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

Report of Inspection

CO Report No. 219/71-1

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

Dates of Inspection: April 6-8, 12, 13, 1971

Dates of Previous Inspection: October 13-16, 28, 29 and November 5, 6, 1970

Inspected by: R. J. McDermott 6/21/71
R. J. McDermott, Reactor Inspector (Principal) Date

Reviewed by : R. T. Carlson 6/21/71
R. T. Carlson, Senior Reactor Inspector Date

Proprietary Information: None

SCOPE

Type of Facility: Boiling Water Reactor

Power Level: 1690 MWt

Location: Forked River, New Jersey

Accompanying Personnel: Mr. D. Pomeroy, Technical Support Branch, CO:HQ on April 6-8, 1971 to review liquid effluent monitoring and the performance of the liquid rad waste system. His report is included as Attachment No. 3.
Dr. C.Pelletier, Technical Support Branch, CO:HQ accompanied for orientation.
Mr. G. Fredrickson, GAO, was present on all dates.

Scope of Inspection: An announced routine inspection was made to review plant operations for the inspection period, review the status of previously identified problems, review liquid effluent monitoring practices and review the performance of the liquid radwaste system.

B/418

SUMMARY

Safety Items -

1. Liquid Waste System Performance - A review of the liquid radwaste system design objectives was not made because the licensee has been unsuccessful in determining what the objectives are. The process design drawings usually available for review at the site apparently do not exist for the Oyster Creek station. A review of the expected performance of the radwaste system as outlined in the FD and SAR was made and compared with the system performance and substantial variances were observed. (Attachment No. 3 and Noncompliance Item No. 5)

2. Liquid Effluents - It appears that the licensee exceeded the annual average concentration limit specified in Technical Specification 3.6.b.1.a. for calendar year 1970. JC has reported* the release of 4.5 curies for calendar year 1970 (approximately 17% of annual limit) which was established by a gross beta analysis of liquid waste. Plant records and independent sample analyses** indicate a nonconservative error of approximately 300% in the gross beta analysis which was being used by JC. Additionally, the ratio of total activity in OC-1 liquid waste to the activity determined by an accurate gross beta analysis by JC has been measured as approximately 300% for a total nonconservative error of 900%. Based on these errors, it would appear that the actual plant releases may have been as high as 40 curies (isotopic analysis) during calendar year 1970 and that the average annual concentration limit for unidentified radionuclides*** (1 x 10^{-7} uCi/ml) was exceeded by a factor of approximately 1.5. From these findings, it would appear that a gross beta analysis alone is insufficient to determine the total activity in liquid wastes. (Attachment No. 3)

Noncompliance Items -

1. Title 10 CFR Part 20.201 (Surveys) states "each licensee shall make or cause to be made, such surveys as may be necessary for him to comply with the regulations in this part".

   Contrary to the above, it appears that adequate surveys (analyses) were not conducted to evaluate the radiation hazards incident to the presence and release of radioactive liquid waste. A significant nonconservative error was noted in the method used for gross beta analysis of the activity in the radioactive liquid waste. This was determined to have resulted from an incorrect calibration of the counting equipment. It was further evident that the ratio of total activity to the activity determined by gross beta analysis in liquid waste was not used as a factor in determining and reporting actual plant releases or in controlling the activity contained in outside storage tanks.

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*Semi-annual Report No. 3.

**Samples analyzed by OC-1, NYOO and Idaho Operations Office.

***Technical Specification 3.6.B.1.a.

2. Technical Specification 3.6.B.1.a. states "The annual average concentration of unidentified radionuclides shall not exceed 1×10^{-7} uCi/cc."

Contrary to the above, it appears that the 1970 annual average concentration of unidentified radionuclides released from the plant in liquid wastes exceeded 1×10^{-7} uCi/cc. This finding was based on an appropriate upscaling of releases that have been reported to the AEC in semi-annual report No. 3. The appropriate correction was determined by a review of analysis records at OC-1 and by comparisons of independent samples analyzed by AEC laboratories with similar samples analyzed at OC-1.

3. Technical Specification 3.6.B.2. states "Radioactive liquid effluents being released into the discharge canal shall be continuously monitored".

Contrary to the above, the continuous in-line monitor that is used to satisfy this Technical Specification requirement was considered to be incapable of performing its intended function as a result of high background radiation. Based on our review, it is estimated that the facility could have released at the rate of 400 curies/year without tripping the in-line monitor alarm.

4. Technical Specification 3.6.C. states in part that "the maximum amount of radioactive liquids in storage in the waste sample tanks, the floor drain sample tanks, and the waste surge tank shall not exceed 0.7 curie."

Contrary to the above, the gross beta analysis disclosed a total of 0.316 curies in these tanks on March 10, 1971. By appropriately upscaling the gross beta analysis to correct for identified errors in this counting technique, it appears that the total curie content of these tanks may have been at least as high as 1.8 curies.

5. Section 3.C.(2) of the license states that "Jersey Central shall report to the Director, DRL, within 30 days of its observed occurrence, any substantial variance disclosed by operation of the facility from performance specifications contained in the facility description and safety analysis report (safety analysis report) or the Technical Specifications".

Contrary to the above, no written report was submitted when the liquid radwaste system performance varied substantially from information presented in the FD and SAR. Specific examples of the above include:

   a. Section IX-4, paragraph 4.2 of the FD and SAR states the waste collection drains are "essentially always reused as condensate." During 1970 less than 10% of waste collection drains were reused as condensate, the remaining was released to the circulating water system.

   b. Table IX-4-2 of the FD and SAR lists the expected concentrations of radioactivity in the floor drain system as a function of fuel performance and interpolation of this table for an off-gas release rate of 10,000 uCi/second (which is typical of station performance for December, 1970 to March, 1971) indicates that the expected floor drain concentration is 0.5×10^{-4} uCi/ml. Actual concentrations range from 10 to 50×10^{-4} uCi/ml or 20 to 100 times the expected value.

c. Section IX-3, paragraph 3.1.3.3. of the FD and SAR states that the chemical waste will be sampled and concentrated as required. Our review indicated that these tanks had never been sampled and that during 1970 essentially no concentrating of these wastes was performed.

d. Table IX-4-3 of the FD and SAR lists the concentrations of various isotopes expected in the liquid effluents for operation with and without fuel failures. The expected I-131 for operation with failed fuel (100,000 uCi/second stack gas) is listed as 0.03% of the total activity. I-131 is not expected in the absence of fuel failures. The Oyster Creek semi-annual report for the period ending December 31, 1970, lists the isotopic composition of a floor drain sample and indicates that iodine-131 comprises approximately 16% of the activity in this batch. Other isotopes were also determined to be at substantial variance from the expected values.

Unusual Occurrences -

1. Torus Vacuum Breaker Valves - Failure to Open* - Both torus vacuum breaker block valves failed to open during surveillance testing on December 18, 1970. Adjustments were made to the valves and the valves were successfully retested on that date but testing on the following date resulted in both valves again failing to open. The problem was determined to be an incorrect valve operator linkage adjustment which was corrected. Subsequent tests have been performed without additional problems. (Section K.1.)

2. Leaking Containment Isolation Valve** - During the containment integrated leakage test conducted in October, 1970, excessive leakage was noted through the torus O_2 sample line isolation valve which was closed. Both the torus and drywell O_2 sample line isolation valves were replaced with a different design valve. JC plans to install double isolation valves and to reroute the O_2 sampler exhaust back to the drywell. (Section K.3.)

Status of Previously Reported Problems - Following the last routine inspection, a form AEC-592***, identifying six (6) items of noncompliance, was sent to the licensee on December 7, 1970. A satisfactory reply was received from JCP&L on

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*Letters from JC to DRL dated December 23, 1970 and February 8, 1971. Inquiry Memoranda Nos. 219/70-K and 219/70-L.

**Letter from JC to DRL dated November 2, 1970 and Inquiry Memorandum No. 219/70-I, dated October 29, 1970.

***Form AEC-592 sent to Mr. R. H. Sims, Vice President, December 7, 1970.

December 22, 1970.* The proposed corrective measures discussed in the reply were reviewed during this inspection and found to have been implemented.

Other Significant Items -

1. Administration and Organization - A review of meeting minutes for both the on-site and off-site safety committees reflected improved performance. Audit schedules have been developed and implemented by the off-site safety committee to review plant operations. The technical staff at OC-1 has been increased by seven new additions since the last routine inspection and attempts are being made to add five additional engineers to the site staff. Three additional persons in the operating organization have received SOL's and additional maintenance personnel have been added. The surveillance testing program is now being coordinated by one responsible individual. (Section B.)

2. Control Rod Performance - Scram time performance appears satisfactory with only a slight upward trend in the time required to fully insert. Totalized withdrawal stall flows for all rods appears to have stabilized. No unusual operating problems were noted. (Section F.3.)

3. Core Spray System - OC-1 has been experiencing water hammer in both core spray loops during surveillance testing (pump starts). Pipe movements of up to 6" are being experienced for piping outside of the drywell. Cause is attributed to: (a) the pump discharge piping not being full of water and completely vented and, (b) starting of the core spray booster pumps which closes the bypass check valve around these pumps. The off-site safety committee has requested Burns & Roe to perform a stress analysis of the piping system and this is currently in progress. Based on the results of this analysis, NDT inspection of selected areas of the piping system will be performed, if warranted. GE is providing a field change to consist of a small filling pump which takes its suction from the torus to keep the piping systems full. JC has requested GE to determine if these pumps could develop sufficient pressure to prevent the core spray pump discharge check valves from opening. It was also determined that a number of pipe anchors had been pulled free during preoperational testing of this system and that the pipe anchors were subsequently modified (strengthened). A weld defect in the core spray system was identified in October, 1970 during JC's review of the original construction NDT records. This review wa required by Section 6.6.A. of the Technical Specifications. This weld was radiographed because the original RT record could not be located.** Repair welding was also performed in October, 1970. Repair records were reviewed and no deficiencies were noted. (Section L.)

4. Drywell Leak Detection Procedures - A review was made of the leak detection procedures and equipment. Revised procedures went into effect during the inspection. Evaluation of an independent means to detect unidentified leakage by weekly sampling of the drywell atmosphere is continuing by JC but no meaningful results have been achieved. (Section K.4.)

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*Reply to Form AEC-592 dated December 22, 1970.
**Letter to DRL from JC dated September 30, 1970.

5. Mini-Stretch Power Increase - Licensed power level was increased from 1600 MWt to 1690 MWt on December 2, 1970. The facility was shut down on December 5, 1970 to make plant modifications required for the power increase. The facility achieved 1690 MWt on December 10, 1970. Testing was completed at this power level and all test results were reported to have met the acceptance criteria. (Sections C and S)

6. Turbine Initial Pressure Regulator Performance - Operation of this control system has remained steady since the last inspection. GE replaced four (4) of the aluminum horizontal "push-pull" linkages for the bypass valves with carbon steel material. GE investigation of the failed linkage* disclosed the material did not meet design specifications. (Sections H.1. and H.2.)

7. Torus Vacuum Breaker Block Valves** - No additional problems have been experienced with the proper opening of these valves. Valve internals were inspected and rubber liners were replaced in January, 1971. (Section K.1.)

8. Containment Integrated Leak Rate Test - This test was successfully completed in October, 1970. A summary report of this test is being prepared by JC. Repairs were required to the torus O_2 sampling isolation valve due to excessive leak-through and therefore the containment will be retested in one year as required by Technical Specifications. The containment isolation valves for both the drywell and torus O_2 sampling lines were replaced with a different design and GE is to provide a design modification to install double isolation valves in these lines and additionally return the exhaust of the O_2 analyzer back into containment. (Section K.3.)

9. Local Power Range Monitors - OC-1 has experienced 13 failures of LPRM channels. Cause is attributed to moisture entering cable connectors within the drywell. (Section F.2.)

10. Gaseous Effluents - A review disclosed that reported results of stack off-gas may be $\sim$ 10% low. The error is attributed to JC's failure to correlate the six radionuclides measured in off-gas releases with the sum of the 22 radionuclides present. JC corrected this condition following the inspection. (Section Q.2.)

