scheduled for an October, 1970, senior operating license exam.

b. Shift Operators - These positions are currently being filled by three licensed operators and one senior licensed operator. Of the four control room "B" operators, three have operating licenses. Five individuals are in training for operating licenses and scheduled to take the exams in October of 1970.

c. Mr. Don Reeves is also in training for the senior operating license exams scheduled for October, 1970. It was reported that following successful completion of the exams, Mr. Reeves will assume the overall responsibility for the surveillance testing program.

3. Site Technical Support Staffing

Mr. D. Ross informed the inspector that the present staffing in the Radiation Protection Group and the Chemical Group is as shown in Figure 1 attached. Mr. Ross reported that Mr. Pelrein, Chemistry Supervisor has had 10 years of radiochemistry experience at KAPL and 8-10 years experience in radiochemistry and chemistry at Industrial Research Laboratories reactor. Mr. Kaulback, Radiation Protection Supervisor, has had prior health physics experience at the Saxton reactor and was present at the OC-1 facility prior to reactor startup.

Mr. Ross was asked to provide the inspector with the summary sheets or other reports that he requires from the Radiation Protection and Chemistry Groups. Mr. Ross provided the inspector with copies of summary sheets that he maintains in his office that are originated within the health physics and chemistry groups. The records consist of the following:

a. Monthly reports which include a summary of the solid wastes discharged from the facility, personnel exposures, building surveys, airborne radioactivity surveys, batch radioactive liquid dumps, identification and amount of activity of liquid waste discharged, total volume of liquid waste discharged.

b. Weekly summary sheets of gaseous releases from the facility including an isotopic breakdown of the gas.

c. Weekly summaries of primary system chemistry.

Records appeared to be current and in sufficient detail to permit auditing of the health physics and chemistry groups' activities.

C. Operations

Following the April - May, 1970 rod work outage, reactor operation was resumed on May 22, 1970, and continued until June 17, 1970, when a scram was initiated from 100% power from closure of the main steam isolation valves (MSIV's). The valves closed on a signal from two faulty bi-metal switches that are located above the turbine control valves. These switches are used to monitor for a limited steam break. All post-scram functions following this scram were reported by the station superintendent to have functioned normally. Reactor operations resumed on June 17, 1970, and continued until July 11, 1970, when a scram was initiated by a partial loss of main circulating water for the condensers which resulted from sea grass accumulation on the intake screens. The reactor was restarted on July 12, 1970, and continued until August 1, 1970, when sea grass accumulation on the intake screens again resulted in a reactor scram (manual). Operations resumed on August 2, 1970 and continued until September 16, 1970, when a manual shutdown was initiated as required by Technical Specifications for a high identified leakage rate in the containment which was caused by a leaking valve packing. Operations resumed on September 17, 1970. During the period of September 17 through September 28, 1970, three additional scrams resulted from malfunctions of the turbine initial steam pressure regulator (IPR). On September 28, 1970, an additional plant disturbance was experienced as a result of a malfunction of the feedwater controller system but the plant did not scram. The plant was shutdown on October 17, 1970, for a planned eight-day outage. Major activities that were scheduled for the outage included: (1) a containment integrated leakage test; (2) a main steam isolation valve leakage test; (3) modifications to the turbine control system including the replacement of linkages and the control valve cams. Listed below is a description of the unscheduled shutdowns during the inspection period:

Date	Cause
June 17, 1970 (Scram No. 45)	Automatic scram from 100% power resulting from the closure of the main steam isolation valves. Just prior to the scram there had been two instances of spurious half-trips on the main steam line break circuit, which appeared to be caused by high readings from the temperature sensors. Checking of the RTD's located in the area above the turbine control valves indicated an ambient temperature of approximately 130° F which did not indicate that there should have been any spurious trips. Before all circuits could be checked out, two temperature detectors, one in each channel, picked up and caused the main steam isolation valves to close which resulted in a reactor scram. Reactor pressure reached 1040 psig during the ensuing transient and the isolation condensers were manually initiated to cool down and depressurize the reactor. All monitored control rods reached 90% insertion within 3.11 seconds. Following this scram and turbine trip, the 4160 volt power supply to MCC 1-A was automatically switched to the startup transformer. However, MCC 1-B did not transfer automatically and resulted in the starting of diesel generator No. 2. Before the diesel generator picked up the loads on MCC 1-D, the operator had synchronized and manually closed breaker S1D restoring power to MCC 1-B, (which feeds MCC 1-D) and the diesel generator was shut down. Thus, all safety equipment, which might have been required, would have operated if needed. A check of breaker S1D after the scram showed it to be operating properly and all interlocks functioning normal. Further investigation disclosed

Date	Cause
	that the malfunctioning may have been attributed to dirty contacts in the breaker. The check of the bi-metal temperature switches disclosed that two of the detectors had set points of $\approx 50^\circ$ F less than 180° which would be the allowable set point for the bi-metal switches in the vicinity of the turbine control valves (Technical Specifications allows a 50° span for the trip point above the ambient measured temperature). The two temperature detectors were reset to 180° F.
July 11, 1970 (Scram No. 46)	Automatic scram from approximately 45% power which was initiated by high reactor pressure. Just prior to the scram the load was being reduced due to a buildup of grass at the intake structure which caused the shearing of pins in the travelling screens and compounded the situation. Two (of four) circulating water pumps were removed from service due to the cavitation which was caused by loss of suction pressure. The water level had decreased at the intake structures and started effecting the service cooling water systems and when the reactor recirculating pump and turbine oil temperatures started to increase, the load was dropped to approximately 200 MWe with recirculation flow and the turbine generator was tripped with the emergency trip buttons. About one minute after the turbine tripped, with nine bypass valves opened, the pressure increased and the reactor scrammed on high pressure which was apparently caused by insufficient condensing ability of the main condenser. The vessel water level dropped to approximately 9 feet-4 inches above the active fuel on this transient. No control rod scram times were obtained on this scram as the recorder apparently stuck. The rod buffer times were examined and appeared normal. Conditions were returned to normal at the intake structure by reducing the flow, installing new sheer pins in the travelling screens, and running the screens continuously until the grass conditions tapered off.
August 1, 1970 (manual shutdown)	During operation at 100% power, sea grass again began to plug the intake structure travelling screens. The reactor was manually shut down and the main circulating water pumps turned off. The reactor was returned to service August 2, 1970.

Date	Cause
September 17, 1970 (Scram No. 47)	Automatic scram from 100% power which was initiated by a high flux signal. Immediately preceeding this scram, the reactor was operating at 1600 MWt, when a power oscillation began. The operator dropped load with recirculation flow from 530 MWe to 500 MWe where a power oscillation again occurred. The load was dropped further to 400 MWe at which time everything leveled out. The turbine then tripped from a high level in the moisture separator drain tank. The turbine trip caused the pressure to increase to 1010 psig which in turn caused the high flux scram from void collapse. Reactor water level dropped to approximately 9 feet-4 inches above the active fuel. Main steam line low pressure (850 psi) occurred approximately 17 seconds after the scram, followed immediately by a main steam isolation valve closure. All control rods reached 90% insertion within 3.06 seconds for the 26 monitored rods. Buffer actions appeared normal on all monitored rods. Just prior to the scram, the condensers were being backwashed, which caused the load to decrease, then increase as each condenser half's flow was reversed. It was thought by JC that these power swings might have contributed to the start of the oscillations. Prior to resuming operation, the operators were instructed to decrease generator load to a more stable cam position before backwashing condensers.
September 22, 1970 (Scram No. 48)	Automatic scram from approximately 95% power, which was initiated by closure of the main steam isolation valves. The valves were closed when the main steam line pressure reached 850 psig. Reactor water level dropped to 8 feet 7 inches above the active fuel and there was no data available from the 26 monitor control rods as the recorder switch was in the off position at the time of the scram. Prior to the scram, the reactor was operating at approximately 1520 MWt. The first indication of the problem was a very small swing up in electrical load, accompanied by a decrease in reactor pressure. The operator immediately started to reduce load, but pressure continued to drop. The pressure drop resulted from a malfunction of the electric pressure regulator (EPR) portion of the turbine steam inital pressure regulator (IPR). The specific component

Date	Cause
	that malfunctioned was the MOOG valve (hydraulic pilot valve), which was found to be plugged with foreign material. This is discussed more fully in CO Report No. 219/70-6.
October 2, 1970 (Scram No. 49)	Automatic scram from approximately 50% power which was initiated by a turbine trip. Immediately preceeding the scram the load was being reduced to check the north side of the "C" condenser for salt water (tube) leaks. Recirculation flow was at the rated 1.6×10^5 gpm. The load had been reduced to 290 MWe by control rod insertions when the turbine trip resulted from high level in the moisture separator drain tanks. This initiated a reactor scram from high pressure. The cause for the unplanned reactor trip was attributed to a false high level from the moisture separator drain tanks. Control rod scram times of the 26 monitored rods ranged from 2.46 to 2.92 for 90% insertion. It was reported that the isolation condensers were not automatically initiated on this scram as the time-pressure conditions were not satisfied to initiate automatic initiation, i.e., reactor pressure did not remain above 1040 psig for 15 seconds.

