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February 27, 1974

Mr. John F. O'Leary, Director
Directorate of Licensing
US Atomic Energy Commission
Washington, DC 20545

Re: Docket 50-155
License DPR-6
Big Rock Point-Semiannual

Dear Mr. O'Leary:

Enclosed herewith are three (3) originals and thirty-seven (37) conformed copies of the nineteenth Semiannual Report of Operations for the Big Rock Point Plant. This report covers the period of July 1, 1973 through December 31, 1973.

Included also, as Attachment A, are corrections to Section IID, Appendix A, Table I and Appendix B, of the eighteenth Semiannual Report.

Yours very truly,

Ralph B. Sewell (Signed)

Ralph B. Sewell
Nuclear Licensing Administrator

RBS/ds

CC: JGKeppler
USAEC

1720

CONSUMERS POWER COMPANY

BIG ROCK POINT PLANT

SEMIANNUAL OPERATIONS REPORT

NO 19

JULY 1, 1973 - DECEMBER 31, 1973

Operating License DPR-6
Docket 50-155

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56pp

CONSUMERS POWER COMPANY

BIG ROCK POINT PLANT

Nineteenth Semiannual Report
July 1, 1973 - December 31, 1973

I. INTRODUCTION - SEMIANNUAL OPERATING REPORT

The plant was base loaded at 69 MWe (gross) during this report period. The off-gas release rate on July 1, 1973 was averaging approximately 2,500 μCi/sec.

The outage beginning on November 1, 1973 to conduct the six-month Technical Specifications testing requirements marked the end of 198 days of consecutive power generation - a new record for domestic operating boiling water reactors. During this 198 day span (beginning on April 16, 1973), a total of 321,172 MWhe(g) were produced for an average output of 67.5 MWe(g) or 90% of rated power.

II. OPERATIONS SUMMARY

A. CHANGES IN PLANT DESIGN

Changes in the design of the plant which were incorporated as facility changes are as follows:

1. Facility Change C-215 (Primary System Leak Rate Equipment) - This change involved the installation of leak trace sampling tees as listed:

   a. A loop seal and leak trace sampling tee was installed in the pipeway cooling unit drain.

   b. A leak trace sampling tee was installed in the collection pot drain from the enclosure spray system relief valves.

   c. Three leak trace sampling tees were installed in the clean-up heat exchanger room on selected relief valves.

2. Facility Change C-216 (Primary System Leak Rate Equipment) - This change involved the addition of a two-inch drainpipe from the reactor recirculation pump seal collection sink to the reactor clean sump to identify process flows.

3. Facility Change C-218 (Primary System Leak Rate Equipment) - This change involved the construction of a temperature and dew point temperature sampling station for sampling air from both the supply and exhaust air ducts as they enter and leave the reactor enclosure.

   An air-cooling coil was modified to use the supply air (after dew point measurement) to cool the exhaust air (prior to its dew point measurement) to enable a wider range of measurement of the exhaust air.

   All points (4) were connected to read out on the dew point recorder located in the control room.

4. Facility Change C-221 (Primary System Leak Rate Equipment) - This change involved rerouting of the collection system for the control rod drive pumps and associated safety relief valve discharge to the reactor enclosure clean sump to identify process flows.

5. Facility Change C-222 (Primary System Leak Rate Equipment) - This change involved the construction of an angle iron dam around the clean sump to prevent water, accumulating or draining on the recirculation

pump room floor, from entering the clean sump. The dam was successfully leak tested using water at 7/8 inch above floor level.

6. <u>Facility Change C-224 (Primary System Leak Rate Equipment)</u> - This change involved the addition of an integrating water meter between the demineralized water storage tank and the condensate storage tank so that the amount of water transferred can be accounted for. Previously, that amount of water used for regeneration and rinsing of the makeup demineralizer had to be subtracted from the demineralizer flow meter integrator.