11. Emergency Power - A review disclosed that load testing of the 125 V station batteries resulted in a terminal voltage at the end of the 8-hour test that was below the acceptance limit. JC corrected this condition following the inspection by cleaning the cell interconnectors. (Section N.2.)

12. Compressed Air Systems - No problems were identified in a review of air systems but the licensee has stated that surveillance practices to determine proper system operation will be reviewed. (Section I.)

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*CO Report No. 219/70-6.

**Letters from JC to DRL dated December 23, 1970 and February 8, 1971; Inquiry Memoranda Nos. 219/70-K and 219/70-L entitled "Failure of Torus Vacuum Breakers"

Management Interview - Exit interviews were conducted with Mr. McCluskey on April 8 and 13, 1971 and by followup telecon on May 11, 1971. Mr. G. Fredrickson of the GAO was present during both site exit interviews.

Exit Interview on April 8, 1971 - Messrs. Pomeroy and McDermott discussed deficiencies and apparent items of noncompliance that were identified during Mr. Pomeroy's review of liquid effluent monitoring and radwaste system performance. (See Attachment No. 3)

Exit Interview on April 13, 1971 and Followup Telecon on May 11, 1971 - Mr. McDermott held discussions with Messrs. McCluskey, Ross, Carroll, Riggle and Reeves. Issues discussed were as follows.

1. Control of Liquid Radioactive Wastes - The inspector stated that his review on April 13, 1971 disclosed that JC had not corrected (in entirety) the discrepancies identified during the April 8, 1971 exit interview concerning the release and storage of liquid effluents. Mr. McCluskey was informed that the facility was still controlling on a gross beta analysis and even though the errors had been corrected in this method of analysis, JC was still not relating this analysis to the total activity in the wastes (isotopic analysis). Mr. Ross proposed then to measure gross beta activity and add to this result the concentration of radionuclides known to be only gamma emitters (Mn-54 and Cr-51). After further discussion, Mr. McCluskey agreed to do an isotopic analysis for each batch of liquid waste before discharging it. The mixture MPC would be used to control effluent releases from the plant on all future dumps until a complete review could be made of the relation between the total activity and a gross beta analysis. He further stated that isotopic analysis would be used to control the curie content of outside storage tanks in the future. (Section Q and Attachment No. 3)

2. Core Spray Syste - Water Hammer - The inspector stated that he would closely follow the planned review of pipe stresses and NDT inspection. JC was requested to provide a written report to DRL when the stress evaluation is completed for the piping system. (Section L)

3. Generator Load Rejection and Turbine Stop Valve Closure Scrams - The inspector stated that his review had disclosed that the trip point for the pressure switches that bypass these two scram features below 45% power were not being verified during quarterly surveillance testing. The inspector pointed out that the Technical Specifications did not specifically require such testing but that testing was implied and expected to be performed. Mr. McCluskey stated that the surveillance test procedures would be changed to require this testing during future quarterly tests.

Mr. McCluskey was informed that no documentation was available to establish the basis for the set point for the 45% scram bypass pressure switches and that the present setting was established by verbal communication with a GE representative. It was pointed out by the inspector that the pressure switch setting was in effect a scram set point as two scram features are added to the safety circuit by actuation of these pressure switches and it would be expected that JC would have required more than a verbal communication to establish the set point. It was further pointed out that no testing was performed to verify the accuracy of this set point when the turbine was restarted after the modification was made. The inspector stated that a review of scram 15 indicated that the set points were adequate or conservative but that testing should have been performed following the modifications. Mr. McCluskey stated that GE would be requested to supply the necessary documentation to provide the basis for the set points.

4. Diesel Generator - Shutdown Devices and Past Failures - The inspector stated that a review of the records for the annual diesel generator inspection (required by Technical Specifications) indicated that no testing had been performed on the devices which shut the diesels down. It was pointed out that a premature actuation of these shutdown devices would reduce the reliability of the system. Mr. McCluskey stated that these checks will be performed, if possible, during future annual inspections.

   The reportability aspects of the past failures* of the diesel generators to start on the first attempt were discussed and Mr. McCluskey stated a report would be submitted in the near future.

5. Emergency Power - The inspector stated that the load testing records for the "B" 125 V station battery during March 1971 had indicated that the terminal voltage at the end of the 8-hour load test had dropped below the acceptance value and that no apparent action was taken to investigate the cause. Mr. McCluskey subsequently informed the inspector by telephone that the battery cell connectors had been removed and cleaned and another load test was completed satisfactorily.

6. Instrumentation - The inspector stated that an audit sampling of the calibration frequency for instrumentation associated with Technical Specification limits and with safeguard equipment indicated that no periodic schedule was in effect. Mr. McCluskey stated that a review of such instrumentation would be made and a program developed. The inspector stated that this area would receive followup review.

7. Stack Releases - The inspector pointed out that JC's method for analyzing stack releases could be as much as 10-15% low. Mr. McCluskey, in a followup telecon, stated that Mr. Ross was now using the necessary correction to eliminate this potential error.

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*CO Report No. 219/70-7, Unusual Occurrence item 2.

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. F. Kossatz, Mechanical Foreman, OC-1
Mr. D. Kaulback, Radiation Protection Supervisor, OC-1
Mr. N. Goodenough, QA Engineer (Radiography), GPU
Mr. T. Pelrine, Chemistry Supervisor, OC-1
Mr. D. Reeves, Technical Engineer, OC-1

B. Administration and Organization

1. Site Organization

a. Additions to Technical Staff

The present technical staff organization is outlined in Attachment No. 1. Since the last routine inspection, the following personnel have been added:

(1) Mr. J. Budd joined as an Associate Engineer on November 30, 1970. Mr. Budd graduated from the New York Maritime Academy with a Bachelor of Science in 1968 and majored in Nuclear Science. He holds a second engineer's license.

(2) Mr. R. Stoudnour joined as a Chemical Engineer in December, 1970. Mr. Stoudnour has a Bachelor of Science in Chemistry and has worked in various water laboratories at conventional stations since June, 1964.

(3) Mr. E. Rosenfeld joined as an Associate Engineer in December, 1970. Mr. Rosenfeld holds a Bachelor of Science in Chemical Engineering and has no previous industrial experience.

(4) Mr. A. Rone joined as an Associate Engineer in December, 1970. Mr. Rone has a Bachelor of Science in Electrical Engineering and no previous industrial experience.

(5) Mr. D. T. Allred joined as a staff engineer in December, 1970. Mr. Allred is on loan from the Jersey Nuclear Corporation.

(6) Mr. C. Konta joined as a Chemical Technician in January, 1971.

(7) Mr. M. Oberstadt joined in January, 1971 as a Radiation Technician.

b. Recruitment Efforts

Mr. McCluskey informed the inspector that attempts are being made to recruit two additional staff engineers, two additional assistant technical engineers, and one additional associate engineer.

c. Operations Organization

Mr. McCluskey informed the inspector that there are currently five complete operating shifts with the exception of the shift foremen who remain on a four-shift cycle. The number of individuals in each position are as listed below:

(1) Shift Foremen - 4

(2) Operating Foremen - 3

(3) A Control Room Operator - 5; B Control Room Operator - 5

(4) A Equipment Operators - 10 ; B Equipment Operators - 5

d. Maintenance Organization

The present maintenance organization is as outlined in Attachment No. 2. There are plans to add one additional welder and one additional A mechanic.

2. Plant Operating Review Committee (PORC) Performance

Between the period between October 2, 1970 and March 11, 1971, there were 13 meetings of the Plant Operating Review Committee. Four of these meetings had no attendance by GORB members. The remaining nine had one or the other of the two permanently assigned GORB members present. The quality of the PORC meeting minutes reflected an improvement in the documentation of the reviews from past performance.

3. General Office Review Board (GORB) Performance

During the time period between October 26, 1970 and January 15, 1971, there were four meetings of the General Office Review Board. The present committee membership is made up of the following personnel:

Mr. W. H. Hurst, Chairman
Mr. D. E. Hetrick, GPU, Vice Chairman
Mr. W. W. Lowe, Pickert and Lowe Associates
Mr. D. D. Reese, GPU
Mr. T. M. Schneider, GE
Mr. W. Southerland, GE
Mr. J. R. Thorpe, GPU
Mr. I. R. Finfrock, Jr., JCP&L

The only changes in the committee membership were the addition of Mr. Hetrick and the removal of Mr. G. H. Ritter. Meeting minutes were reviewed by the inspector and considered to be of improved quality from past performance.

4. GORB Audits of Site Activities

Audits of site activities were conducted by Mr. W. W. Lowe and Mr. N. Goodenough on October 14, 1970 and by Messrs. Hetrick, Crimmins, Reppert and Goodenough on January 12 and 13, 1971. At the time of the inspection an audit was being conducted by two GORB members. The minutes of GORB meetings reflected that the General Office Review Board has reviewed the results of the audits, as well as the action taken to correct identified deficiencies.

5. Surveillance Testing Program Administration

Discussions were held with Mr. D. Reeves during the inspection and the following information was obtained. Mr. Reeves was assigned overall coordinating responsibility for the surveillance testing program in October, 1970. The system currently in effect to schedule, test, and review results was reviewed with Mr. Reeves and the only identified deficiencies were as follows:

a. Not all testing procedures have been developed to date. Test procedures that are still lacking involve testing that is required during refueling outages.

b. Surveillance test procedures that are used during operational checks have not all been reviewed by Mr. Reeves to determine their adequacy.

c. The testing records do not lend themselves in entirety to evaluating the performance of the systems being tested. Examples of this would be no operating data is recorded during the testing of core spray pumps, and in many instances the as-found instrument calibrations or trip points are not recorded during calibration and trip checks.

These discrepancies were discussed with Mr. D. Reeves and improvement in these areas is expected.

6. Jersey Nuclear Company

Mr. McCluskey informed the inspector that Mr. D. Allred was provided to Jersey Central on loan from Jersey Nuclear Corporation in December, 1970. Jersey Nuclear Corporation has made available to the JC organization up to four nuclear engineers. Jersey Central has contracted Jersey Nuclear to supply reload fuel. The stated purpose of Jersey Nuclear providing nuclear engineers is to familarize these personnel with the operation of the Oyster Creek facility. At present, Mr. Allred is working for Mr. D. Ross on special assignments. Jersey Nuclear Corporation has not stipulated any restriction in the work assignments for the personnel they would provide to Jersey Central.

C. Operations

The facility was shut down on October 17, 1970 to perform the containment integrated leakage test. Miscellaneous maintenance was also accomplished during the shutdown and the reactor resumed operation on October 27, 1970. Operations continued until December 5, 1970 when the reactor was shut down for one day to make the final tie-in modifications required for the mini-stretch power increase (1600 - 1690 MWt). Reactor operations resumed on December 6, 1970 and continued until December 12, 1970 when the reactor was shut down because of an air leak within the drywell. The plant reached 1690 MWt on December 10, 1970. Operations resumed on December 13, 1970 and continued until December 25, 1970 when scrams 14 & 15 (described below) occurred. Operations resumed on December 27 and continued until January 25, 1971 when the reactor was shut down because the identified leakage increased to approximately 14 gpm in the drywell. The source of the leakage was determined to be a leak in the closed cooling water system coil located inside the drywell equipment drain tank. Operations resumed on January 27, 1971 and continued until February 13, 1971 when the reactor was shut down to inspect the turbine control valve seats. This investigation was prompted by the recent failure of the valve seat captive screws which occurred at the Nine Mile Point reactor. Operations resumed on February 19, 1971 and continued until March 3, 1971 when the unidentified leakage into the drywell increased to approximately 4.5 gpm. The source of this leakage was determined to be a pipe nipple leaking in the closed cooling water system supply to the cooling coil in the drywell equipment drain tank. Operations resumed on March 4, 1971 and no additional shutdowns have been experienced by the facility during the inspection period. OC-1 plans a fall 1971 outage for poison curtain removal.