D. Facility Procedures

Mr. F. Nolan, CO:HQ, was at the site during the period of October 13-15, 1970 to review:

1. Assigned responsibilities for initiating required procedures or test documents.

2. Review and approval methods for procedures.

3. Periodic updating controls.

4. Controls to assure procedure or test modification following equipment modification.

5. Controls to insure effective communications of procedure or test changes to operating personnel, including related retraining.

6. Responsibilities and assignments of individuals reviewing and approving surveillance test results.

7. The adequacy of the surveillance test records to permit evaluating of the results of the tests.

Mr. Nolan's findings will be included in a separate report.

E. Primary System

1. Unidentified Leakage in the Containment

The unidentified leakage in the containment increased from approximately 1 gpm to 4.5 gpm during a period of about two weeks operation. The reactor was shut down on September 16, 1970 to investigate the source of the leakage. Investigation disclosed that the source of the leakage was a packing on the "E" recirculation pump discharge valve. Following repairs to the packing, the measured indicated unidentified leakage rate in the containment returned to approximately 0.8 gpm.

This packing leak had been suspected as the unidentified leakage into containment had started to increase after cycling the discharge valve. The "E" recirculation pump had been removed from service and isolated (suction and discharge valves closed) to work on the recirculating pump MG set brushes on September 1, 1970.

The method used to calculate the unidentified leakage rate in the containment was reviewed with Mr. J. Carroll and he informed the inspector that the integrated flow readings of the containment sump pump are obtained each hour and that the change in integrated flow for a 24-hour period is converted into a gallon per minute leak rate. The leak rate is plotted daily by both the operations supervisor and technical engineers. Discussions were held with Mr. J. Sullivan and Mr. J. Carroll to determine if other sources are available to independently identify or verify leakage in the containment. Both indicated that at present there is no other reliable instrument to measure unidentified leakage in the containment although containment temperature and humidity are plotted daily. The inspector reviewed the temperature and humidity data and could not correlate variations in these measured parameters with the recent increase in unidentified leakage in the containment.

Subsequent to this September 16, 1970 shutdown, JC implemented a program to sample the containment atmosphere for radioactivity on a weekly frequency. They intend to use the results of the sampling program to determine if this method could provide a sensitive leak detection method. Mr. Carroll also informed the inspector that the instrument used for determining the relative humidity in containment will be relocated to a new position (adjacent to the temperature sensor) during the planned October, 1970 outage. The stated purpose for this relocation was that it is expected to provide more meaningful data for determining unidentified leakage in the containment.

2. Chemistry

Primary coolant chemistry records were reviewed for the period from May 28, 1970 to October 12, 1970. Typical values, as well as the recorded range of values, are tabulated below:

	Range	Typical Value
pH	5.8 - 7.3	6.1
Chlorides	$<$ 20 - 140 ppb	$<$ 20 ppb
SiO_2	40 - 900 ppb	70 ppb
H_3	4×10^{-4} to 7×10^{-4} mCi/ml	5×10^{-4} mCi/ml
Gross $\beta \cdot \gamma$ activity	5.8×10^{-4} - 1.5×10^{-2} mCi/ml	
Iodine	1×10^{-2} - 1.6×10^{-1} uCi/ml	
Conductivity	0.28 - 9.6 umho/cm	

A review of the records reflected that the main coolant chemistry has remained within the Technical Specification limits for the inspection period. OC-1 has experienced some salt leaks in the main condensers which have been corrected by plugging of tubes.

3. North Core Spray Nozzle Wall Thickness Determinations

As was previously reported,* linear defects were observed during a LP check of the OC-1 north core spray nozzle safe end overlay cladding. This LP check had been made during the April - May, 1970 outage as a result of the Nine Mile Point nozzle cracking problem. The investigation of the defects (boat sample) disclosed that the material was inconel and not 308-L as stated in the application. The defects (microfissuring) were determined to be the result of weld solidification during the application of the overlay cladding. The licensee removed all the defects by grinding during the April - May outage and measured the remaining safe end wall thickness by radiographic techniques. CO:I (Tillou and McDermott) reviewed the techniques used for measuring the wall thickness and advised JC (following the May 18-22, 1970 inspection) that it would be prudent to perform additional wall thickness measurements in the light of the indicated small margin over minimum code requirements. Mr. McCluskey stated that additional checks would be considered.

During this inspection, Mr. McCluskey was asked if additional wall thickness determinations were planned. He informed the inspector that

*CO Report No. 219/70-5, Other Significant Items No. 8 and Section E.2.

Mr. N. Goodenough, QA Engineer (Radiography), GPU was scheduled to perform additional wall thickness determinations during the planned October 1970 outage. The results of these measurements will be reviewed during the next inspection.

4. Isolation Condenser Initiating Logic Circuitry

A licensee review of schematics was prompted by a separation of a fitting on a primary system instrument sensing line. This separation resulted in a closure of an excess flow check valve, which in turn resulted in a loss of sensed pressure.* The JC review disclosed that the high steam pressure (1060 psig for 20 seconds) initiating logic for the isolation condensers was defeated by the closure of the excess flow check valve in the sensing line. As this was not the design intent, GE was requested to provide the required design change and testing procedure. Both the design change and the testing procedure were supplied by GE and reviewed and approved for installation by PORC. The design change was completed in August of 1970.

The corrective action consisted of interchanging the sensors (high pressure) that operate relays 6K10 and 6K11. (See attached Figures 2 and 3). The present automatic initiating logic for the isolation condensers on sensed high reactor pressure will not be defeated by the closure of a single excess flow check valve (which would remove the sensed pressure from two high pressure sensors). Additional fuses were added to the power supply to relays 6K9, 6K10, 6K11 and 6K12 to preclude the failure of a single fuse from negating the initiating logic circuitry. The change details are shown on Figures 2 and 3 attached.

5. Temporary Strainers

Based on the Nine Mile Point experience** of a failed strainer in the feedwater system, the inspector asked Messrs. Carroll and Riggle if there were any temporary strainers in any of the systems at OC-1. They informed the inspector that temporary strainers had been installed during the construction phase, but that all temporary strainers had been removed from the systems by February, 1970.

F. Reactivity Control and Core Physics

1. Control Rods

a. Scram Times

Station records were reviewed for the period which began on May 22, 1970, at the completion of the rod inspection outage. All control rod drives were removed, repaired and reinstalled during a five-week

*CO Report No. 219/70-5, Section F.2.g.