7. <u>Facility Change C-225 (Primary System Leak Rate Equipment)</u> - This change provided for the addition of a drain in the clean-up demineralizer room. No floor drain exists in this room. However, to facilitate early detection of unidentified leakage, the two-inch pipe stub (drain to enclosure dirty sump) in the clean-up demineralizer room floor was drilled and tapped for 1/4 inch pipe (two holes). These holes were made approximately 1/4 inch from the floor level and a screen was placed around the pipe stub. This arrangement will limit the total leakage to approximately 18 gallons before drainage to the dirty sump begins.

8. <u>Facility Change C-234</u> - This change involved the removal of the generator and bus instantaneous differential overcurrent relay (250B). Following the inadvertent tripping of the unit, a study conducted by the Consumers Power Electric Engineering Department revealed this relay scheme to be unnecessary. Adequate fault protection is provided through other existing relays.

9. <u>Facility Change C-235</u> - This change involved elimination of one and relocation of two annunciator circuits associated with the system transmission lines. These changes were the result of the installation of the 345 kV transmission line to this area and relocation of alarm systems external to the plant.

B. PERFORMANCE CHARACTERISTICS

At the start of this report period, the unit was on-line at 69 MWe(g) (220 MW_t).

A fuel inspection team from General Electric (GE) arrived on site July 7, 1973 for removal of individual fuel rods from various fuel bundles. These fuel rods were scheduled for metallurgical testing at the GE Vallecitos Test Center as a part of GE's fuel development program. Four shipments (25 fuel rods) were made this report period with a total of six shipments (40 fuel rods) shipped throughout the year.

Four shipments of spent fuel (32 fuel bundles) were made this report period to Nuclear Fuel Services at West Valley, New York for reprocessing, with the first shipment leaving on July 12, 1973 and the last on August 30, 1973. A total of eight shipments (72 fuel bundles) were made in 1973.

On July 20, plant load was reduced to 10 MWe to permit investigation of a component cooling water leak in the recirculating pump room. Component cooling water was found leaking from a line to the motor thrust bearing on the No 2 recirculating pump. Following repairs, the pump was returned to service and the plant load increased to 69 MWe.

On July 25, power was reduced to $\sim$200 MW_t by flow control during a test to determine the effect of recirculating pump flow on steam drum tilt. The recirculating flow was alternately decreased in each loop by throttling the recirculating pump discharge valves. Following the test, operation was resumed at 220 MW_t.

On August 16, power was again reduced to 10 MWe to permit entry into the recirculating pump room to investigate for component cooling water leakage. The flex line from the small heat exchanger on No 1 recirculating pump was found to be leaking. The system was repaired and the plant was returned to operation at 69 MWe.

One irradiated cobalt rod (11,000 Ci) was shipped to Neutron Products, Inc on August 18. This rod had been left in bundle D-61 during the recent cobalt shipping campaign and was later removed and stored.

On September 19, the clean-up pump tripped off and could not be restarted. This necessitated a power reduction to 10 MWe to permit entry into the recirculating pump room for the purpose of isolating the clean-up system. After this was accomplished, power was increased to 69 MWe on September 20. Following installation of the spare clean-up pump on September 22, power was once again reduced to permit entry into the recirculating pump room to valve the clean-up system into service. Upon completion, power was again increased to 69 MWe.

The semiannual containment component leak rate test was completed on October 8. Results indicated the leak rate was 45% of the maximum leak rate allowed in the Technical Specifications.

On October 18, the containment ventilation system was removed from service for a period of 1.5 hours to effect repairs to the four (4) solenoid valves (CV-9151, CV-9152, CV-9153 and CV-9154), which control the supply and exhaust ventilation valve air operators. A repair kit was installed in each solenoid valve to correct for excessive air leakage. Following repairs, the supply and exhaust ventilation valves were operated satisfactorily and returned to service.

On October 26, during fuel pool draining operations, an irradiated fuel rod was discovered on the bottom of the pool. Positive identification could not be made at the time and it was decided to store the rod in the fuel transfer cask. For accountability purposes, the rod is being carried in the plant records as an unirradiated E-type tie rod until such time that it can be positively identified.