Attachment No. 4 lists all scrams since the beginning of commercial operation. Listed below is a description of the unscheduled shutdowns during the inspection period:

Date	Cause
December 12, 1970	An orderly reactor shutdown was prompted by increasing O_2 levels (up to 4.5%) within the drywell. The source of oxygen was determined to be an instrument air leak that resulted from a broken air line oiler in the drywell. Operations resumed on October 13, 1970.
December 25, 1970 (Scram No. 50) (No. 14 since start of commercial operation)	An automatic scram from 100% reactor power resulting from turbine stop valve closure. Just prior to the scram, turbine stop valves were being individually tested. When the No. 2 turbine stop valve was tested, all four turbine stop valves closed. Strip charts for key parameters were reviewed by the inspector and the following information was obtained. The pressure peaked at 1120 psig. All 8 APRM's dropped to 0 immediately with no increase from the pre-scram equilibrium value. Turbine bypass valves operated normally and went full open following the closure of

Date	Cause
	the turbine stop valves. The events recorder did not function properly following the scram and no useful information could be obtained from this source. Electromatic relief valves C or D (or both) lifted for a short period during this transient. Reactor vessel level dropped to approximately 8 feet 6 inches above the top of the active fuel (1 foot on the level recorder) and then increased to the full scale value (8 feet on the recorder) momentarily. See Section H.1 for additional details of this event. Following the scram, the turbine stop valves were investigated by the licensee and no cause could be found to explain the closure. The reactor was restarted on the same date and power level increased to approximately 250 MWe where additional turbine stop valve testing was performed. See scram 51 below.
December 25, 1970 (Scram No. 51) (No. 15 since start of commercial operation)	Automatic scram from approximately 45% reactor power due to the closure of the turbine stop valves. Just prior to the scram, the No. 2 turbine stop valve was being tested. This resulted in the closure of all four turbine stop valves, which in turn caused the scram. The reactor remained shut down until December 27, 1970 while a detailed inspection was made of the No. 2 turbine stop valve. It was determined that this valve was closing in approximately 4 - 5 seconds. It was postulated by JC and GE that this closure time resulted in a hydraulic shock wave being established in the turbine control oil system which resulted in the closure of the other three turbine stop valves. The No. 2 turbine stop valve was adjusted to close in 12-17 seconds. The change in the closure time which resulted in the scrams was attributed to a loose lock nut on one of the pilot valves which controls the valve closure speed.
January 25, 1971	Unscheduled reactor shutdown when the identified leakage rate increased to approximately 14 gpm. Investigation by the licensee disclosed that a cooling coil within the drywell equipment drain tank had become eroded and was leaking fresh water coolant into the drain tank. The problem was corrected by reorienting the coil within the tank and weld repairing the cooling coil. Operations resumed on January 27, 1971.

Date	Cause
March 3, 1971	An orderly reactor shutdown resulted when the unidentified leakage rate into the containment drywell reached approximately 4.5 gpm. Investigation by the licensee disclosed a pipe nipple leak in the closed cooling water system supply (outside the tank) to the cooling coil in the drywell equipment drain tank. The nipple leak was repaired and operations resumed on March 4, 1971.

E. Primary System

1. Electromatic Relief Valves

Mr. Riggle informed the inspector that the pilot valves for the electromatic relief valves were inspected in December, 1970 and adjustments made to the stroke of the pilots to preclude the binding experienced at another boiling water reactor. This was considered to be a temporary fix and General Electric Company has provided JC with a field change to permanently correct the problem by changing the pilot internals. Mr. Riggle informed the inspector that this field change will be installed during the planned October, 1971 poison curtain removal outage.

2. Isolation Condenser Isolation Relay Failure

Mr. Riggle informed the inspector that during surveillance testing in March, 1971 one of the four, five-second time delay relays used to isolate the isolation condenser had failed to operate properly. The nature of the failure was the time delay feature was lost due to a bellows failure in the relay which resulted in the relay operating instantaneously. He stated that two of the four relays had been exchanged for a different type (Agastat) and that the other two relays are scheduled to be replaced when replacement parts become available. It was determined that the loss of the single relay would not have affected the normal isolation feature. Mr. McCluskey stated (telecon) that a report of this event would be submitted to DRL within 30 days in accordance with Paragraph 3.C.(2) of the license.

F. Reactivity Control and Core Physics

1. Turbine Scram Trips

The inspector reviewed the modifications made to the plant safety system in early December, 1970 that were required for the mini-stretch power increase. Two new additional scram circuits were added; turbine stop valve closure scram at greater than 45% of power level and generator load rejection (acceleration relay) scram trip at greater than 45% of power level. The installation and testing records did not reflect the pressure setting of the pressure switches mounted on the turbine third stage that affect the 45% bypass provision for these two scram trips. Discussions with Mr. Riggle indicated that the trip settings had been provided verbally by a GE representative at the site and that the instruments had been set at 200 psig (representing approximately 45% of rated power). Calibration records for the switches were reviewed and the last noted trip check of the instruments was performed on December 29, 1970. As found, readings during this check varied from 150 to 205 psig for the four switches. The inspector informed Mr. Riggle that the installation of these pressure switches on the third stage of high pressure turbine did not agree with the information provided in the application, i.e., that the pressure switches would be installed on the first stage of the high pressure turbine. Mr. Riggle stated that sufficient instrument tap-in points were not available to satisfy the separation criteria (for the pressure switches) on the first stage, and therefore, the third stage was selected for installation. He stated that the pressure switch setting (200 psig) was based on the expected third stage pressure that would be experienced at 45% of rated power. The inspector inquired into whether or not the 45% pressure setting had been verified (during turbine startup) following resumption of operation on December 6, 1970. Mr. Riggle stated that it had not.

A review was made of the quarterly surveillance test procedures that are required by the Technical Specifications for these trip circuits. The test procedures did not require that the 45% bypass function pressure switch setting be verified during each check. It was pointed out to Mr. Riggle that as most of the surveillance testing is done at 100% of rated power, that the 45% trip setting would not be verified with the proposed method of testing. This item was also discussed during the exit interview and Mr. McCluskey agreed at that time to perform set point verification for these pressure switches during future quarterly surveillance tests. Also discussed during the exit interview was the concern that JC had set safety system trip points solely on the basis of verbal communications between GE and JC. Mr. McCluskey agreed to provide documentation to the inspector during the next routine inspection verifying the correctness of the 45% pressure switch settings.

2. Local Power Range Monitors (LPRM)

During discussions with Messrs. Ross and Sullivan, the inspector was informed that there are currently 13 failed LPRM's. The outputs from these LPRM's are bypassed from their respective APRM inputs. A review of the locations of the LPRM's disclosed that the facility is operating within the requirements of the Technical Specifications (3.1.B).

Jersey Central submitted change request No. 6 to DRL on March 3, 1971 requesting a change to Technical Specification 3.1.B.2. This change would permit the continuous use of a traveling in-core probe (TIP) chamber as an input to an APRM if all LPRM monitors in a particular radial core position have failed.

Mr. Ross informed the inspector that he considered the cause for the failure of the LPRM's to be due to moisture entering connectors external to the reactor vessel, but within the drywell. Replacement of LPRM's requires the reactor vessel head removal.

3. Control Rod Performance

Control rod scram times were reviewed by the inspector for scrams 49, 50 and 51. Summarized below are the results of the scram times for the 26 selected rods.

Scram No.	Average For 90% Insertion	Maximum Individual Rod	Minimum Individual Rod
49	2.72 seconds	2.92 seconds	2.46 seconds
50	2.79 seconds	3.11 seconds	2.59 seconds
51	2.85 seconds	3.18 seconds	2.60 seconds

The totalized withdrawal stall flow results on March 29, 1971 indicated a total of 255 gpm. The previous monthly totalized stall flow (February, 1970) was 273 gpm. It was reported to the inspector that the reduction resulted from the venting of some drives. Currently, one control rod (42-19) has a stall flow of greater than 5 gpm. Withdrawal stall flow for this rod is 5.8 gpm and it is reported that the rod operates satisfactorily at near normal pressure. It was reported that there are no operating problems with the drives.

4. Reactor Vessel Level Instrumentation

Mr. Riggle was questioned by the inspector as to whether or not OC-1 had experienced any problems with the collection of noncondensable gases in the steam condensing pots used for the reference legs for the GE/MAC reactor vessel level instruments (used in the feedwater control system) and for the low-low-low level instruments (used in the reactor protection system). He informed the inspector that prior to plant turnover from GE to JC, some problems had been experienced and at that time the steam condensing pots were reoriented 90° from a vertical to a horizontal position. He also informed the inspector that no measurements had been taken by JC to determine if the steam condensing pots were free venting, i.e., if the slope of the line from the condensing pot to the point where the instrument line taps into the reactor vessel is positive during reactor operation. Mr. Riggle did inform the inspector that one of the GE/MAC level transmitters was reading low by approximately 1 foot below the indicated Yarway indication.

This issue was discussed with Mr. McCluskey subsequent to the inspection and he stated that a determination will be made as to whether or not the subject steam condensing pots are free venting during reactor operation. This issue will rec /e followup during the next routine inspection.

H. Power Conversion System

1. Turbine Initial Pressure Regulator Performance

Since the last routine inspection on October 13-16, 1970 no additional pressure disturbances have been experienced as a result of malfunctions or oscillations of the turbine initial pressure regulator system.

Scram No. 50 (See Section C) highlighted the possibility of a new transient not previously experienced by the facility. Mr. Carroll informed the inspector that when the turbine stop valves are closed by an electrical signal or on demand from the control room operator, the turbine control valves also close. This results in prompt opening of the turbine bypass valves. Scram No. 50, which resulted from the closure of the turbine stop valves due to hydraulic oil pressure disturbances would not have resulted in the prompt opening of the turbine bypass valves. The effect of a delay in turbine bypass valve opening could not be obtained from recorder charts as the control room operator had promptly initiated (control switch action) tubine stop valve closure when he heard the stop valves shut.

2. Turbine Bypass Valve Linkage Replacement

The linkage failure in the turbine bypass valve control system which was described in Section C.3. of CO Report No. 219/70-6 resulted in the licensee replacing all the aluminum horizontal push-pull bars (four) with carbon steel material. The GE LSTG Division performed a metallurgical analysis of the failure. Mr. McCluskey stated that GE had informed him that the aluminum material used in the bar was not in accordance with design specifications. He also stated that GE was testing an aluminum linkage made with the specified material but no results were available. Mr. McCluskey stated that efforts had been made by JC to obtain a metallurgical report of the failure, but that GE has been reluctant to provide any information on the subject.

I. Auxiliary Systems

1. Air Systems

Discussions were held with Mr. J. Carroll and the inspector informed him of the problems with contaminants in air systems experienced at two other facilities. Mr. Carroll informed the inspector that the instrument air system at OC-1 which serves the majority of the vital equipment is constructed with both carbon steel and copper pipe. He stated that no routine dew point measurements are obtained of the air supply to determine drying tower performance, but the original preoperational test verified that the drying tower was capable of achieving a -25° F dew point. He stated that current surveillance practices include a blowdown of air receivers once per shift and a check by the operators to insure that the drying tower is operating normally and transferring properly. He stated that there was no preventive maintenance schedule for testing the dessicant in the drying towers or inspecting of filters or equipment within the system. Mr. Carroll stated that he will review the surveillance practices in effect with the operations group and additionally review the need to perform preventive maintenance on air system equipment. This item will receive followup inspection.

K. Containment

1. Torus Vacuum Breaker Valves

Discussions were held with Mr. W. Riggle to determine the nature of the problem with the torus vacuum breaker valves and the repairs that were made to the system. Mr. Riggle informed the inspector that during surveillance testing on December 18, 1970, both vacuum breaker valves failed to open. The butterfly valves in question are designed to open on approximately 0.5 negative pressure differential between the torus and the reactor building. The problem was reported to be that the butterfly discs had locked in the seats in the closed position and that the air operators were unable to open the valves. One of the valves was found to have an improperly adjusted linkage which resulted in over travel in the closed direction and the other valve was reported to have a misadjustment of the butterfly stem which caused the binding condition. Both conditions were corrected and the valves successfully retested on that same date.