**Inquiry Memorandum No. 220/70-C, Niagara Mohawk Power Corporation "Failure of Temporary Strainer in Feedwater System" outage in April - May, 1970. Following reassembly, cold depressurized scram times were obtained, as were hot pressurized scram times which were taken on May 22, 1970. Since that time, five scrams have resulted and meaningful scram time data has been obtained for three of the scrams. Listed below are the results of the measurements:

	Date	Range for 90% Insertion
Hot pressurized scram test (all rods)	May 22, 1970	2.42 - 3.4 seconds (average time 2.77)
Scram No. 45 (26 rods)	June 17, 1970	2.42 - 3.11 seconds (average time 2.82)
Scram No. 46	July 11, 1970	No scram time data obtained as the recorder apparently stuck
Scram No. 47 (26 rods)	Sept. 17, 1970	2.53 - 3.06 seconds
Scram No. 48	Sept. 22, 1970	No scram time data collected as the recorder switch was found in the off position
Scram No. 49 (26 rods)	Oct. 2, 1970	2.46 - 2.92 seconds (average time 2.72)

Buffer times appeared normal for all scrams that were recorded on the scram time nonitor (brush recorder).

b. Stall Flows

Station records were reviewed to monitor the performance of the seals as indicated by stall flow measurements which are taken monthly. Tabulated below are the results of the review:

	No. of individual rods with stall flows in indicated range:			Totalized stall flow measurements
	≥ 3 gpm	≥4 gpm*	≥5 gpm*	
May	4	1	1	167 gpm
June	8	4	1	191 gpm
July	8	4	0	199 gpm
August	9	0	0	189 gpm
September	7	2	0	212 gpm
October	14	0	0	218 gpm

*High individual stall flows were corrected by reworking the directional solenoid operated control valves that control normal movement of the rod.

c. Followup Inspection Items*

Mr. D. Pomeroy, TSB, CO:HQ identified four items for followup inspection during his assist inspection at the facility on May 20 and 21, 1970. These included: (1) The reassembly reports that had been contaminated and were unavailable for review during this inspection, (2) The results of repeat friction test for four drives whose original test indicated marginal conformance, (3) The results of pressurized scram and stall flow tests, and (4) The results of continuing surveillance, i.e., scram times, buffer action, monthly stall flow tests.

Items (3) and (4) are discussed in paragraphs a. and b. above. The inspector was informed during this inspection that no reassembly reports were available for review. Mr. Pomeroy was previously informed by station personnel that these forms were contaminated and were in the process of being copied. During this inspection, discussions were held with Messrs. Carroll and Goodenough and they informed the inspector that these forms had not been used, but that the GPU QC engineers had observed the reassembly of all drives. Mr. Goodenough informed the inspector that the GPU QC engineers had rejected 15-20 drives (primarily for bulged index tubes) during the inspection of the drives that otherwise would have been re-installed into the reactor by GE.

During Mr. Pomeroy's May 1970 assist inspection he identified four rods that did not appear to meet the specified acceptance criteria of a 15 psi deviation for continuous rod withdrawal. JC stated at that time that individual "notch out and settle" tests would be performed for these rods. The stated acceptance criteria for the "notch out and settle" tests was a minimum of 30 psi settling pressure. During the most recent inspection the records for these tests were reviewed and in all cases the rods met the acceptance criteria.

2. Inadvertent Rod Drop

During power operation on September 26, 1970 a control rod inadvertently dropped into the reactor from notch 32. On September 25, 1970 (the day before) this rod had been valved out of service to remove and repair the scram accumulator. The accumulator had been reinstalled and it was thought that all valving was returned to normal for the drive. Subsequent investigation of the cause of the rod drop disclosed that the scram inlet and discharge valves had opened and that this was caused by a manual air supply valve being left closed which allowed the air pressure to slowly bleed off the diaphragms of the scram valves. The air pressure on the diaphragms eventually reduced to the point where the spring loading on the scram valves opened the valves and scrammed the rod into the reactor. Following the finding of the valving error, the rod was withdrawn from the reactor to its normal position within two hours.

*CO Report No. 219/70-5, Addendum 4.

The procedure for isolating the charging scram accumulator calls for closing the air supply valve in question and also the check list calls for the valve to be reopened after the work is completed. The check list used was reported to have been signed off by the operator but it is believed by station management that an error was made on the operator's part and that the valve was not reopened.

3. Instrumentation Setting for Isolation Condenser Isolation (Item of Noncompliance)*

Mr. T. McCluskey informed the inspector during a telecon on the morning of July 1, 1970, that the present trip settings (at that time) for the steam line and condensate line break instrumentation (used to isolate the condensers in the event of a line break) were 20 psig Δ P and 59 inches Δ P H_2O respectively. The Technical Specifications Paragraph 3.1. item H.2. requires that the condensate line instrumentation be set at $\leq$ 27 inches Δ P H_2O. At that time Mr. McCluskey was informed that the current trip setting of the condensate line instrumentation was in violation of the Technical Specifications for a limiting condition for operation and that the plant was operating in noncompliance with the Technical Specifications. Mr. McCluskey was subsequently contacted at 2:30 p.m. on July 1, 1970, and requested to immediately contact DRL to discuss this situation. Mr. McCluskey informed the inspector at 6:30 p.m. that discussions were held with DRL and that JC had decided to reduce the trip point to the specified value and to report in writing to DRL when this had been accomplished. JC made the required change and did report on July 2, 1970.**

During this inspection it was ascertained that the original change in instrumentation setting from 27 inches to 59 inches Δ P H_2O was made on December 9, 1969, as a result of GE instructions to the JC maintenance group. GE had requested that the instrument be set at 68 inches ΔP H_2O but during an attempt to set the instrument at 68 inches, it was found that the total range of the instrument was limited to 60 inches. It was therefore decided to establish the trip point at 59 inches. The inspector was informed that the change was prompted by an inadvertent isolation of the isolation condensers when they had been tested under full load conditions for the first time. At the time of the inadvertent isolation, the condensate line break instrumentation was set at 27 inches Δ P. At that time it was decided to incorporate the four-second time delay (initiates isolation of the condensers four seconds after the flow trip) and also to increase the trip set point from 27 inches to 68 inches Δ P H_2O.

During the July 1, 1970, telecon between Messrs. McCluskey and McDermott, Mr. McCluskey informed the inspector that he considered the Technical Specifications to be in error as GE had verbally provided calibration data for the condensate line flow instrument. The calibration indicated

*Inquiry Memorandum No. 219/70-H.

**TWX to Dr. P. A. Morris, Director, DRL from Mr. I. Finfrock, Manager, Nuclear Generating Stations, JC, dated July 2, 1970.

that a setting of 68 inches would be permissible and that this setting would satisfy the conditions specified in the basis of the Technical Specifications of isolating the isolation condensers at less than or equal to 300% of rated flow through the condensate line. Mr. McCluskey was informed at that time that information available to Compliance, Region I (Mr. D. Pomeroy's calculations) were not in agreement with the numbers provided by GE and that based on the past history of GE supplying various numbers for the trip set points, JC was encouraged by the inspector to make an independent review.

During the most recent inspection, followup correspondence between JC and GE on July 31, 1970, on this matter, was reviewed by the inspector. GE stated in this correspondence that the error in the specified set point provided by GE to JC on July 2, 1970 (letter), was due to errors made in the assumptions on the elbow radius and a calculational error involving a misplaced density term. GE, at this time, provided a new value of 300% of design flow, which was 25 inches ΔP H_2O and 19.4 psig for the steam. JC requested the GPU technical support group to recheck the figures. GPU has reviewed the latest settings supplied by GE and their calculations agree with GE supplied numbers.

Since the instrument setting change on July 2, 1970 (back to 27 inches ΔP H_2O), the isolation condensers have been manually placed in service, without any additional spurious isolations of the condensers. No automatic initiation of the isolation condensers (from high pressure) have resulted but the manual initiation should simulate to a large degree the system response. It appears that, based on the experience with several manual initiations of the isolation condensers, the reduced setting, i.e., 27 inches ΔP H_2O on the condensate break instrumentation will not result in additional spurious isolations of the system. This issue was discussed during the exit interview as an item of noncompliance.