On November 1, the plant was taken off the line for a scheduled outage to perform the semiannual control rod drive checks. After the reactor had been taken subcritical on November 1, a scram occurred on low condenser vacuum (22.8" Hg). The turbine bypass d-c isolation valve failed to close automatically on the loss of condenser vacuum but was closed manually after being exercised. This was corrected by resetting the limit and torque switches and relubricating the valve gear train.

The following tests were completed successfully during the outage:

1. CRD As Found Hot Withdrawal Timing
2. CRD Cold Scram Timing
3. CRD Coupling Integrity, Jog and Position Indication
4. CRD Cold Withdrawal Timing
5. One- and Two-Rod Shutdown Margin Checks

In addition, special tests of the recirculating pump interlocks and the poison system squib valve firing circuit were performed to verify system operability. The recirculating pump interlocks were tested successfully. Tests of the poison system squib valve firing circuit conducted during this outage were inconclusive. However, tests were successfully accomplished during the outage in December 1973.

The unit was returned to service following the outage on November 4 and reached 69 MWe on November 6. Sampling of the emergency condenser shell side water following return to power revealed leakage had occurred from the primary to secondary side during start-up. The north tube bundle of the emergency condenser was isolated and repairs were scheduled for the next outage.

On December 3, power was reduced to 58 MWe in order to decrease the off-gas release rate below 15,000 μCi/sec. The control rod withdrawal sequence was modified as well in order to limit the off-gas response to control rod withdrawal. On December 6, the off-gas release rate was again approaching 15,000 μCi/sec and a second power reduction was ordered, this time to 53 Mwe (or 1 feed pump operation).

On December 8, the unit was forced off the line (by means of a controlled shutdown) due to a packing failure on the level instrumentation lower root valve at the east end of the reactor steam drum. Repairs were completed on December 11, but the outage was extended to permit investigation of the emergency condenser tube leakage. The plant remained out of service throughout the remainder of the report period while the emergency condenser was being repaired. This consisted of repairs to three leaking tubes at the tube-to-tube sheet welds in the north tube bundle and modifications to the baffle plates in the inlet water box heads of both the north and south tube bundles. This latter work was contracted to South-West Research Institute for both design and installation of a baffle plate that would meet the system thermal stresses.

Other work performed during this extended outage included the successful testing of the liquid poison system squib valve firing circuit, which has previously been discussed, and an investigation into the off-gas holdup piping system. Following replacement of the off-gas isolation valve, an isolation test conducted on November 30 failed to demonstrate isolation capability. During the December outage, the valve was removed from the off-gas line and bench tested. It was found that the torque was insufficient to close the valve fully. The moment arm was increased to provide the necessary torque and the valve was then successfully bench tested. Following reinstallation into the off-gas line, the entire off-gas pipe from the flow orifice below the air ejectors to the isolation valve in the stack base was successfully pressurized to 6.5 psig. A volume test of the off-gas holdup pipe yielded a volume of 367 ft^3, in good agreement with plant as-built specifications.

At the start of the report period, the off-gas release rate was approximately 2,600 μCi/sec. This demonstrated a gradual but steadily increasing trend until just prior to the November 1, 1973 outage when the release rate had reached approximately 9,500 μCi/sec. Following return to power, the off-gas release rose sharply and reached 23,000 μCi/sec. Power was reduced on December 3 to hold the releases below 15,000 μCi/sec.

C. CHANGES IN PROCEDURES WHICH WERE NECESSITATED BY A AND B OR WHICH OTHERWISE WERE REQUIRED TO IMPROVE THE SAFETY OF FACILITY OPERATION

The following procedural changes were made with respect to plant operations:

A3.0	- Defines additon of Operations Engineer and further defines duties and responsibilities of plant staff.
A2.2	- Defines operator requirements for control room.
A2.6	- Revised the application of "Switching and Tagging Orders."
A3.7.7	- Defines the responsibility for "Locked Door and Valve Control" and redefines its application.
A8.0	- Incorporates a new section - "Maintenance, Test, Refueling and Special Procedures."