During surveillance testing on the following day, December 19, 1970, the valves again failed to open and manual assistance was required. Once open, they continued to operate satisfactorily. It was noted that when the valves were left in the closed position for approximately 5 minutes, that the linkage assembly on the operator continued to exert force on the butterfly disc and continued to drive it into the seat. An adjustment was made to the air regulator which positions the valve operator so that when traveling in a closed direction the linkage would not tend to drive beyond the closed position. Subsequent testing on December 20, 21, 22. 23, 24 and 31, 1970 and January 7 and 13, 1971 and in monthly tests thereafter have not disclosed any additional failures. The valves were disassembled in January, 1971 and the rubber liners replaced in both valves as the liners appeared worn.

Following the final linkage adjustment, local leak rate measurements at 35 psig were conducted to determine linkage adjustments which resulted in the minimum valve leakage while still permitting satisfactory block valve operation. These tests determined a total of approximately 18 SCFH valve leakage for both valves. Technical Specifications* limits the total leakage rate through these valves and the containment air purge valves (measured leakage rate 7.98 SCFH) to 111.0 SCFH. The butterfly valves in question are 20 inch Rockwell Model LCW 2434 W, designed for 50 psig.

*Paragraph 4.5.F.c.

2. Drywell Nitrogen System

JC stated in their reply to the form AEC-592 on December 22, 1970, that the target date for the installation of a nitrogen system to supply instrument air needs in the drywell was February, 1971. During the inspection, Mr. Riggle informed the inspector that the drywell N_2 system is 85-90% complete. The final design will incorporate two compressors and an air receiver. The nitrogen supply will be from the liquid nitrogen storage tank and evaporator. The system will supply (through a ball transfer valve) N_2 to the drywell instrument air system. The function of the ball transfer valve is to transfer back to the instrument air supply should the drywell N_2 system fail. Mr. Riggle stated that the necessary parts to complete the system should be at the site in the near future and would be installed at the first outage of sufficient duration.

3. Torus Oxygen Sample Line Isolation Valves

During the conduct of the containment integrated leakage test in October, 1970, a considerable amount of air leakage was observed through the oxygen analyzer located downstream of the torus oxygen sample line isolation valve. The valve was checked and found to be closed, but the flow as measured through the oxygen analyzer with an air flow meter disclosed a leakage of greater than 180 CFH. The subject valve was replaced with a new solenoid valve of different design. The local leakage rate through the new valve was measured and added to the final recorded containment leakage rate.

Mr. Riggle informed the inspector during this inspection that both the torus and the drywell O_2 sample line isolation valves had been replaced. In addition, the General Electric Company is to provide a field change to provide double isolation valves in each line and additionally reroute the exhaust from the O_2 sampler back to the drywell. The current status of the O_2 analyzer is that it exhausts into the ventilation exhaust from the reactor building.

4. Drywell Leak Detection Procedures and Equipment *

The "small leak detection procedure" and equipment in use at the facility were reviewed by the inspector. Also reviewed were the two unscheduled shutdowns that resulted on January 25, 1971 and March 3, 1971 from high identified and unidentified leakage rates.

Discussions were held with Mr. J. Carroll, Operations Supervisor to determine the leak detection procedures in effect at the station. The control room copy of the small leak detection procedure was also reviewed with Mr. Carroll. Pending changes to this procedure, to reflect the Dresden 2 experience, were reported by Mr. Carroll to have been circulated for comment to all concerned parties, but that the final revised procedure had not yet been placed into effect. Mr. Carroll stated that the final revision to the procedure would be put into effect on that date. A copy of the revised procedure was provided to the inspector by Mr. Carroll and

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*Memo J. P. O'Reilly to Regions "Leak Detection Equipment and Procedures" dated March 2, 1971.

a review of same indicated that the procedure appeared adequate. The procedure included specified action points for increasing leakage rates at 5 gpm for unidentified leakage and the 25 gpm for total leakage (both identified and unidentified). If the action points were reached, it specified that the reactor should be shut down to the cold standby condition.

Leak detection equipment for unidentified leakage consists of level probes installed in the drywell sump. These probes, and associated timer, function during sump pump out. If the level between the two measuring probes can not be reduced by pump operation within the specified timer setting, an alarm sounds and the pump continues to empty the sump. The alarm condition can only be reset in the rad waste system control room where the timer can also be read. There is a correlation between the timer setting and sump in-leakage that permits an evaluation to be made of the unidentified leakage rate. After the initial alarm is received it requires approximately 29 minutes (at the 5 gpm leak rate) to refill the sump to re-check the alarm point. The operators obtain pump flow integrator readings at four hour intervals as an additional check on the leakage rate. Twenty-four hour integrator readings are also obtained by [illegible]e engineering group and the operations group to plot the daily aver[illegible]leakage for this source of leakage.

As a result of the ACRS letter to JC on December 12, 1968, JC committed to provide a supplemental and redundant method of leak detection. Station records were reviewed with Mr. D. Ross and disclosed that weekly drywell air samples have been taken for analysis during the past year. Sample collection is achieved through both particulate and charcoal filters and both filters are subjected to a gamma scan. JC has been investigating selected radionuclides identified in this gamma scan in attempts to correlate measured activities with known leakage rates. To date, no meaningful results have been obtained with this method of analysis and Mr. Ross stated that the sample analysis will be continued in attempts to evaluate the method of leak detection.

Identified leakage into the drywell is monitored in a similar manner as described above for unidentified leakage.

A review of station records for the periods preceeding and including the two unscheduled shutdowns on January 25 and March 3, 1971 disclosed that the identified leakage rate increased to approximately 14 gpm on January 25, 1971 before the plant was shut down. The source of this leakage was determined to be a leaking cooling coil inside the drywell equipment drain tank. The coil was repaired by welding and reoriented within the tank. Station records also disclosed that the unidentified leakage rate increased to approximately 4.5 gpm on March 3, 1971. The source of leakage was determined to be a pipe nipple leak in the closed cooling water system supply to the cooling coil within the drywell equipment drain tank. This nipple was repaired and operations resumed on the following date.

L. Emergency Core Cooling Systems

1. Core Spray Nozzle Wall Thickness Determinations

Due to the inspector's previously stated concerns, reinspection of the north core spray nozzle wall thickness (by radiography) was performed by Mr. N. Goodenough of the GPU Quality Assurance Staff during October, 1970. Wall thickness was evaluated at 8 points with the minimum thickness being 0.465 inches. The required (by code) wall thickness was stated to be 0.401 inches. The GORB reviewed the reinspection results during meeting No. 22 on January 15, 1971.

2. Core Spray System Water Hammer

During GORB meeting No. 22 a review of plant piping system vibrations was undertaken. The chairman of the GORB had previously requested information from the site regarding any systems that were known or suspected to have excessive vibration levels. Mr. J. Carroll, Operations Supervisor, generated a list of systems and provided this list to the GORB. Based on a review of this list, the GORB requested the GPU Quality Assurance Group to determine the feasibility of testing all of the welds in the core spray piping which had been subjected to vibration from water hammer. GORB also requested qualitative vibration measurements for all piping systems identified to enable a determination if they are within design limits. Two GORB members were at the site on February 3, 1971 to observe and measure the vibration levels on the core spray systems. A review of a letter generated by these two members that was sent to both the PORC and GORB members on February 8, 1971 disclosed the following information:

a. There is no high point vent on the core spray loop piping associated with pumps A and C. This piping is filled by watching loop pressure in the control room after opening the condensate fill line valve. The other core spray system does have a high point vent.

b. A review of the pipe motion measurements discussed in this letter indicated pipe movements at selected points of up to 6 inches following pump starts.

c. The letter also discussed that during the preoperational program, when severe hammers were noted, a system was added to supply condensate to fill and pressurize the system to 15 psig. It was later determined that this system could not be maintained in constant use due to small leakage past the core spray pump check valves, which allowed the condensate to leak into the torus. This in turn created a water handling problem as the torus at OC-1 is chromated and the State of New Jersey has a 0 release limit for chromates. The system is now used to fill and vent the core spray loops prior to each monthly test.

The chairman of the GORB (two other GORB members also present) was contacted by telephone by the assigned inspector on March 12, 1971. Mr. W. Hurst, GORB chairman, informed the inspector that GORB had requested Burns and Roe to measure and determine the stress levels in the core spray piping. In addition, the GORB has also recommended that the results of this stress analysis be used to determine the need to perform an NDT inspection of piping system materials.

JC has been working with GE on a design change to the system which it is hoped will eliminate the water hammer problems. The design change will incorporate two small pumps (one for each core spray loop) which will take a suction from the torus and keep the core spray piping filled and pressurized at all times. Thus, any leakage back to the torus will not increase the level in the torus. This design change was reviewed by the PORC and has been returned to GE with additional questions. It is anticipated that all questions will be satisfactorily resolved and that the design change will be installed within three months.

During the exit interview conducted on April 13, 1971, the assigned inspector stated that Compliance would closely follow the progress and results of the GORB recommended actions.

3. Core Spray System Weld Defect

Jersey Central provided DRL in letters dated July 31, 1970 and September 30, 1970 information on the nondestructive testing report for selected balance of plant systems outside the primary containment. JC reported*that they were unable to locate a radiograph or acceptance report for one weld located in a 10 inch line at the discharge of a core spray pump. JC planned to perform a radiographic examination of the weld at the earliest convenient time permitted by plant operations. During this inspection, it was determined that on November 11, 1970 the subject weld was re-radiographed and two defective areas were identified. Discussions were held with Messrs. Goodenough, GPU and Kossatz, Maintenance Foreman, and they informed the inspector that the defects consisted of both porosity and excessive "suck-back" in the root weld. Defective areas were repaired by grinding and rewelding and acceptable radiographs of the repairs were obtained on November 11, 1970.

A review of the repair records disclosed that the radiographic inspection had been performed by Conam. The NDT personnel were qualified in accordance with SNT-TC-1A requirements. The weld repair procedure, dated November 10, 1970 was also reported to be used. The welding procedure used was qualified (in 1969) by the welder performing the repair. The finished repair weld was LP examined using an approved GPU LP procedure. Mr. Kossatz stated that the film was interpreted using USAS B31.7 - 1969 requirements.

*Letter to DRL from JC dated September 30, 1970.

N. Emergency Power

1. Diesel Generator Annual Inspection

Station records disclosed that both diesel generators were given a thorough inspection as required by Technical Specifications (4.7.A.3) on November 12-13, 1970. Present during the inspection was the General Motors representative. The inspection report from the GM representative to JC was reviewed and the only problem areas noted were that new starting solenoids were installed on both the No. 1 and No. 2 diesel generator's starters. This item is considered resolved. (See Management Interview item 4 for discussion on diesel generator shutdown devices).

2. 125 Volt Station Batteries

The review was made by the inspector of surveillance testing records for the A and B 125 volt station batteries. Review disclosed that monthly checks are now being performed to determine specific gravities and the temperature of every fifth cell, the voltage of every cell, the water level of every cell and the amount of water added. These tests are now considered to satisfy the Technical Specification (4.7.B) requirements.

Discharge load tests were noted to be conducted for both the A and B batteries in March, 1971 with acceptable measured ampere-hour capacity. One battery cell in the B battery was replaced on January 8, 1971 as a result of a lower than recommended cell voltage which was noted at the termination of the discharge load test conducted in September, 1970.

The inspector noted during the review of the load discharge testing records for March, 1971 that the B battery terminal voltage had been reduced to less than 105 volts. The 105 volts were previously stated by Mr. Riggle to be an acceptance criterion for the test. Mr. Riggle stated that he doubted the accuracy of the recorded terminal voltage as the summation of individual cell voltages would indicate a higher terminal voltage (at the completion of the load discharge test). He stated that this aspect of the test would be reviewed and resolved. Subsequently, Mr. McCluskey informed the inspector (telecon on May 11, 1971) that this issue had been resolved by cleaning cell interconnections and successfully load testing the B battery.