H. Power Conversion System

1. Loss of Main Circulating Water

On two occasions (July 11 and August 1, 1970), sea grass plugged the traveling intake screens for the main circulating water suction wells, and resulted in two reactor scrams.* The heavy accumulation of sea grass was reported by the licensee to be a seasonal condition which results from the grass breaking off the bottom of Barnegat Bay in heavy quantities during the summer months. During these periods, the screens are run in a "continuous advance" mode and station personnel man the heavy log screens (upstream of the traveling screens) to assist in the removal of the sea grass. On one occasion heavy sea grass accumulation on the traveling screens resulted in the shearing of pins which then did not allow the screens to advance and eventually resulted in a partial loss of the main circulating pumps suction supply. The water which is passed through the screens also supplies, in addition to the main

*Section C of this report.

circulating pumps, the two service water pumps and the four emergency service water pumps. These pumps would eventually lose suction if the main circulating pumps were allowed to continue to operate when the screens are plugged. Mr. McCluskey informed the inspector that there are two separate suction bays (with separate intake screens) and the two bays are physically isolated from each other so that a loss of a single bay does not effect the other bay. One bay supplies two main circulating pumps, one service water pump and two emergency service water pumps.

2. Initial Pressure Regulator Performance

Five recent disturbances to the steam pressure (flow) at the OC-1 facility resulted during the period of Sept. 17-28, 1970. These disturbances have resulted in part from poor design of the cams which operate the main turbine inlet control valves and in part, from dirt within the control oil system. The disturbances had been manifested by power oscillations ranging up to $\pm$ 25 MWe and in power spiking which has ranged up to 55 MWe. These events are discussed fully in CO Report No. 219/70-6.

K. Containment

1. Containment Inerting (Item of Noncompliance)

Station records were reviewed during the inspection. It was observed on June 4, 1970, that the O_2 level within containment exceeded the specified 5% limit.* The time period for which the O_2 concentration was logged to be in excess of the 5% specified was $\approx$20 hours. Records disclosed a step change in O_2 from $\approx$0% to 7.5% occurred at $\approx$1300 on June 4, 1970 and the O_2 level remained $>$5% for the stated 20-hr period. O_2 levels of $\approx$0% were recorded for a period of six days from May 28 through June 4, 1970. Mr. Carroll stated that the instrument had not been performing properly and that during this period, O_2 samples had been taken with a portable instrument and found to be within limits. The inspector requested Mr. Carroll to provide him with records of the sample results which were taken with a portable sampler. Mr. Carroll informed the inspector after reviewing the shift supervisor and operations log and finding no entries of O_2 sampling, that apparently no O_2 samples had been taken until after June 4, 1970.

PORC committee minutes were reviewed and indicated that the committee did review this item on June 9, 1970. The minutes did not reflect that any recommendations for followup or that a Technical Specification limit had been exceeded.

This subject was discussed during the exit interview. Mr. D. Ross was questioned to provide the basis for the statement in the PORC minutes that no Technical Specification limit had been exceeded. He

*Technical Specification paragraph 3.5.A.6.

informed the inspector that JC had interpreted the Technical Specifications to allow for O_2 to be in excess of the specified 5% limit for periods of up to 24 hours. In that the logged values of O_2 in containment did not indicate that the containment atmosphere was in excess of 5% O_2 for a period of 24 hours, they therefore concluded that the specifications had not been exceeded. The inspector requested Mr. Ross to provide the basis of how the measured O_2 in containment took a step change (between hourly readings) from 0% to 7.5% on June 4, 1970. (This is the period when this is the start of the 20-hour period that the O_2 was in excess of the 5% limit). Mr. Ross could provide no such basis to the inspector. The inspector informed Mr. Ross that CO:I would review this matter further but that the tentative finding was that the plant had operated in noncompliance with the Technical Specifications.

2. Unidentified Leakage in Containment

On September 16, 1970, the plant was shutdown when the unidentified leakage rate into the containment reached 4.5 gpm. The leakage had increased during the period of September 1-15, 1970, during which time leakage increased from 1 gpm to 4.5 gpm. Following the shutdown it was found that the packing was leaking on the E recirculation pump discharge valve. Repairs were made to the packing and the reactor was returned to service on September 17, 1970. The unidentified leakage returned to 0.8 gpm following the repairs. Discussions were held with Mr. Carroll and he informed the inspector that the unidentified leakage is determined by flow integrator readings that are taken hourly on the containment sump pump. The unidentified leakage rate into containment is determined over a 24-hour period by using the flow integrator readings taken at midnight each night and calculating the average in-leakage. Mr. Carroll informed the inspector that the other indicators that are also measured in containment (relative humidity, temperature, and pressure) are not presently able to provide a more sensitive or equally sensitive means for determining in-leakage. Humidity, temperature, and pressure records were reviewed by the inspector and it was noted that a poor correlation of these parameters could be made with the increase in in-leakage that occurred from September 1 to September 15, 1970. Mr. Carroll also informed the inspector that following the shutdown on September 16, 1970, a weekly grab sampling program was implemented to obtain base line data for future investigation into the sensitivity of using containment activity as a leak detection method.

3. Containment Integrated Leak Rate Testing

Mr. Mc Cluskey informed the inspector that Chicago Bridge & Iron Company (CB&I) has been contracted to perform a containment integrated leak rate test. The test is currently scheduled for October, 1970. The results of this test will be documented in the next report.

4. Main Steam Isolation Valves (MSIV's)

Mr. McCluskey informed the inspector that the main steam isolation valves will be tested for leak tightness during the planned October, 1970 outage. The results of this inspection will be documented in the next report. It was also made known that GE has supplied OC-1 with eight pneumatic valves to replace those presently installed at OC-1. These valves are the pilot valves that control the closure of the MSIV's. Dresden 2 has had poor experience with their pneumatic valves. Mr. McCluskey was unsure as to whether or not the pneumatic valves would be changed out and stated that JC was currently questioning GE to obtain the basis for the changeout as OC-1 has not experienced difficulties with these valves.

5. Secondary Containment Testing

A review was made by the inspector of the method of secondary containment testing. It was disclosed that the testing is performed with all double air-locked doors closed and sealed. The results of the recent testing indicated that the leakage limit specified in the Technical Specifications was being met. Mr. D. Ross was questioned by the inspector to determine if OC-1 had plans to test secondary containment with the airlock doors in various positions (one door opened and one door closed). Mr. Ross stated that JC had no such intentions of testing in this manner due to practical considerations. The inspector asked Mr. Ross if JC had considered routine monitoring of some variable within the ventilation control scheme to continuously monitor the status of secondary containment. Mr. Ross informed the inspector that JC had not considered a continuous monitoring scheme. He was responsive to this question and stated he would review the matter to ascertain if this was feasible and practical.

6. Standby Gas Treatment Filters (Item of Noncompliance)

Testing records were reviewed and it was disclosed that both the charcoal and particulate filters had not been tested within the required six-month interval (Technical Specifications 4.5.K and 4.5.L). The charcoal filters were tested on January 31, 1970 and again on August 20, 1970 exceeding the specified six-month interval. The particulate filters were tested on January 19, 1970 and again on August 18, 1970 also exceeding the specified six-month interval. Mr. McCluskey stated during the exit interview that the retest of both sets of filters has now been scheduled for intervals of less than six months to prevent a recurrence.

N. Emergency Power

1. Provisional Instruction 3000/1 "Survey of Security Measures for Emergency Power System"

a. Access Control

The physical arrangements and barriers to preclude or control access were examined and the following is a description of these barriers and controls. The entire facility is surrounded by a security fence, the gates to which are manned 24 hours per day by a security guard. Access to the reactor and turbine building is controlled by locked outside doors. The administrative security aspects of the facility include authorization for entry, escorts for personnel outside the area of the administrative offices, sign-in and sign-out requirements. The diesel generators and their fuel oil supply are housed in separate outside buildings (within the security fence) that do not have controlled access. Physical access to the diesels or the generator controls is restricted by locked cubicles within the structures housing the diesel generators. There is unlocked access (within the confines of the security fence) to the fuel oil storage tank and the fuel oil supply valve from the storage tank to the day tanks. The rooms within the reactor and turbine building that house the vital rotating and electrical switch gear equipment are controlled by locked entry. Entry into these areas is authorized for station staff personnel, the operations group personnel, and the maintenance personnel. A roving operator patrols the reactor, turbine and diesel generator buildings each shift.

b. Controls and Control Indications

A review of schematic drawings was conducted at the site and discussions were held with Mr. Riggle to determine if control room indications would be obtained to alert the operator for abnormal conditions that would result in a loss of availability of emergency power systems or equipment. Figures 3 and 4 attached are elementary one-line diagrams of the AC and DC normal and emergency power systems at the facility. These figures should be used to assist the reader in the following description.