B1.3.3.5 - Corrects the incore calibration calculation.

B8.2.1- B8.3.1 - Revises plant operating requirements to reflect changes in Technical Specifications.

B11.3.2.8) B11.3.3.10) B11.3.4.27) - Further clarifies use of the radwaste discharge valves to the canal.

B15.2.6 - Includes the addition to survey the condenser by radiation protection personnel if opened for maintenance purposes.

B24.0 - Includes a new section on "Plant Operating Requirements," for chlorinating condenser and service water systems.

B28.6.0 - Includes a new secton on the "Emergency Diesel Cooling Water Pump Sealing System," describing its use and precautions.

B29.3.1 - Includes a section on placing the "Reactor Recirculating Pump Seal System" in service.

D14.0 - Revises the fuel shuffling winch procedure to prohibit its use for shuffling fuel in the reactor vessel.

E2.1.1 - Defines the "Chemical and Rad Protection Supervisor" duties.

E2.2.2 - Changes title of "Chemical and Rad Protection Engineer" to "Chemical and Rad Protection Supervisor."

E4.1.4.5 - Includes additional information on survey instruments for detecting beta radiation.

E4.1.8.1 - Incorporates a new section to describe the "Snoopy" neutron survey instrument.

E5.4 - Adds a new section to describe the limitations in the "Shipment of Waste and Other Radioactive Material."

E6.1 - Defines the responsibilities in controlling radiation protection records.

D. RESULTS OF SURVEILLANCE TESTS AND INSPECTIONS REQUIRED BY TECHNICAL SPECIFICATIONS

The following listing shows the systems tested, the required test frequency, the dates tested during this report period and the results of the tests.

1. Containment Isolation

   a. System: Containment isolation valve controls and instrumentation.

      Required Frequency: Quarterly (Conducted Monthly)

      Test Date: July 10, August 7, Sepetember 5, October 2, December 3

      Results: The automatic controls and instrumentation for eight of nine isolation valves were checked and found to function properly. One valve (main steam drain, MO 7065) is maintained in the closed position, de-energized and not used. Therefore, testing the automatic controls of this valve is not required.

   b. System: Isolation valve leak and operability test.

      Required Frequency: Twelve months or less.

      Test Date: Not required during this report period.

      Results: None.

   c. System: Containment sphere penetration inspection (visual).

      Required Frequency: Twelve months or less.

      Test Date: Not required during this report period.

      Results: None.

   d. System: Containment sphere integrated leak rate test.

      Required Frequency: Every two years.

      Test Date: No test required during this report period.

      Results: None.

   e. System: Containment sphere component leak rate test.

      Required Frequency: Six months or less.

      Test Date: October 5, 1973 to Ocotber 8, 1973.

Results: The containment component leak rate test was performed using air at ≥20 psig. The results of this test showed a total leakage of 45.2% of the allowed limit. Seventy-one percent of this leakage was from the supply ventilation valve. It is anticipated that this valve will be replaced at a convenient future plant outage.

2. Control Rod Drive System and Associate Tests

a. System: Reactor safety system scram circuits (not requiring plant shutdown to test).

Required Frequency: One month or less.

Test Dates: July 10, August 7, September 5, October 2, November 3, December 3.

Results: The reactor safety system was tested using the switches provided to simulate sensor trips. All channel trips occurred as designed. In addition, the neutron monitoring power range and intermediate range channels were tested for trip setting. All of these tests showed the trip settings to be within 120 ± 2% of power and 10-second period setting.

b. System: Control rod performance - run.

Required Frequency: Each major refueling and at least once every six months during power operation.

Test Date: November 2, 1973.

Results: The control rod drive continuous withdrawal and insertion test, including withdrawal timing, was performed for each drive. This test is performed during reactor shutdown following completion of other drive performance tests and adjustments and represents the results of the final timing of each drive under cold conditions. The results of this test showed all drives to be operating satisfactorily with most withdrawal times at 36 seconds. No withdrawal time was less than 23 seconds.

c. System: Control rod performance - jog.