3. Diesel Generator Starting Batteries

A review of the surveillance records required by the Technical Specification (4.7.A.5) for these batteries indicate that all the required tests are now being performed. A review of the load discharge test records conducted on December 4, 1970 indicated that the batteries had a measured capacity in excess of the 240 ampere-hour requirement that was stated by Mr. Riggle to be the acceptance criterion for the batteries.

4. Reportability of Past Diesel Generator Problems

Mr. McCluskey was informed that the past instances of failures of the diesel generators to start on the first attempt were considered reportable by Compliance (3.C.(2) of license). He stated that a written report of these events will be submitted to DRL in the near future.

5. Diesel Generator Shutdown Devices

The inspector noted in a review of the diesel generator annual inspection records (See N.1. above) that no checks had been made to determine the accuracy of the set points for devices which can shut down the diesel generator. This issue was discussed with Messrs. Riggle and McCluskey and they stated that these devices will be tested during future annual inspections.

Q. Radioactive Waste Systems

1. Liquid Effluents

See Attachment No. 3.

2. Independent Analysis of Liquid Radioactive Sample

As a result of the discrepancies noted in Attachment No. 3, a sample of liquid waste was split with OC-1 for comparison analysis. Independent analysis of the CO sample was performed by the New York Operations Office (NYOO) and the Idaho Operations Office (IOO). The results of the analyses for this sample are discussed below:

a. Gross Beta Analysis (uCi/ml)

OC-1	NYOO	IOO
7.46×10^{-6} (1)	3.3×10^{-5} (4)	1.3×10^{-5} (5)
2.12×10^{-5} (2)		3.6×10^{-5} (6)
6.86×10^{-6} (3)		3.5×10^{-5} (7)

(1) Method in use prior to inspection. Counted with GM tube using ST-90/YT-90 as calibration source. Counted 1505 hours on April 9, 1971.

(2) Counted with an internal proportional counter using thallium-204 as a calibration source. Counted 1330 hours on April 9, 1971. (See Q.4. below)

(3) Counted with an internal proportional counter using thallium-204 as a calibration source. Counted 1600 hours on April 12, 1971. (See Q.4. below)

(4) Counted by liquid scintillation technique using ST-90/YT-90 as a calibration source. Counted on April 9, 1971.

(5) Counted with an internal proportional counter using Cs-137 as a calibration source. Counted on April 14, 1971.

(6) Counted by liquid scintillation technique using Cs-137 as a calibration source. Counted on April 14, 1971.

(7) Counted by Cerenkov radiation detector using Cs-137 as a calibration source. Counted on April 14, 1971.

The gross beta analysis performed by OC-1 resulted in a difference (between old and new method) of ≈ 3 which indicated that release concentrations which were determined by the "old" analysis were low by a factor of 3. In addition, the "new" method of gross beta analysis is still a factor of 3 lower than the concentration determined by isotopic analysis. (See Q.3.b. below). These results support Mr. Pomeroy's findings as discussed in Attachment No. 3.

b. Isotopic Analysis (uCi/ml)

Results were corrected to the time of sampling for decay.

Isotope	OC-1 (2)	NYOO (1)	IOO (1) (4)
Ba-140	1.7×10^{-6}	4.7×10^{-7}	$< 6.9 \times 10^{-7}$
La-140	1.7×10^{-6}	1.9×10^{-6}	$< 5.5 \times 10^{-6}$
Co-60	1.5×10^{-6}	1.3×10^{-6}	2×10^{-6}
Co-58	1.3×10^{-6}	1.1×10^{-6}	1×10^{-6}
Mn-54	1.2×10^{-7}	2.2×10^{-7}	$< 5 \times 10^{-7}$
Fe-59	4.9×10^{-7}	8.6×10^{-8}	-
I-131	1.6×10^{-6}	1.7×10^{-6}	1.7×10^{-6}
I-133	$< 1.4 \times 10^{-6}$	1.8×10^{-7}	-
Np-239	1.1×10^{-5}	7.7×10^{-7}	-
Cr-51	1.2×10^{-5}	1.2×10^{-5}	1.2×10^{-5}
Zn-65	-	2.5×10^{-7}	6×10^{-7}
Tc-99m	2.3×10^{-6}	2.8×10^{-6}	-
Mo-99	$< 2.3 \times 10^{-6}$	2.5×10^{-6}	-
Ce-141	-	8.6×10^{-7}	4.4×10^{-6}
Xe-133	1.4×10^{-5} (3)	9.9×10^{-6}	-
Xe-135	1.8×10^{-5} (3)	4.0×10^{-6}	-
TOTAL	6.9×10^{-5}	4.0×10^{-5}	2.9×10^{-5}

(1) As the exact time of analysis was not available, a 24-hour correction for decay was applied to the NYOO results. Similarly, a six-day correction for decay (sample counted April 14, 1971) was applied to the reported IOO results. Both NYOO and IOO counted a liquid sample using a GE (Li) detector.

(2) Sample counted after evaporating to dryness using a NaI crystal.

(3) Sample counted in liquid form.

(4) IOO reported Hg-203 at a concentration of 0.5×10^{-6} uCi/ml and Ce-143 at a concentration of 4.0×10^{-5} uCi/ml. NYOO and IOO were recontacted to request that they recheck their analyses in an attempt to resolve this discrepancy. No resolution was obtained.

3. Gaseous Radioactive Effluents

Discussions were held with Messrs. Ross and Pelrine to determine the licensee's method for analyzing and reporting the stack release rate. The following information was obtained:

a. The concentration of six radionuclides are analyzed for in grab samples which are obtained on a weekly frequency from upstream of the steam jet air ejector off-gas delay line. The six radionuclides include Xe-138, Kr-85, Kr-88, Kr-85m, Xe-135 and Xe-133.

b. The off-gas delay time is computed by weighing a sample of off-gas before and after spark testing. As the test is done, submerged in water, the spark test results in the recombination of the major constituents (O_2 and H_2) of radiolytic gas. A major fraction of the remaining sample is assumed to be due to air in-leakage into the turbine main condenser. Typical delay time measurements as determined by this method were approximately 62 minutes.

c. Flow through the off-gas system is recorded on three ΔP instruments in the control room. This source of measurement is not used however, in determining the off-gas flow rate. A comparison of this method with the method described in b. above indicated 71 cfm through the three ΔP instruments vs 105 cfm as determined from the spark test. The use of the 105 cfm value would result in the shortest holdup time determination.

d. The stack gas monitor is calibrated each week. The calibration is performed using the results of the grab sample analysis which is corrected for delay to determine the stack activity. OC-1 uses a standard 1.9×10^5 cfm dilution flow in the stack. A counts per second/microcurie per second correction factor is determined for the stack gas monitor on a weekly basis.

e. Results reported in the semi-annual report to the AEC are derived from weekly "eyeball" integrations of the stack gas recorders.

f. Trip functions for the stack gas monitor were currently set at 300 cps, which corresponds to approximately 0.03 Ci/sec. This was noted to be one decade below the annual average release rate specified in Technical Specification 3.6.A.1.a. The second high-high alarm is set at 1,000 cps which corresponds to approximately 0.1 Ci/sec, which is one decade below the 48-hour release rate specified in Technical Specification 3.6.A.1.b.

g. Off-gas monitor alarm set points are set to trip at approximately 0.3 Ci/sec for a high-high alarm and 0.03 Ci/sec on the high alarm.

Mr. Ross was questioned as to why OC-1 did not utilize the GE recommended practice of measuring the activity for the sum of the six gases analyzed and converting this (by the GE-supplied computer code printout) to the total activity contained in 22 gases which are present. Mr. Ross did not indicate an awareness of this GE code, but Mr. Pelrine stated that he was aware of the code and additionally provided a copy of the code. In the inspector's review of information, it appeared that a 10-15% (low) deviation would result from measuring only the six gases for the current values of holdup time. The inspector was informed (telecon May 11, 1971) following the inspection that the code conversion to determine the activity of 22 gases would be used in the future.

4. OC-1 Calibration Error - Gross Beta Counting

The calibration error discussed in Attachment No. 3, item A, was reported to the inspector by Mr. D. Ross, to be corrected on April 8, 1971. The new calibration source (standard) for gross beta analysis was reported to involve the use of thallium-204 deposited on a 2-inch diameter planchet. Gross beta analysis will be performed in the future at OC-1 by use of an internal proportional counter. The counting efficiency of this counter has been measured at 56%.

5. Carbon-14*

Mr. Ross informed the inspector that a sample of liquid radioactive waste was sent to Isotopes, Inc. in February, 1971 for C-14 analysis. No results have been reported back on this sample. Mr. Ross also stated that a sample will be sent to Radiation Management Corporation for analysis and OC-1 will attempt an analysis with their liquid scintillator. The results of the analysis will be reviewed during the next routine inspection.

6. Control of Curie Content in Outside Storage Tanks (Item of Noncompliance)

Section C of Attachment No. 3 discusses an apparent item of noncompliance with Technical Specification 3.6.C. which states that "the maximum amount of radioactive liquids in storage waste sample tanks, the floor drain sample tanks, and the waste surge tank shall not exceed 0.7 curies". A gross beta analysis (which has been established as being in error by at least a factor of 6 from the total activity) performed on March 10, 1971 indicated 0.316 Ci. Scaling up this analysis result by the factor of 6 would yield $\approx$ 1.8 Ci vs the allowable 0.7 Ci limit.

Discussions were held at exit interviews on April 8 and 13, 1971 and Messrs. McCluskey and Ross stated that the Ci content of these tanks will be controlled in the future on a gamma spectrum analysis.

*CO Report No. 219/70-7, Other Significant Items No. 10.

S. Experiments and Tests

Mini-Stretch Power Testing - Meeting minutes indicated that the PORC reviewed and approved the test procedures that were used for the power level increase from 1600 to 1690 MWt* during their meeting on November 20, 1970. It was reported by Mr. D. Ross that the test procedures had been developed by Mr. D. Hetrick's group at GPU. This group was reported to have also reviewed the test results and provided test result summaries. Mr. Ross stated that all test results had met the acceptance criteria.