As shown on Figure 3, the normal station power is fed to MCC's 1A and 1B from the output of the main generator during normal operation. Breakers S1A and S1B are normally open when the generator is on the line. In the event of a main generator trip or when the generator is shut down for extended periods, S1A and S1B close automatically to provide power to MCC 1A and MCC 1B from the startup transformers SA and SB. 4160 volt MCC's 1C and 1D are normally fed from MCC 1A and MCC 1B respectively. In the event of an undervoltage condition on either MCC 1C or MCC 1D, its associated diesel generator starts automatically and assumes the load of that bus. The cross-tie breaker between MCC's 1C and 1D, although normally opened, can be closed to parallel at the 4160 volt level.

The diesel generator breakers DG-1 and DG-2 are controlled by an auto-manual selector switch located inside the locked cubicle for each diesel generator. Control room annunciation is provided when the auto-manual selector switch is placed in a manual position which would result in a loss of auto-start capability. There are no other front panel controls within the locked control cubicles for the diesel generators that could defeat the diesels automatically being placed in service and assuming the load on MCC's 1C and 1D.

The six 460 volt motor control centers 1A1, 1A2, 1A3, 1B1, 1B2 and 1B3 power the majority of the operating equipment within the turbine and reactor buildings. There are provisions to cross-tie 1A1 to 1B1, 1A2 to 1B2, and 1A3 to 1B3 but interlocks are provided to not permit cross-tieing in such a manner as to parallel the 4160 volt level system through the 460 volt level i.e., one of the two associated feeder breakers for either 1A1 or 1B1 must be opened before the cross-tie breaker is permitted to close. 1A2 and 1B2 feed the vital MCC's 1A2 and 1B2. The vital MCC's in turn power critical power panels. Loss of either vital MCC-1A2 or MCC-1B2 would be accompanied by numerous control room alarms. The majority of the auto transfer switches that feed power to vital loads and that are powered from either vital MCC-1A2 or vital MCC-1B2, have "off-normal" alarms which would alert the control room operator to a loss from either of these buses. During a loss of off-site power, vital buses MCC-1A2 and MCC-1B2 are powered from the diesel generators as the feeder breakers from MCC's 1C and 1D to 1A2 and 1B2 respectively remain closed. The loss of either MG set No. 1-1 or MG set No. 1-2 (feed protection system panels No. 1 and No. 2) would be annunciated in the control room as this would result in a 1/2 scram. All 460 volt MCC's and vital switch gear are located in controlled entry areas (locked doors) within the reactor and turbine buildings with the exception of the isolation valve MCC 1AB2 which is located on the 23 foot level in the reactor building.

The DC system including the emergency supplies is shown in Figure 4 attached. The battery chargers MGA and MGB which are powered from vital motor control systems 1A2 and 1B2 respectively, normally assume the station DC load. In the event of a loss of AC motor power for the battery chargers, the 125 volt station batteries (A and B) assume the DC load and are designed to carry the load for an eight-hour period. A static charger which is also powered from either vital MCC-1A2 or MCC-1B2 is provided to accommodate planned maintenance on either MG set A or B. All rotating and electrical switch gear for the DC system is located in a locked room within the reactor building. Mr. Riggle informed the inspector that the opening of the battery breakers, either Battery A or Battery B, would not be annunciated nor indicated within the control room but would be detected during the normal shift inspection by a roving operator of these spaces. This condition would be detected by a loss of trickle charging current.

c. <u>Surveillance of Emergency Power Equipment</u>

Discussions were held with Mr. J. Carroll and he informed the inspector that a roving operator patrols both the reactor building and the turbine building, and the diesel generator building on a shift basis.

Check sheets that require initialing, are provided the operator to ensure that all spaces are checked. A review of the check sheets disclosed that there are no specific entries required for logging information such as breaker positions, battery charging currents, or any other locally monitored variables in the emergency power system. Mr. Carroll informed the inspector that the operator would be expected to record all off-normal conditions on the remarks section of the log sheet.

Surveillance checks to determine the availability and functional operability of the emergency power systems include a bi-monthly startup and partial loading (20%) for each diesel generator and the functional test of the diesel generators (100% loading in 15 seconds) during each refueling outage. These tests are required by the Technical Specifications. In addition, the 125 volt station batteries are discharge load-tested each six months.

2. Diesel Generator Performance

Review of the station surveillance testing records for the diesel generator weekly starts disclosed the following:

a. During the period from February 21, 1970 to October 3, 1970 there were five instances in which No. 1 diesel generator did not either start on the first attempt (three cranking cycles without starting) or tripped from some other problem. On four occasions (February 21, 1970, July 26, 1970, September 20, 1970, and October 3, 1970) diesel generator No. 1 would not start on the first attempt. After several trys at starting the diesel, the diesel did start and came up to speed. On June 28, 1970, the No. 1 diesel generator tripped as a result of a 55 relay actuation for which no cause could be found. The diesel generator was restarted shortly thereafter with no trouble. On September 20, 1970, after starting the diesel, the diesel generator tripped out while trying to synchronize. This condition was reset and the diesel started again satisfactorily but while shutting down the diesel the governor was very unsteady. Following the October 3, 1970, event when the diesel generator did not start automatically on the first attempt, a work order was issued and repairs were currently in progress during the inspection to relocate the holding brackets for the dual starting motors on each diesel generator.

b. During a surveillance test of the No. 2 diesel generator on June 28, 1970, the output breaker opened at a load of approximately 400 kilowatts. No explanation could be provided to the inspector to either the corrective action taken or the problem with the breaker.

The subject of the diesel generators performance was reviewed during the exit interview. Mr. Ross was informed by the inspector that it was apparent that the surveillance testing records were not being reviewed to determine potential or real problems with safeguards equipment. Mr. Ross was also informed that out of five instances of problems with the No. 1 diesel generator there were only two work requests that had been issued to correct the troubles and one of these work requests was incomplete in that it did not include the failure of the diesel generator to start on the first attempt. Mr. Ross informed the inspector that as soon as Mr. Don Reeves completes the senior operating exam (scheduled for October, 1970) he will be assigned the overall responsibility for the surveillance testing program and he would be expected to review surveillance testing records to detect such problems.

3. Diesel Generator Starting Battery Testing (Item of Noncompliance)

No records were available to determine if the quarterly temperature and electrolyte levels for the starting batteries was being accomplished in accordance with Technical Specifications paragraph 4.7.A.5. Mr. Riggle also informed the inspector that these tests were not being accomplished. This item was identified as an item of noncompliance during the exit interview and Mr. Ross stated that the tests would be accomplished as required.

4. 125 Volt DC Station Load Testing

Surveillance testing records for the semi-annual station battery discharge load test were reviewed with Mr. Riggle. Records for the A battery indicated that on two of the last five discharge load tests, 100% of ampere-hour capacity was not obtained. In addition, on four of the last five tests, the manufacturer's recommended minimum cell voltage was exceeded during the discharge load test. Mr. Riggle informed the inspector that the acceptance standards for the load testing were: (a) 1200 ampere-hours capacity over an eight-hour discharge rate, (b) a minimum cell voltage of 1.75 volts, and (c) a minimum battery terminal voltage of 105 volts. The discharge test is started after completing a 24-hour equalizing charge on the battery. A tabulated summary of the A battery discharge testing is provided below:

A Battery

Date	Beginning Voltage	Rate of Discharge	Duration of Discharge Test	Pilot Cell Voltage	Terminal Voltage	% of Ampere Hour Capacity
3/16/69	125	150	8 hrs	1.70	107	100%
9/18/69	121	150	8 hrs	1.79	108	100%
4/1/70	125	150	7 hrs 40 min	1.67	105	95.7%
9/29/70	124	150	7 hrs 40 min	1.57	103	89.6%
10/13/70	124	150	8 hrs	1.62	105	100%

Surveillance test records for the B battery indicated that on two of the last four tests, the measured ampere-hour capacity has been less than rated. Tabulated below are the results of the tests.