Required Frequency: Each major refueling and at least every six months during power operation.

Test Date: November 2, 1973.

Results: Satisfactory latching of all drives.

d. System: Control rod performance - scram.

Required Frequency: Each major refueling and at least once every six months during power operation.

Test Date: November 2, 1973.

Results: The control rod scram test was performed for each drive. The test included time from system trip to 100% of insertion at a reactor temperature of about 150°F. The results of this test were satisfactory for all drives.

e. System: Reactor safety systems scram circuits (requiring plant shutdown).

Required Frequency: During each major refueling outage but not less frequently than once every 12 months.

Test Date: Not required during this report period.

Results: None.

f. System: Reactor safety system response time (requiring plant shutdown).

Required Frequency: During each major refueling shutdown, but not less frequently than once every 12 months.

Test Date: Not required during this report period.

Results: None.

g. System: Control rod withdrawal permissive interlocks function.

Required Frequency: Twelve months or less - the refueling interlocks will be tested prior to each major refueling.

Test Date: Not required during this report period.

Results: None.

h. System: Control Rod Drive Friction Test

Required Frequency: During each major refueling, but not less frequently than once each year.

Test Date: Not required during this report period.

Results: None.

3. Emergency Cooling

a. System: Core Spray System Check Valves

Required Frequency: Twelve months or less.

Test Date: Not required during this report period.

Results: None.

b. System: Post incident spray system automatic control operation.

Required Frequency: Twelve months or less.

Test Date: Not required during this report period.

Results: None.

c. System: Reactor Emergency Core Cooling System Trip Circuit

Required Frequency: Twelve months or less.

Test Date: Not required during this report period.

Results: None.

d. System: Containment Sphere Isolation Trip Circuits

Required Frequency: During each major refueling shutdown, but not less frequently than once every 12 months.

Test Date: Not required during this report period.

Results: None.

4. Miscellaneous Systems

a. System: Reactor Shutdown Margin Test

Required Frequency: After each refueling, after certain core component changes, if the system is cooled to atmospheric conditions and after 35,000 MWd_t have been generated.

Test Date: November 3 and 4, 1973

Results: The shutdown margin of 0.003 $\Delta k/k$ with the strongest rod fully withdrawn from the core was verified. In addition, the shutdown margin of 0.003 $\Delta k/k$ was verified with two adjacent rods fully withdrawn from the core.

b. System: Nil Ductility Transition Temperature Calculation

Required Frequency: At least once each year.

Test Date: Not required during this report period.

Results: None.

c. System: Moderator Temperature Coefficient Test

Required Frequency: Following each major refueling outage.

Test Date: Not required during this report period.

Results: None.

d. System: Subcriticality Checks

Required Frequency: During core alterations which increase reactivity.

Test Date: Not required during this report period.

Results: None.

e. System: In-Service Primary System Inspection

Required Frequency: A continuing program being conducted during some major refueling outages.

Test Date: Not performed during this report period.

Results: None.

f. System: Refueling Operation Control

Required Frequency: Each major refueling.

Test Date: Not required during this report period.

Results: None.

g. System: Reactor Refueling Safety System Sensors and Trip Devices

Required Frequency: Each major refueling.

Test Date: Not required during this report period.

Results: None.

5. Poison System

a. System: Liquid Poison System Firing Circuit Test

Required Frequency: Two months or less.

Test Date: September 5, 1973 and November 3, 1973

Results: Satisfactory. However, the test performed September 5 was untimely. It should have been performed on or about August 6. This was reported on a deviation report and reviewed by the Plant Review Committee.

b. System: Explosive Valve From Equalizing Line

Required Frequency: Twelve months or less.

Test Date: Not required during this report period.

Results: None.

c. System: Explosive Valve From Nonequalizing Lines

Required Frequency: Twelve months or less.

Test Date: Not required during this report period.

Results: None.

6. Radiation Monitoring

a. System: Air Ejector and Off-Gas Monitor System

Required Frequency: One month or less.