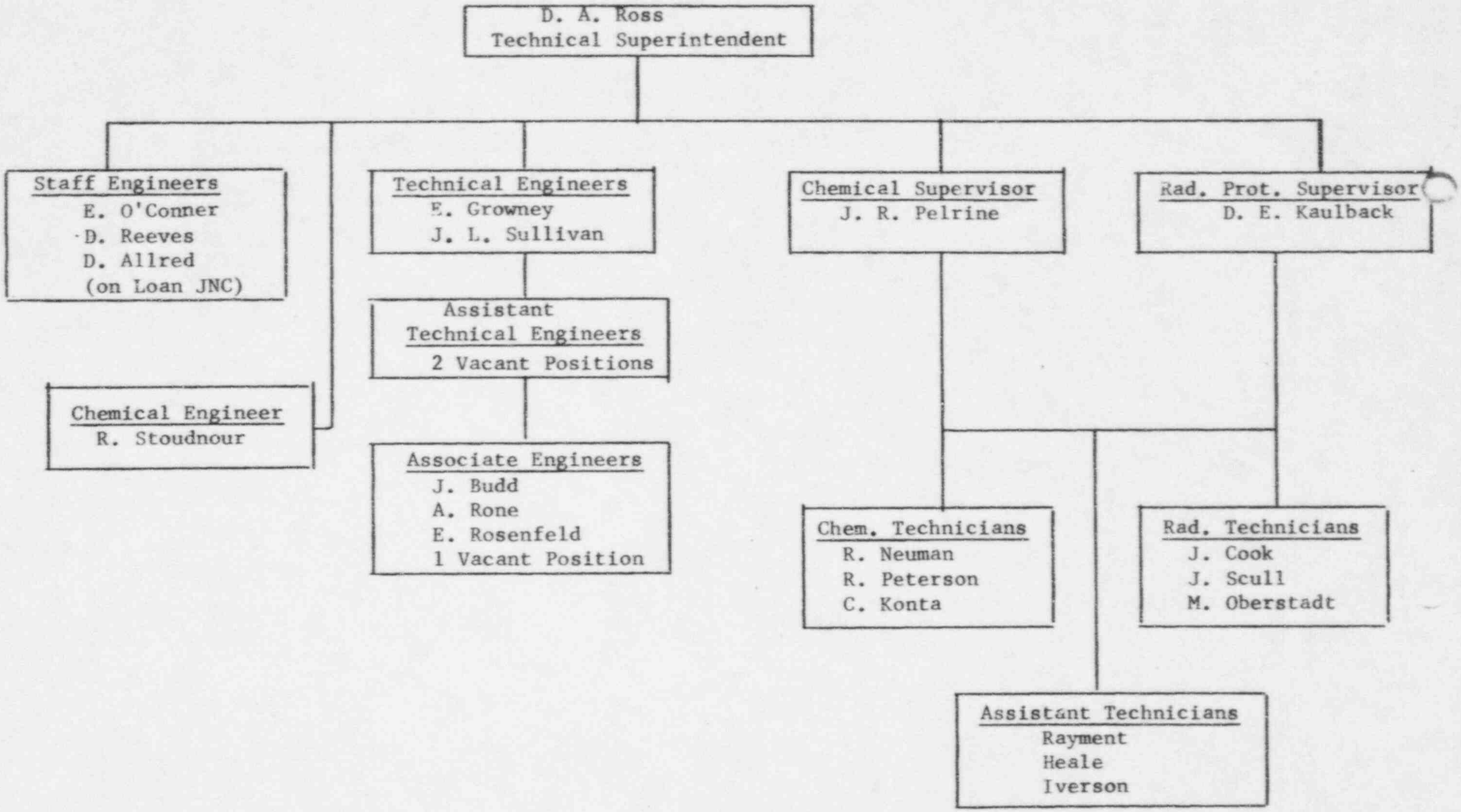
U. Miscellaneous

Aircraft Overflights - Mr. McCluskey informed the inspector that he had no knowledge of any scheduled military or commercial aircraft overflights at the OC-1 facility.

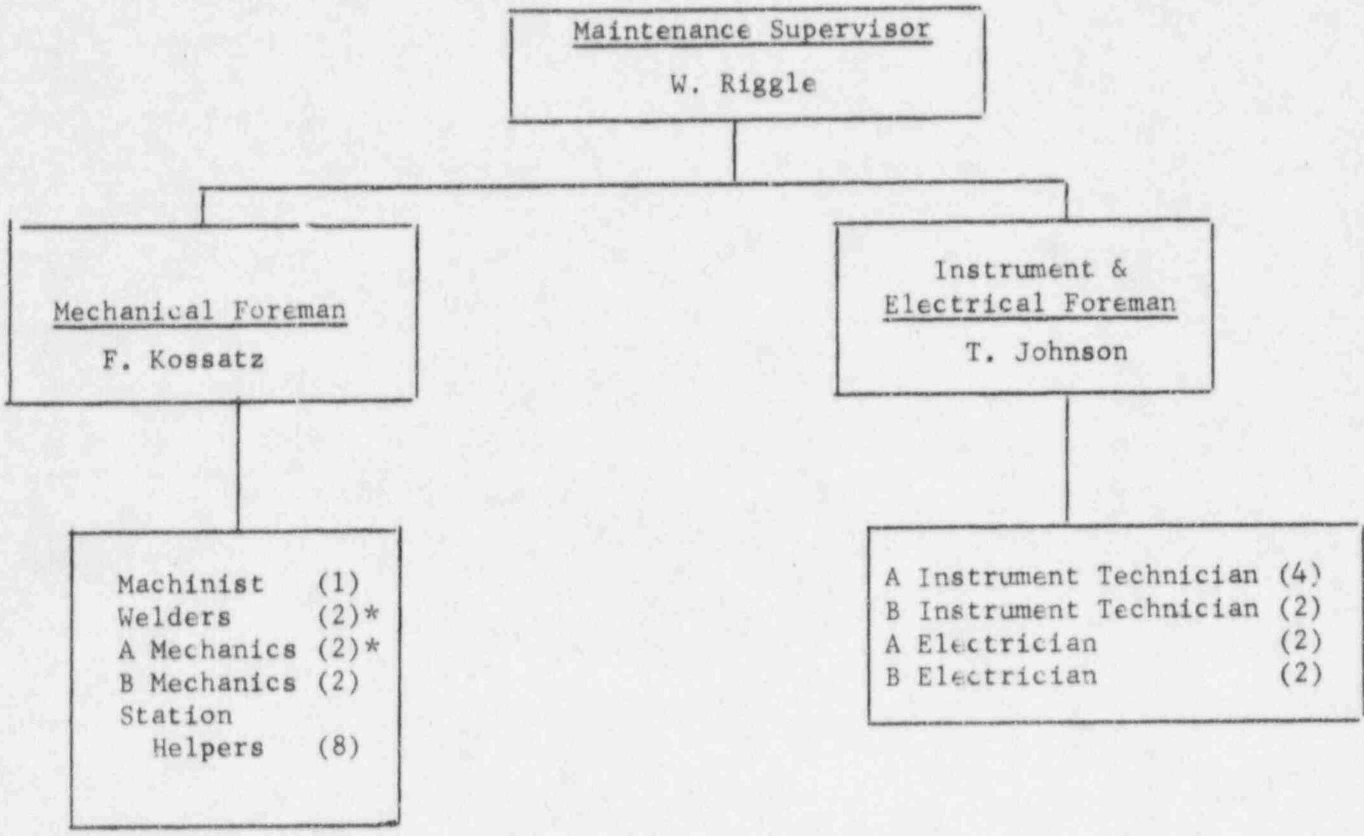
*Described in Amendment No. 55 to the license.

JERSEY CENTRAL POWER & LIGHT CO.
CO Report No. 219/71-1
Attachment No. 1

TECHNICAL GROUP



PRESENT MAINTENANCE ORGANIZATION



* One additional person planned to be added in each category.

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

APR 23 1971

R. T. Carlson, Senior Reactor Inspector, CO:I
THRU: H. R. Denton, Chief, Technical Support Branch, CO:HQ

ASSIST INSPECTION - OYSTER CREEK LIQUID WASTE SYSTEM, DOCKET NO. 50-219

Introduction

This memo contains the results of an assist inspection conducted by Mr. D. Pomeroy, CO:HQ, on April 6-8, 1971, of the liquid waste system at Oyster Creek. Dr. C. Pelletier, CO:HQ, accompanied the inspector for training. The assigned inspector, Mr. R. McDermott, participated in the discussions of the noncompliance items with the licensee's management.

Scope

The inspection followed the procedures in PI-3300/1 for liquid waste systems with particular emphasis on Section 3315.04.C which required a determination of the extent to which waste treatment facilities are being utilized. Operations for 1970 and for the first three months of 1971 were reviewed. Within these areas the inspection consisted of selected examination of representative records, interviews with plant personnel, and observations by the inspector.

Principal Results

A. Monitoring of Radioactivity in Liquids

The licensee discharges liquids in batches and each batch is sampled and analyzed prior to release to the discharge canal. The analysis consists of the evaporation to dryness of a small quantity of the waste sample on a 2-inch diameter planchet and the subsequent counting of the planchet with a 1-1/4-inch diameter thin window G-M tube (end window or gross beta counter). The counter has been calibrated and found to have a 10 percent efficiency with a 1-1/4-inch diameter strontium 90 disc source. This counting efficiency is used for the 2-inch diameter planchets even though more than half the area of the planchet is not under the detector.

In order to determine the adequacy of gross-beta counting an examination of the records of a complete gamma analysis was made for the batch whose analysis had been summarized in the licensee's third semiannual report. The activity in the batch analyzed in the semi-annual report totaled 9.6×10^{-4} μCi/ml. The release records showed that the concentration for this batch determined by the routine method was only 1.41×10^{-4} μCi/ml, indicating a γ/β of 6.8. Mr. D. Ross, the technical supervisor, stated that the reported gamma analysis was the only one performed during 1970 and that they were now performing isotopic analyses about twice a month. An examination of the records of gamma analyses conducted during 1971 was made. For those analyses where a gross-beta count was also available a comparison is made in table I which indicates an average γ/β of over 6.

Mr. Ross also stated that a check of the end window counter with thallium 204 on a 2-inch diameter planchet yielded an efficiency of about 4 to 5 percent.[1/] This result indicates that Oyster Creek gross-beta analysis is low by a factor of 2 to 2-1/2. Results of the licensee's analysis of waste water on their internal proportional counter indicate that their end window gross-beta analysis produces values that are low by about a factor of 3. The results from a sample the licensee split with Isotopes Incorporated indicates that the licensee's routine gross-beta analyses are low by a factor of 5 to 7.

At the time of the inspection the licensee was still controlling and formally recording the release and storage of liquid wastes based on the end window gross-beta analysis at a 10 percent efficiency. It appears that the licensee's surveys conducted to evaluate the radiation hazards incident to the presence and release of radioactive materials are inadequate and, therefore, in non-compliance with Title 10 CFR Part 20.201 (Surveys).

B. <u>Control of Radioactivity in Liquids Discharged</u>

An estimate of the actual concentrations and amounts of radioactivity at Oyster Creek can be made considering the monitoring errors reported above. These estimates may then be compared with recent data from Nine Mile Point.[2/] The Oyster Creek results indicate that their gross beta, when compared to total activity, is low by about a factor of ~7. The results also indicate that beta counting errors above account for about a factor of 2.5. The factor of 3 remaining is then presumed to be the gamma activity not counted by the gross beta detector or true γ/β ratio. The results from Nine Mile Point indicate a factor

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1/ Nine Mile Point uses a 4% efficiency for their end window counter by calibration with a 2-inch diameter thallium 204 source.

2/ Memo Pomeroy to Carlson March 27, 1971, Assist Inspection - Nine Mile Point.

of 2.5 more beta than Oyster Creek (4% Vs 10% counting efficiency) and the average γ/β ratio for the past 14 months at Nine Mile Point has been about 3. This confirmation of the limited Oyster Creek data indicates that the overall factor 7 is reasonable.

Using this factor the 4.5 gross beta curies reported by the licensee[3/] (and shown in table I) for 1970 would be 31.5 total curies. The Technical Specifications (3.6.B.1.a) state that the annual average concentration of unidentified radionuclides in the discharge canal shall not exceed 1×10^{-7} µCi/cc. The basis for the specification considers conditions of minimum flushing in the bay and reconcentration in biota. The basis defines the specification to mean a limit of "0.067 curies per day or a yearly limit of about 25 curies." On the gross beta basis the licensee's report indicates an annual average concentration of 0.166×10^{-7} µCi/ml, the factor of seven to account for total curies would indicate concentration of 1.16×10^{-7}. The licensee, therefore, appears to be in noncompliance with the Technical Specification requiring that the discharge canal not exceed 1×10^{-7} µCi/cc on an annual average basis.

It should be noted that if the Technical Specifications required only control on gross beta (e.g., Nine Mile Point and Dresden) the licensee's report would be low by only the gross beta error factor of 2.5, yielding 11 curies and 42% of the annual limit. It should also be noted that if isotopic identification and mixture MPC's were allowed by the Technical Specifications, the one sample for 1970 indicates that I^{131} is controlling and that the mixture MPC is about 18 times larger than the 1×10^{-7} µc/ml required. On this basis the curies released are still 31.5 but the percent of annual limit is only ∿ 6%.