B Battery

Date	Beginning Voltage	Rate of Discharge	Time of Discharge	Pilot Cell Voltage	Terminal Voltage	% of Ampere Hour Capacity
3/12/69	129	150	7 hrs 15 min	1.75	106.5	90%
9/19/69	126	150	8 hrs	1.74	106	100%
3/24/70	122	150	7 hrs 30 min	1.72	105	92%
9/23/70	125	150	8 hrs	1.76	107	100%

This subject was discussed during the exit interview and used as an example to indicate that management was not adequately reviewing the results of surveillance tests. Management was only aware of one (September 29, 1970) failure of the load testing on these batteries to meet the acceptance criteria. Mr. Ross again indicated that Mr. Don Reeves will assume the overall responsibility for surveillance testing pending his successful completion of the senior operator test which was scheduled for October, 1970. The inspector did not identify this issue as an item of noncompliance but did state his concern for the failure of the station management to recognize that the test had not meet the specified acceptance criteria. The inspector was informed during the exit interview that JC was currently discussing the battery performance with the battery supplier to determine if replacement cells were warranted.

5. Surveillance Testing of 125 Volt Station Batteries - Quarterly Tests (Item of Noncompliance)

Surveillance testing records were reviewed and they disclosed that the quarterly tests on that 125 volt station battery had not been performed in entirety. Specifically lacking were records to reflect that electrolyte level and the temperature of every fifth cell had been measured, as required by Technical Specification 4.7.b.3. This subject was discussed during the exit interview and Mr. Ross stated that it was JC's intention to perform these tests. The inspector identified this as an item of noncompliance.

6. Diesel Generator Annual Overhaul Surveillance Testing (Item of Noncompliance)

Station records disclose that the annual diesel generator inspection required by Technical Specification 4.7.A.3. had not been completed since the issuance of the provisional operating license on August 9, 1969. Discussions with Mr. W. Riggle disclosed that a maintenance procedure for the inspection had been written and was awaiting PORC review and approval before its implementation. This issue was discussed during the exit interview with Mr. Ross. He stated that it was JC's intention to meet this Technical Specification and that the annual inspections would be done during the month of November, 1970. The inspector identified this issue as an item of noncompliance.

Q. Radioactive Waste Systems

1. Carbon-14 Issue*

Discussions were held with Mr. D. Ross and the inspector informed him of the Public Health Service survey of the Yankee effluents. He was informed that carbon-14 had been detected and he was appraised of the relative concentrations in both the radioactive liquids and gaseous wastes at the Yankee plant. Mr. Ross was encouraged, in view of the fact that no sampling for carbon-14 has been performed at the Oyster Creek facility, to analyze for this radioisotope. He informed the inspector that a sample of gaseous and liquid wastes would be obtained and the concentrations of carbon-14 in the samples would be determined by an off-site laboratory.

2. Stack Sampling

As a result of prior commitments and understandings between the licensee and CO:I, JC installed a stack sampler to sample at the 240 foot elevation. Sampling was performed during the period of July 13-17, 1970. Plant conditions at the start of the sampling were that the plant had been operating at 530 Mwe for three weeks with the chemistry in the primary system in equilibrium. Two sample holders were installed approximately 18 feet down from the origin of the sample point at the 240 foot elevation. Tabulated below are the results of the sampling program:

Iodine 131

Date and Time	Sample Location: Base of Stack	Sample Location: 240' Elevation
0830 6/30/70 to 1317 7/2/70	1.66×10^{-10} uCi/cc	
1320 7/2/70 to 1005 7/7/70	1.73×10^{-10} uCi/cc	
1010 7/7/70 to 0825 7/10/70	1.70×10^{-10} uCi/cc	
0830 7/14/70 to 1600 7/14/70	2.06×10^{-10} uCi/cc	
1555 7/14/70 to 0831 7/15/70		1.77×10^{-10} uCi/cc
0831 7/15/70 to 2000 7/15/70		1.48×10^{-10} uCi/cc
2000 7/15/70 to 0840 7/16/70	2.06×10^{-10} uCi/cc	
0840 7/16/70 to 2010 7/16/70		1.64×10^{-10} uCi/cc
2020 7/16/70 to 0855 7/17/70	1.25×10^{-10} uCi/cc	

*Memorandum from J. P. O'Reilly to Senior Reactor Inspectors, "Detection of Carbon-14 in Power Reactor Effluents", dated June 17, 1970.

Iodine 133

Date and Time	Sample Location: Base of Stack	Sample Location: 240' Elevation
0830 7/14/70 to 1600 7/14/70	1.23×10^{-10} uCi/cc	
1555 7/14/70 to 0831 7/15/70		1.41×10^{-10} uCi/cc
0831 7/15/70 to 2000 7/15/70		1.19×10^{-10} uCi/cc
2000 7/15/70 to 0840 7/16/70	2.25×10^{-10} uCi/cc	
0840 7/16/70 to 2010 7/16/70		1.53×10^{-10} uCi/cc
2020 7/16/70 to 0835 7/17/70	1.76×10^{-10} uCi/cc	

The samples were collected on a standard CESCO charcoal cartridge and counted on a Nuclear Data model 2200 multi-channel analyzer utilizing a 3 x 3 inch NaI (Tl) crystal. Calculations were based on the 0.36 and 0.53 Mev peak of I-131 and I-133 respectively.

During the sampling periods of July 14-15, 1970 and July 15-17, 1970, charcoal cartridges were in both the 242 foot elevation and the normal stack sampler at the base of the stack, i.e., filters in series offering a check on efficiency of the individual cartridge for iodine retention. In both instances the first cartridge (240 foot) retained greater than 80% of the iodine collected.

Gelman fiber type E filter (99.5% removal for DOP) preceded the charcoal cartridge. After allowing 24 hours for decay of the short-lived daughters of the fission gases, the filters were counted on a multi-channel analyzer. No γ emitting nuclides were noted.

Repeat tests for particulate and iodine are planned when either (a) the fission gas levels increase by a factor of approximately 5, or (b) evidence of long-lived (greater than eight day 1/2 life) particulate activity is found on the routine weekly samples of the gaseous effluents.

3. Current Gaseous Release Rate

Records were reviewed during the inspection and disclosed that the off-gas grab sample records indicated a range of 3.27 to 7.0×10^3 uCi/second. Grab samples are taken each week from the discharge of the air ejector condenser after the gas has been delayed for approximately 30 minutes. Normal flow through the off-gas line was reportted to be approximately 100 cfm.

4. Liquid Radioactive Waste

Liquid sampling records for activity released from the facility for the month of August, 1970, were reviewed and disclosed the following information. Tritium - 2.56 curies identified isotopes 0.46 curies.

T. Facility Modifications

1. Modifications to the Isolation Condenser Initiating Logic Circuitry

A change in the relay matrix logic was made in August, 1970, to the initiating circuit to prevent the closure of a single excess flow check valve from preventing automatic initiation of the condensers (1060 psi for 20 seconds). (See Section E.4.)

2. Control Rod Drive Inner Filters

All operating control rod inner filters were changed to 10 mil filters during the April-May, 1970, rod work outage.

V. Reliability Information

1. Diesel Generator Performance

From February to October, 1970, there have been five occasions when the No. 1 diesel generator either failed to start on the first attempt or tripped off the line due to electrical problems. During this period the No. 2 diesel generator also tripped off the line due to an electrical problem during surveillance testing. (See Section N.2.)

2. 125 Volt Station Battery Load Testing

Since March of 1969, the semi-annual load testing of battery cells A and B failed on five occasions to meet the minimum acceptance criteria recommended by the battery manufacturer. (See Section N.4.)