Test Date: July 26, August 23, October 1, October 31, November 30, December 28, 1973

Results: Checks showed the calibration to be satisfactory (within 20% of the 2.5×10^3 cps alarm setting). The automatic closure function of the isolation valve timer was checked. The test showed the timer calibration to be satisfactory (within 3% of the maximum timer setting) and the isolation valve closed as specified.

b. System: Calibration and Functional Test of the Stack Gas Monitoring System

Required Frequency: One month or less.

Test Date: July 26, August 23, October 1, October 31, November 30, December 28, 1973

Results: The stack gas monitoring system was checked using the built-in Cs-137 calibration source. The instrument check showed the calibration to be satisfactory, resulting in the alarm point occurring within the specified 0.1 curie per second release rate.

c. System: Analyses of Stack Gas Particulate and Iodine Filters

Required Frequency: Weekly.

Test Date: The analyses were conducted weekly.

Results: The results of analyses of the stack gas particulate filter and iodine filter are reported in terms of curies released in Appendix A of this report.

d. System: Calibration of Emergency Condenser Vent Monitor

Required Frequency: One month or less.

Test Date: July 31, August 23, September 28, October 31, November 29, December 28, 1973

Results: The emergency condenser vent monitors are checked by comparing with a calibrated portable instrument. The checks showed the vent monitor calibration to be satisfactory with all monitor checks within ± 5% of full scale.

e. System: Calibration of Canal Liquid Process Monitor

Required Frequency: One month or less.

Test Date: July 26, August 23, October 1, October 31, November 30, December 28, 1973

Results: The calibration of the canal liquid process monitor is a comparative calibration used to demonstrate operations of the monitor and to detect gross calibration changes and/or instrument drift. The results of these monthly calibrations showed that a monitor drift has occurred since the last calibration which utilized certified standards. Recalibration of the monitor with liquid standard sources will be completed shortly. Also, an acceptance criteria for a process monitor calibration will be developed.

f. System: Canal Liquid Collection Sample

Required Frequency: Daily.

Test Date: The analysis was conducted daily.

Results: Satisfactory.

E. THE RESULTS OF ANY PERIODIC CONTAINMENT LEAK RATE TEST PERFORMED DURING THE REPORT PERIOD

No integrated containment leak rate test was performed during the report period.

F. TECHNICAL SPECIFICATIONS CHANGES

During this report period, one Technical Specifications change was authorized by the Commission.

Change 39 - This change describes changes in plant organization and titles associated with the creation of the Operations Engineer and Maintenance Engineer job classifications.

G. CHANGES IN PLANT OPERATING ORGANIZATION INVOLVING KEY SUPERVISORY PERSONNEL

1. On July 1, 1973, the plant organization was changed. These changes were made to make the plant organization more responsive to present day operating requirements. The changes involved eliminating the Assistant Plant Superintendent position and adding the positions of Operations Engineer and Maintenance Engineer. The Operations,

Maintenance and Technical Engineers all report directly to the Plant Superintendent as does the Quality Assurance Engineer.

Mr. George Tyson assumed the job of Maintenance Engineer. Mr. Tyson had previously held the position of Assistant Plant Superintendent at Big Rock Point since he first reported there in 1968. Mr. Tyson has held a Reactor Operator's license at Big Rock Point since 1969.

Mr. Charles R. Abel was promoted to the position of Operations Engineer on July 1, 1973. Mr. Abel has been at the Big Rock Point Plant since 1967 except for a brief period when he served on a special assignment at the Pickering Power Station in Canada. Mr. Abel has held a Senior Reactor Operator's license at Big Rock Point since June 1969.