C. <u>Control of Radioactivity in Stored Liquids</u>

Technical Specification 3.6.c limits the radioactivity stored in the five outside tanks to 0.7 curies. An examination of the surveillance records indicated that the required sampling and analyses were being performed (required by Tech Spec 4.6D - every 72 hours). The limit of 0.7 curies is considered by the licensee to include tritium. The basis for this specification indicates that only the unidentified

3/ Semiannual Report No. 3

radionuclides were considered, and it is, therefore, not necessary to include tritium in this calculation. The review of recent sampling indicated that most results were in the 0.1 to 0.5 curie range, including tritium. The higher numbers usually indicate that the 100,000 gallon surge tank is being used. The highest recent result is listed below:

Date	Gross Beta	Tritium	Total
3/10/71	0.316 Ci	0.347	0.663

The gross beta is determined by using the same methods and is subject to the errors discussed above. Applying a correction factor of 7 to the gross beta analysis would result in greater than 0.7 Ci of activity stored in the outside tanks, even if tritium was excluded. The licensee, therefore, appears to be in noncompliance with the 0.7 curie limit for storage of radioactive liquids.

D. Continuous Liquid Monitor

An examination was made of the calibration records for the continuous liquid monitor. Also examined were plant operating procedures for liquid effluent releases and records of actual releases. From these it appeared that a normal release of 1 x 10^{-3} μCi/ml water would be made at 10 gpm and that the continuous liquid monitor would be expected to read about 2,000 counts per second above background. Mr. Ross stated that considerable trouble had been experienced with the background of this instrument and that the background was currently 14,000 counts per second and that the alarm was set at 100,000 cps. Mr. Ross stated that the high background was caused by a buildup of contamination in the spool of pipe that the detector is mounted on and that a replacement spool of stainless steel is being fabricated. The instrument has only been calibrated to 5,000 cps. Using this calibration, an estimate of the concentration required to trip the alarm at 100,000 cps of about 1 x 10^{-1} μCi/cc appears reasonable. This concentration is higher than the current primary coolant activity and with this setting and the current volumes of waste, the facility could release at the rate of 400 curies per year without tripping the alarm. Mr. Ross agreed with the inspector's statement that the instrument appeared to be out of service. Technical Specification 3.6.B.2 states that "radioactive liquid effluents being released into the discharge canal shall be continuously monitored." Therefore it appears that the licensee has been in noncompliance with this Technical Specification in that the high background, setpoint problem has rendered the continuous in-line monitor incapable of performing its intended service.

E. System Performance

A review of the liquid radwaste system design objectives was not made because the licensee has been unsuccessful in determining what the objectives are. The process design drawings usually available for review[4/] at the site apparently do not exist for the Oyster Creek station. A review of the expected performance of the radwaste system as outlined in the FD and SAR was made and compared with the system performance. The following substantial variances were observed:

1. Waste Collection System

Section IX - 4 Paragraph 4.2 of the FD and SAR states that waste collection drains are "essentially always reused as condensate." During 1970 less than 10% of waste collection drains were reused as condensate. The majority of this water was released to the circulating water system. Mr. Ross stated that a chemical engineer (Mr. B. Stoudnour) was added to the plant's staff in December 1970 and that his first priority was to make condensate from all waste collection waters. Table II indicates that approximately 50% of this water is now being recovered.

2. Floor Drain Activity

Table IX - 4-2 of the FD and SAR lists the expected concentrations of radioactivity in the floor drain system as a function of fuel performance. Interpolation in this table[5/] for an offgas release rate of 10,000 microcuries per second (which is typical of station performance for December 1970 to March 1971) indicates that the expected floor drain concentration is 0.5×10^{-4} µCi/ml. Actual concentrations range from 10 to 50×10^{-4} µCi/ml or 20 to 100 times the expected value.

3. Chemical Waste System

Section IX 3.2, paragraph 3.1.3.3 of the FD and SAR states that the chemical waste will be sampled and concentrated as required. Mr. Ross stated that these tanks had never been sampled and that during 1970 essentially no concentrating of these wastes were performed. The wastes were released to the circulating water

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4/ Memo Pomeroy to Thornburg, Dec. 4, 1970. Assist Inspection Dresden 2.

5/ As shown on Figure 1.

system through the floor drain system. Mr. Ross stated that the evaporator is now operational and the chemical wastes are being concentrated and that it is planned to use the evaporator to the fullest extent practical by concentrating floor drains during periods when it is not being used for chemical wastes.

4. Isotopic Distribution

Table IX - 4-3 of the FD and SAR lists the concentrations of various isotopes expected in the liquid effluents for operation with and without fuel failures. The expected iodine 131 for operation with failed fuels (100,000 μCi/sec stack gas) is listed as 0.03%. Iodine 131 is not expected in the absence of fuel failures. The Oyster Creek semiannual report for the period ending December 31, 1970 lists the isotopic composition of a floor drained sample and indicates that iodine 131 comprises approximately 16% of the activity in this batch. Other isotopes are also a substantial variance from the expected values.

The licensee appears to be in noncompliance with section 3.C(2) of the facility license in that he has not reported the above four substantial variances disclosed by operation of the facility from the performance specification contained in the FD and SAR.

Exit Interview

During the exit interview at the conclusion of the inspection Mr. McCluskey was informed that the abnormal performance of the radwaste system was considered reportable under the license. He stated that they were not in a position to challenge the alleged variances and that they will consider reporting them.

Mr. McCluskey was informed that we consider the continuous liquid effluent monitor to be out of service and that liquid releases without the monitor to be in noncompliance with the Technical Specifications. Mr. McCluskey stated that a fix was underway which would reduce the background and restore the instrument to an operable condition. Mr. McCluskey phoned the instrument shop during the interview and reported that the installation of the stainless steel spool had been completed and that the system was being hydroed and that the instrument would be operational today.

Mr. McCluskey was informed that we consider the "surveys" (that is, the counting methods) currently being used to control the release and presence of radioactive material in liquids to be inadequate and that a review of the station records indicate that the current procedures are underestimating the activity released and stored by about a factor of 7. He was also informed that if the factor of 7 is the appropriate correction the annual average limit for liquid released during 1970 was exceeded; the current release rate indicates that the 1971 level will also be exceeded and that the curie limit for the outside tanks has been exceeded on a number of occasions. Mr. McCluskey responded by saying that they are aware of certain inadequacies in their counting techniques and that they are not surprised that we found this problem. He stated that he believed that they have made an honest mistake; that it was not recognized at the time of the December 16 sample and not fully realized until February and that they are taking action to correct the problem but they are not ready to change the basic system at this time. He outlined the progress on the problem by stating that changes had been made in the gross-beta system to reduce the loss of iodine by adding silver nitrate to the solution before drying and by drying of a smaller sample volume to reduce coincident counting errors. He also stated that their bi-monthly isotopic analysis, their internal proportional counting and the sample sent to Isotopes Incorporated were all demonstrations of their effort to solve the problem and obtain a conservative method for controlling radioactivity in liquids. He stated that they were not prepared to change the method yet and offered no time schedule. No commitment was made concerning the correction of the previously reported data.

D. L. Pomeroy

D. L. Pomeroy
Nuclear Engineer
Technical Support Branch, CO

cc: A. Giambusso, CO
R. H. Engelken, CO
L. Kornblith, CO
J. P. O'Reilly, CO
N. C. Moseley, CO:I
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J. W. Flora, CO:IV
G. S. Spencer, CO:V

TABLE I

Floor Drain Tank Samples

Gross Beta Vs Total Activity

Date	Gross-Beta[1] µCi/ml	Total Activity µCi/ml	Difference
12/16/70	1.41×10^{-4}	9.6×10^{-4} (2)	6.8
1/26/71	4.04×10^{-4}	17.4×10^{-4}	4.3
2/6/71	1.98×10^{-4}	22.5×10^{-4}	11.3
3/18/71	16.5×10^{-4}	52.7×10^{-4}	3.2

(1) Based on 10% counting efficiency

(2) Reported in semiannual report No. 3

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OYSTER CREEK LIQUID WASTE SYSTEM

TABLE II

1970	Gross Beta Curies	Waste Collection System Gallons x 10^3 Input	Dumped	Floor Drain System Gallons x 10^3 Dumped
Jan	0.283	560	All	763
Feb	0.424	460	"	574
Mar	0.196	690	520	729
Apr	0.185	600	All	448
May	0.060	480	"	492
June	0.613	700	"	418
July	0.721	680	"	509
Aug	0.461	860	"	646
Sept	0.353	780	"	523
Oct	0.392	800	"	438
Nov	0.495	827	740	406
Dec	0.320	980	340	333
Total	4.5			6,300

1971				
Jan	0.354	900	120	472
Feb	0.518	940	518	245
Mar	0.952	900	680	285

TABLE III

Primary Coolant Activity - µCi/ml

Date	Gross Beta 10^{-3}	Total Iodine 10^{-3}	Tritium 10^{-4}
1970			
Jan.	6	5	6
Feb.	7	5	3
Mar.	9	8	5
Apr.	8	7	5
May	.03-12	0.06-17	2-4
June	9	15	4
July	10	20	5
Aug.	10	25	5
Sept.	12	25	5
Oct.	14	35	5
Nov.	22	40	6
Dec.	22	40	7
1971			
Jan.	25	50	9
Feb.	50*	60	9

*Method change - reduced volume of sample to reduce coincidence losses.

TABLE IV

Sample Comparison
Gross Beta $\mu Ci/ml \times 10^{-3}$

Sample	O. C. Normal	O. C. IPC	Isotopes Inc.
A	1.27	4.40	8.88
B	1.26	4.72	6.91

Notes:

Oyster Creek Normal - Sr^{90} - 10% eff.
Oyster Creek IPC - Internal proportional counter

TABLE V

Estimate of Liquid Releases

Date	Reported Curies	Correction factor	Est. Actual Ci
1970	4.5	6.8	31
Jan '71	0.35	4.3	1.5
Feb '71	0.52	11.3	5.8
Mar '71	0.95	3.2	3.0

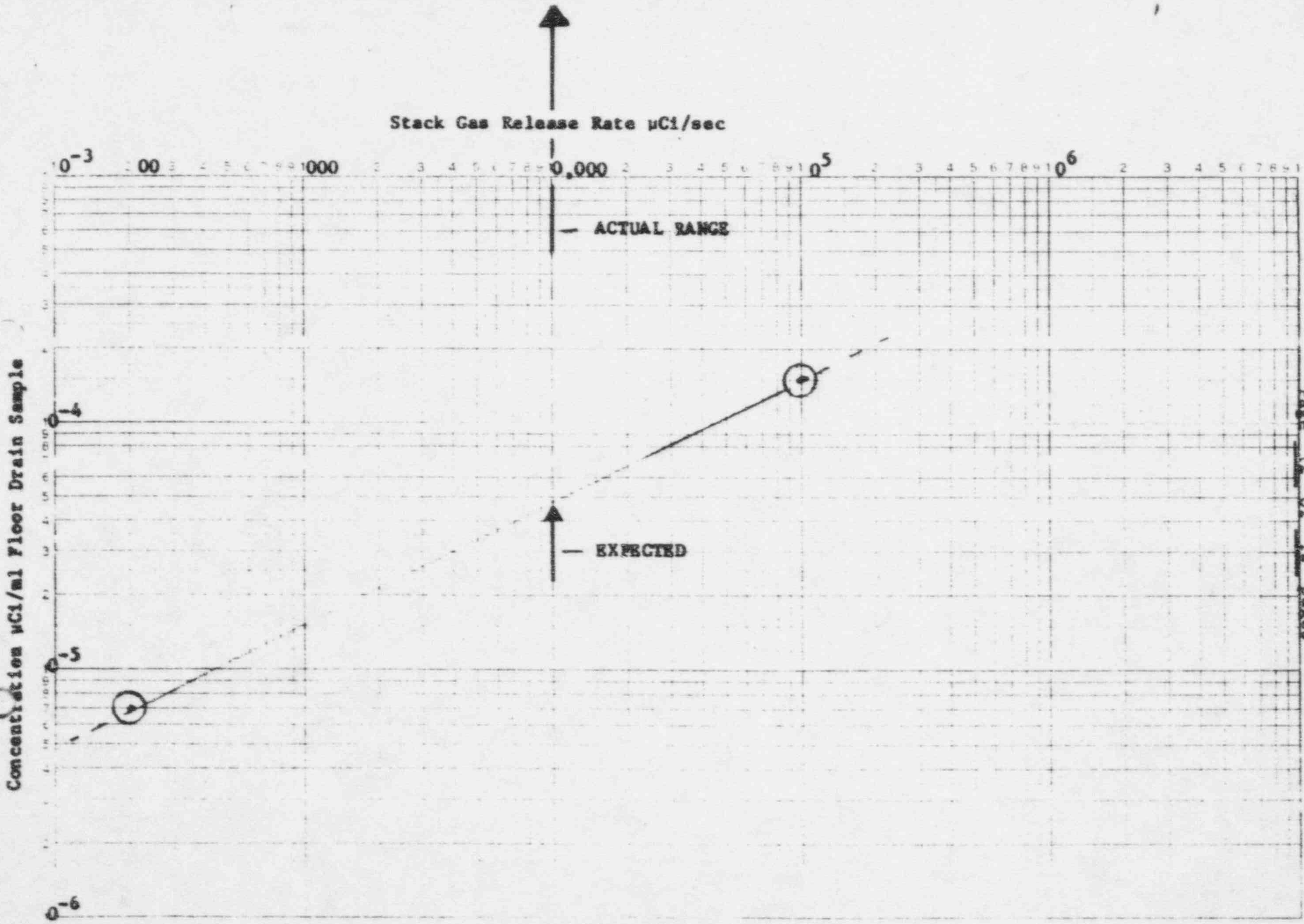

Figure 1 Floor Drain Concentration Vs Stack Gas Rate

OYSTER CREEK LIQUID WASTE SYSTEM

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JERSEY CENTRAL POWER & LIGHT CO.
CO Report No. 219/71-1
Attachment No. 3