Figure 1

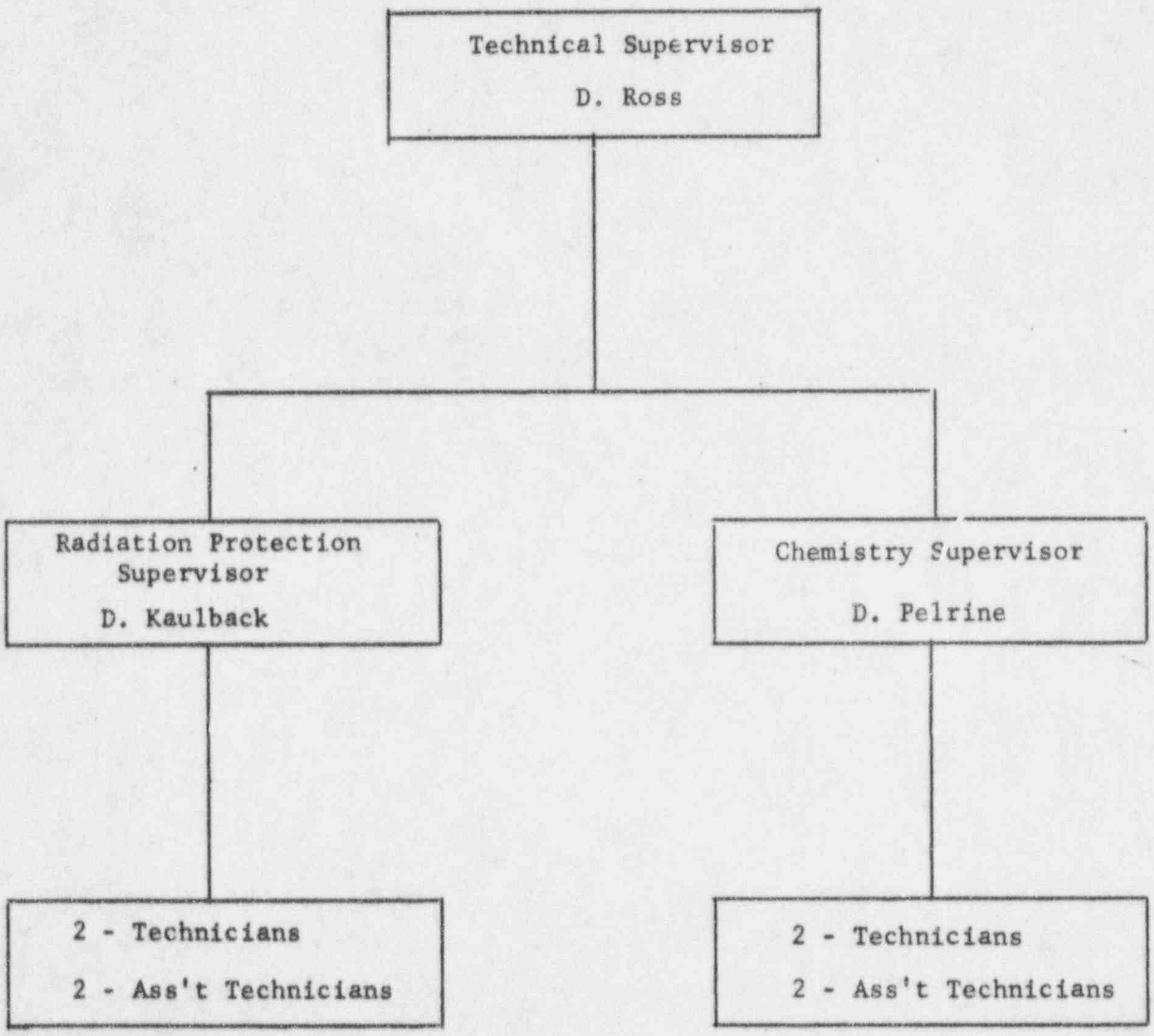

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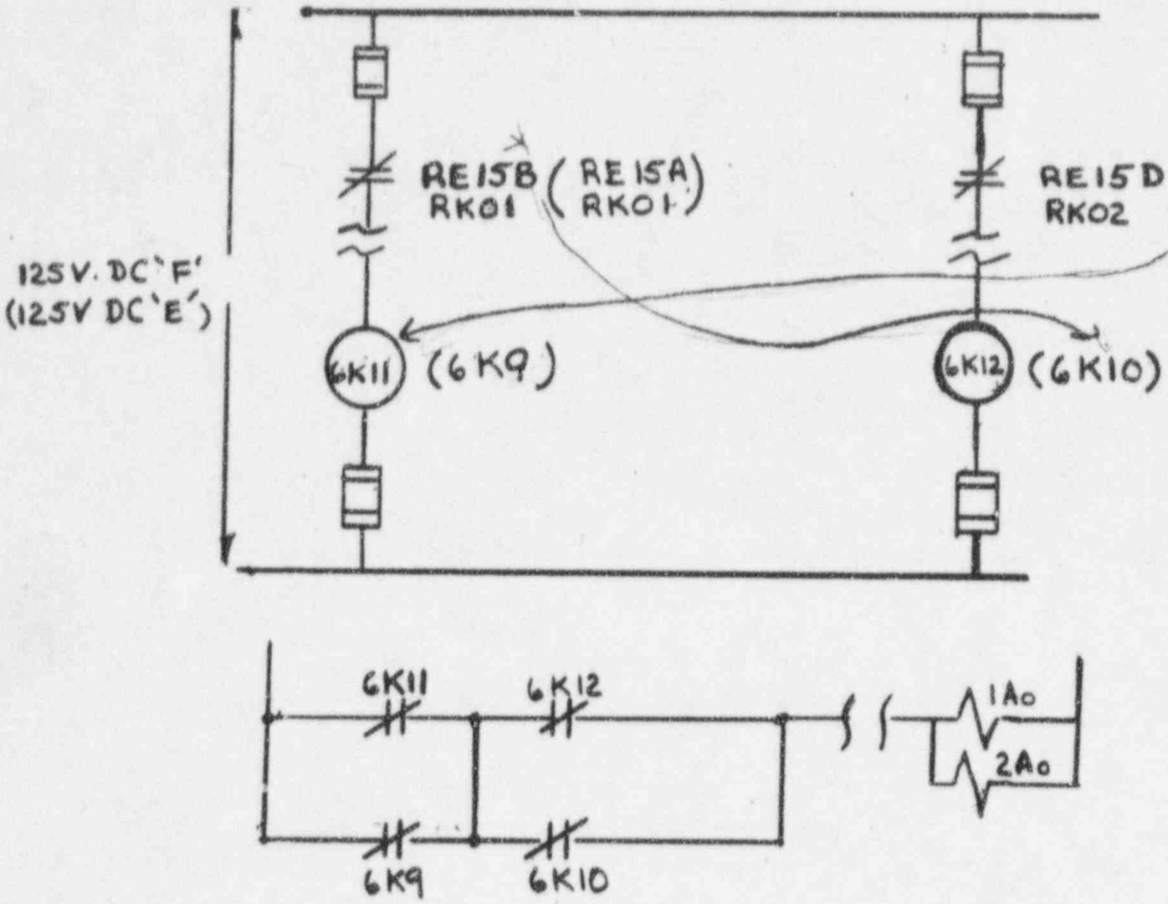
Figure 1

Figure 2



RE15A, B, C, D ~~open~~ close on high reactor pressure 1060 psi.
6K9, 10, 11, 12 are 20 sec TDDO relays.
1Ao and 2Ao are relays that put the isolation condenser in service when de-energized, i.e., open the condensate return line valve.

NOTE: The original circuit would have, under conditions of a loss of pressure sensing capability for one instrument rack (RK01 or RK02), been unable to actuate the isolation condensers on a high reactor pressure. With the modified design shown above, a loss of RK01 would only prevent 6K9 and 6K11 from proper operation as the actuating pressure switches for these relays (RE15A and RE15B) would be the only pressure switches affected. Under conditions of a loss of one instrument rack, the power can still be interrupted to 1Ao and 2Ao by operation of 6K10 and 6K12 which are actuated from RE15C and RE15D and which are located in the other instrument rack RK02. The circuit modifications from the old to the new involve the interchange of actuating pressure switches RE15B and RE15C which previously actuated 6K10 and 6K11 respectively. RE15A and RE15B remain in RK01 and RE15C and RE15D remain in RK02 as in the original design. Two additional sets of fuses were added to provide individual fusing for all of the actuating relays 6K9, 10, 11 and 12 to preclude one blown fuse from preventing proper operation. In the original design relays 6K9 and 6K10 were powered through a common fuse as were relays 6K11 and 6K12. Thus, if a fuse had blown, the automatic actuating capability would have been lost.