Jersey Central Power & Light Company
(Oyster Creek 1)
(Commercial Operation - Dec. 1969)

Scram No.	Date of Scram	Power Level	Primary Cause	Comments
1	12/31/69	1600 MWt	Equipment Failure	Low water level trip resulting from malfunction of three element controllers on feedwater flow caused by stuck excess flow check valve on steam line sensor.
2	1/ 3 /70	1420 MWt	Equipment Failure	Condenser low vacuum trip caused by in-leakage through the expansion joint between "C" condenser and the exhaust section of the "C" low pressure turbine.
3	1/ 9 /70	1600 MWt	Equipment Failure	Low water level trip resulting from a loss of the steam flow signal to the feedwater control system.
4	2/15/70	1350 MWt	Design Inadequacy	Following a load reduction, high level in the moisture separator drain tank cause a turbine trip and a resulting reactor high neutron flux scram.
5	2/18/70	160 MWt	Operator Error	High neutron flux in the IRM's while holding the reactor in hot standby.
6	4/ 7 /70	1600 MWt	Operator Error	Main steam isolation valves closed due to a steam line low pressure trip signal. While making a trip test of the spare exciter breaker, a short tripping three (3) recirculating pumps occurred lowering the D.C. control voltage. The slow closure of the valves caused the pressure to momentarily drop below 850 psig, thereby causing the low steam pressure signal.
7	4/14/70	1600 MWt	Equipment Failure	Low water level trip resulting from EPR pressure oscillations.
8	4/19/70	160 MWt	Operator Error	Neutron monitoring scram in the IRM-APRM overlap range while shutting down plant.
9	6/17/70	1600 MWt	Equipment Failure	Main steam isolation valves closed after receiving a main steam line break signal due to high ambient tunnel temperatures.

Oyster Creek 1 (Cont'd)

Scram No.	Date of Scram	Power Level	Primary Cause	Comments
10	7/11/70	960 MWt	Natural Phenomonom	Turbine load was reduced manually to 200 MWe and a turbine trip manually initiated. Reactor scram caused by high pressure following turbine trip. Heavy grass at intake structure required load reduction and turbine trip.
11	9/17/70	1200 MWt	Equipment Failure	Power oscillations occurred at 1600 MWt, reactor power was reduced to 1200 MWt by reducing recirculation flow. A turbine trip occurred caused by high level in moisture separator drain tank. The reactor scrammed on high flux.
12	9/22/70	1520 MWt	Equipment Failure	EPR malfunction caused low main steam line pressure. Reactor scrammed on MSIV closure.
13	10/ 2 /70	1000 MWt	Design Inadequacy	Electrical load was reduced to 290 MWe to repair condenser leak. Turbine tripped on high moisture separator drain tank level. Reactor scrammed on high pressure.
14	12/25/70	1690 MWt	Equipment Failure	Rapid closure of #2 stop valve during valve test caused actuation of turbine trip anticipatory scram.
15	12/25/70	810 MWt	Equipment Failure	After complete check of all systems pertinent to stop valve closure and anticipatory trip, #2 stop valve was retested at light load with a repeat of the event described in scram #50.

April 26, 1971

J. P. O'Reilly, Chief, Reactor Testing & Operations Br.
Division of Compliance, Headquarters

INQUIRY MEMORANDUM NO. 219/71-D
JERSEY CENTRAL POWER & LIGHT COMPANY (OYSTER CREEK 1)
LOSS OF HEATING BOILER - COLLAPSE OF HEATING BOILER STACK

The following information was received by the assigned inspector, Mr. R. McDermott, in a telephone call from the plant superintendent, Mr. T. McCluskey on April 23, 1971. Supplementary information concerning the occurrence was obtained in a telephone call to Mr. D. Ross, Technical Supervisor, by Mr. Higginbotham on April 26, 1971.

1. During the period April 20-22, 1971 it was noticed that the fixed particulate and charcoal filters on the stack monitor required changing at a shorter frequency than was normal. The air flow through the filters is self-compensating to maintain a nominal 2 cfm; an alarm is received in the control room when this flow drops and stays below normal.

2. Investigating possible causes for this occurrence, a maintenance man inspected the inside of the main ventilation stack and found that the ventilation stack for the heating boiler had collapsed. (The heating boiler stack was fabricated from "light weight asbestos-like" material.) This was found and the heating boiler was secured on Thursday, April 22, 1971.

3. This caused soot and flue gas to be picked up in the sample probe and lines where it was deposited on the filters of the stack sampler restricting the air flow.

4. Inspection of the interior of the stack revealed visual damage to the support structure for the isokinetic sampling probes for the main stack monitor. The licensee intends to repair this damage as soon as it is possible to do so. However, the repair will require a shutdown of the plant.

5. Inspection of the chart trace of the stack noble gas monitor showed that there has been no discernible change in indications of gas releases. The normal, for several weeks, indication has been about 100 to 150 counts/second (equivalent to about 10,000 to 15,000 uCi/sec); no particular deviation from these nominal indications were found.

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B/419

OFFICE ▶	COMPLIANCE				
SURNAME ▶	HIGGINBOTHAM:maz	CARLSON			
DATE ▶	4/26/71				

6. The Plant Operations Review Committee reviewed the occurrence and, based on information in item 5 above, decided that plant operation could continue.

7. At the time the heating boiler was secured on Thursday, April 22, 1971, fresh patticulate and charcoal filters were installed in the stack monitor. These filters were exchanged on Monday, April 26, 1971, and the results of counting and analysis of these filters will be compared to results of filter counting during and prior to the occurrence. An attempt will be made to ascertain if the "crud" deposited on the filters affected the evaluation of releases of particulates and halogens during the time the heating boiler stack was damaged.

8. A temporary stack was erected for the boiler, outside the main ventilation flue, and the boiler was returned to service Sunday evening, April 25, 1971. The use of the radioactive liquid waste concentrator was lost during the period of time Thursday to Sunday evening.

9. Inspection by plant personnel revealed no visual structural damage to the main ventilation stack.

The licensee plans to submit a written report of the occurrence to DRL within 30 days in accordance with the requirements of the facility license. The matter will be reviewed further during the next routine inspection of the facility and CO:HQ will be informed of additional details as appropriate.

R. T. Carlson
Senior Reactor Inspector

cc: E. G. Case, DRS (3)
P. A. Morris, DRS
R. S. Boyd, DRL (2)
R. C. DeYoung, DRL (2)
D. J. Skovholt, DRL (3)
P. W. Howe, DRL (2)
A. Giambusso, CO
L. Kornblith, Jr., CO
R. H. Engelken, CO
Regional Directors, CO
REG Files

March 24, 1971

J. P. O'Reilly, Chief, Reactor Testing and
Operations Branch, Division of Compliance, HQ

INQUIRY MEMORANDUM 219/71-C
JERSEY CENTRAL POWER AND LIGHT COMPANY (OYSTER CREEK 1)
ISOLATION CONDENSER ISOLATION RELAY FAILURE

Mr. T. McCluskey, Station Superintendent, informed the assigned inspector, Mr. McDermott, during a telecon on March 23, 1971, that during routine surveillance testing of the isolation condenser isolation feature on that date, one of the five second time delay relays in the isolation logic failed to operate. The ability to isolate the isolation condensers was not lost but the relay failure resulted in a loss of redundancy. The subject relay was manufactured by Numatic Corporation and was exchanged for an Agastat type relay. JC is currently planning to change three additional Numatic relays in the initiating logic. Mr. McCluskey stated that JC would submit a 30 day report of this event in accordance with license requirements. We plan a routine site inspection during early April 1971, at which time this event will be reviewed further.

R. T Carlson
Senior Reactor Inspector

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Regional Directors
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B/420

~~8305060011~~ (1p)

OFFICE ▶	CO					
SURNAME ▶	McDermott/jd JJM	Carlson RC				
DATE ▶	3/24/71					

March 2, 1971

J. P. O'Reilly, Chief, Reactor Testing & Operations Br.
Division of Compliance, Headquarters

INQUIRY MEMORANDUM NO. 219/71-B
JERSEY CENTRAL POWER & LIGHT COMPANY (OYSTER CREEK 1)
MAIN STEAM ISOLATION VALVE PERFORMANCE - FAILURE OF VALVE TO MEET LIMITING LEAK RATE

The assigned inspector, Mr. R. McDermott, was contacted by Mr. T. McCluskey, Station Superintendent on February 19, 1971* and informed that, during a recent outage (to inspect turbine control valves), all main steam isolation valves had been tested for leak tightness in accordance with prior commitments made in their letter to DRL dated March 20, 1970. Mr. McCluskey stated that the valves were tested using 20 psi air and that the two inside valves showed no detectable leakage. The measured leakage for the north outside isolation valve was 3.6 SCFH. He reported that the final test on the south outside isolation valve measured 4.3 SCFH. The technical specification leakage limit for an individual valve is 11.5 SCFH.

Mr. McCluskey stated that the initial closure of all main steam isolation valves was accomplished with less than 100 psig in the main steam system and with no steam flow. He stated that the first two tests conducted on the south outside main steam isolation valve were unsuccessful. The first and second test resulted in a measured leakage of ≈380 SCFH. This isolation valve was cycled between these two tests at atmospheric pressure conditions in the main steam system. The final leakage test (results - 4.3 SCFH) for this valve was conducted following a hydrostatic test on the primary system without further valve cycling. Mr. McCluskey stated that his basis for confidence that the valve would meet the leak tightness requirements of technical specifications was that it was tested in October, 1970 and found to be leak tight and that the station's prior experience with this valve indicated that it would not be leak tight if closed when there was less than 100 psi steam pressure in the system. He further stated that normal steam flow through the main steam isolation valves is in such a direction as to assist the closure of the valve and improve the leak tightness measurements, if the valve were to close with steam flow to the turbine. He stated that the timing of the closure of the main steam isolation valve did not indicate any abnormalities with past performance and that there were no obvious mechanical difficulties with the valve. Operation of the reactor was resumed on February 19, 1971.

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*Inquiry memorandum submitted at request of J. Keppler, CO:HQ on March 1, 1971.

OFFICE ▶	COMPLIANCE				
SURNAME ▶	McDERMOTT:maz RMcD	CARLSON RC			
DATE ▶	3/2/71				

Mr. McCluskey was recontacted on March 1, 1971, and agreed to submit a written report on the failure of the south outboard to DRL within the 30 days required by the license. We plan to review these leak tests in more detail during the next routine inspection scheduled for the week of March 15, 1971.

R. T. Carlson
Senior Reactor Inspector

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