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Figure 2

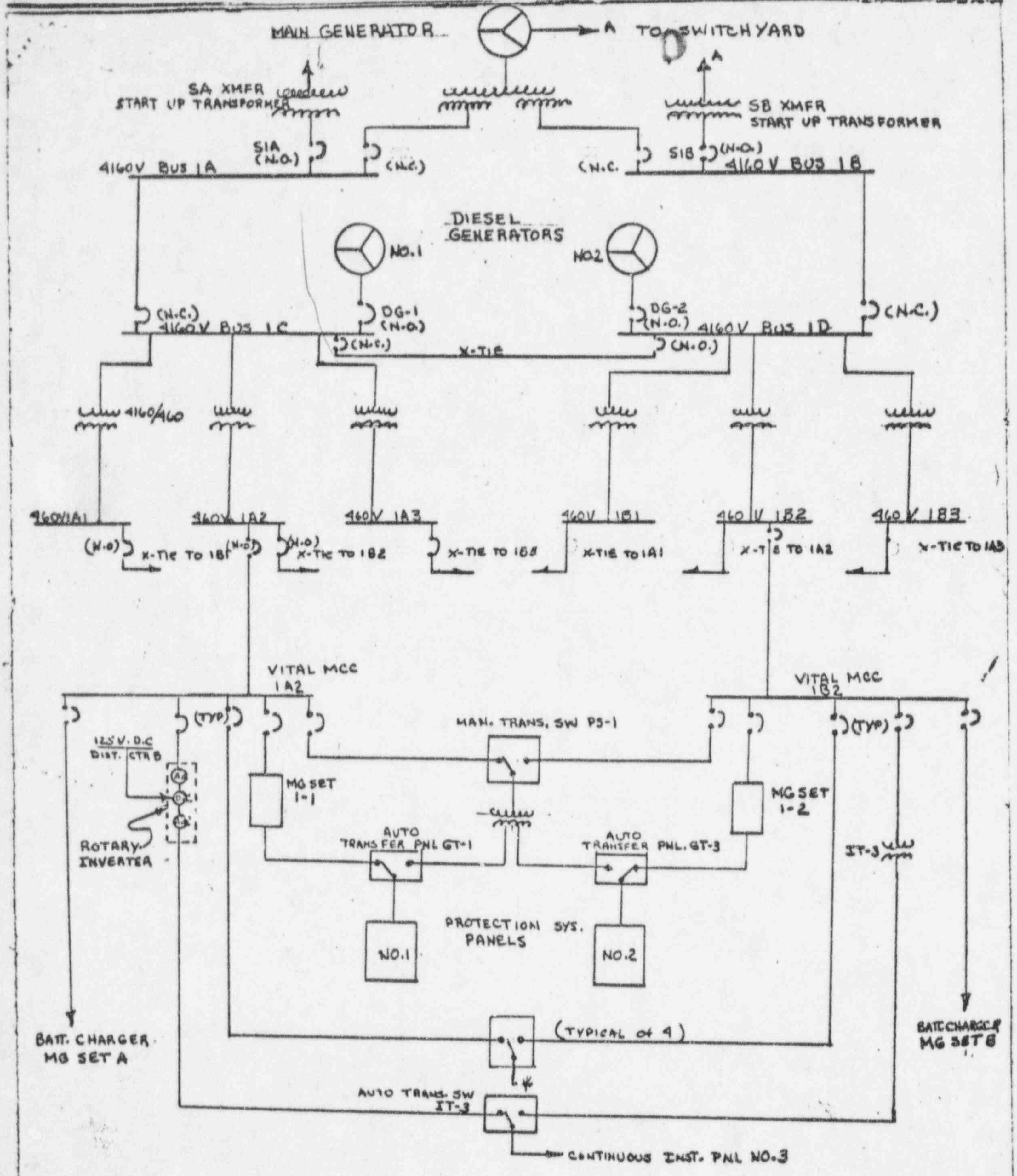

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Figure 3

* AUTO TRANSFER SWITCHES FEEDING MCC-1AB2, INST. PNL NO.4, PNL VACP-1, VITAL LIGHT NO.1 [illegible]

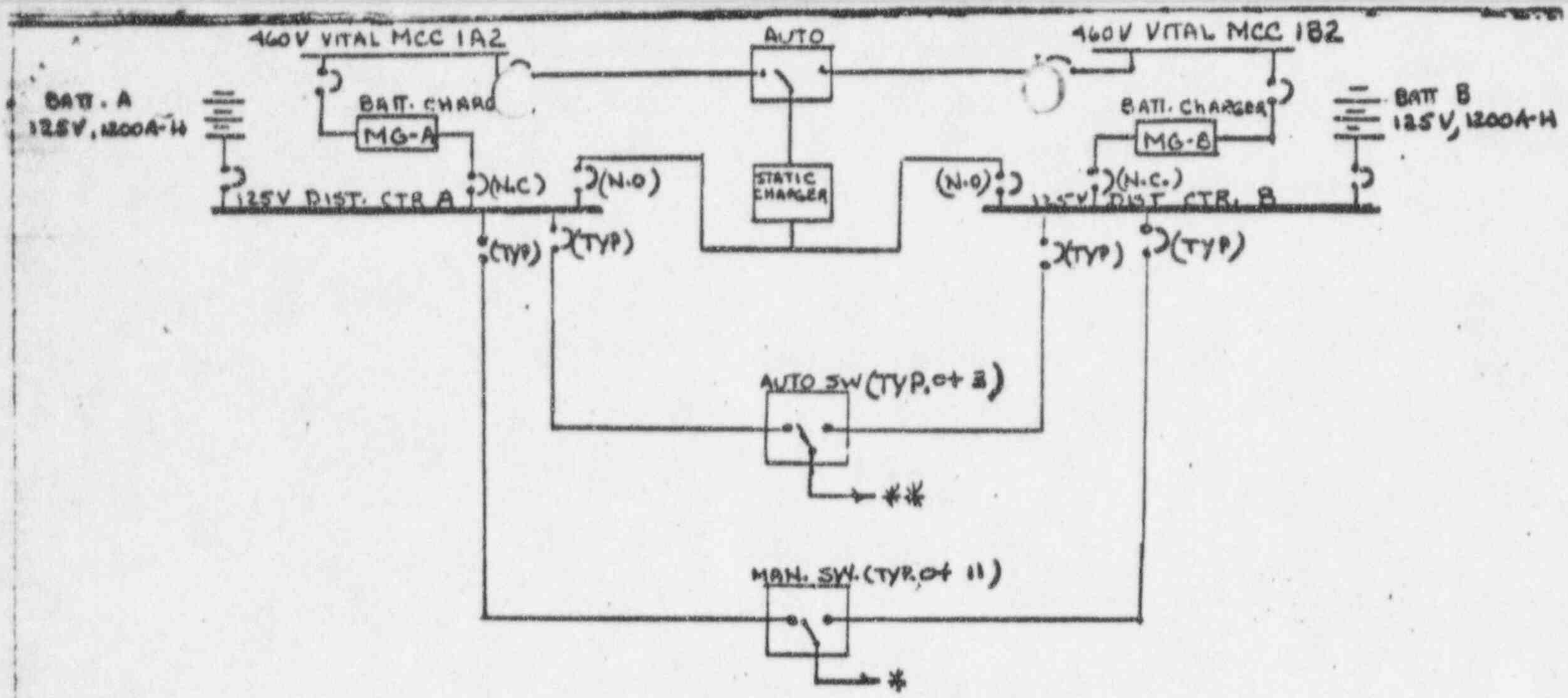

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Figure 4

* MANUAL TRANSFER SWITCHES FEED CONTROL PWR TO 4160 AND 460 V SWITCH GEAR AND DG CONTROL PWR.
** AUTO TRANSFER SWITCHES FEED MCC DC-1, PWR PNLS D AND E