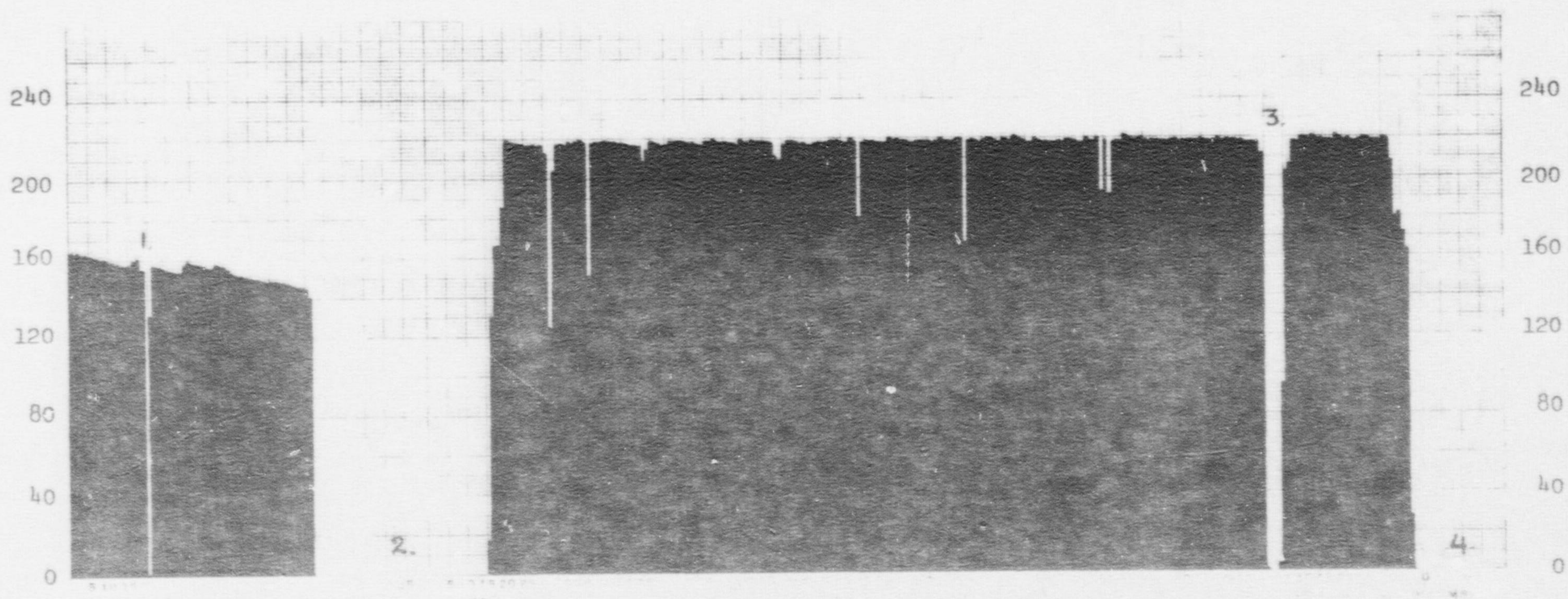
2. On November 15, 1973, Mr. Earl F. Peltier was promoted to the position of Assistant Shift Supervisor. Mr. Peltier has been at Big Rock Point continuously since March 1962 where he was on the original operating crew in the position of Control Operator No 1. Mr. Peltier has held a Reactor Operator license at Big Rock Point since 1962.

III. POWER GENERATION

	Report Period	Total To Date
1. Thermal Power Generated (MWh_t)	808,763	12,194,476
2. Gross Electric Power Generated (MWhe(g))	253,148	3,885,350
3. Net Electric Power Generated (MWhe)	240,287.2	3,680,070
4. Hours Critical (h)	3,757.2	68,664.8
5. Hours Generator On-Line (h)	3,751.0	66,882.3

OUTAGES

1. January 20: Replaced packing of clean-up system discharge valve (CU-1).
2. March 2: 10th refueling outage (scheduled).
3. November 1: Semiannual control rod drive checks (scheduled).
4. December 8: Replaced packing on the steam drum east level instrumentation lower root valve and repaired emergency condenser baffle plates and tubes.



IV. SHUTDOWNS

A. TYPE - SCHEDULED

1. Unit Off Line - 11/1/73 0036
2. Unit On Line - 11/4/73 2339
3. Length of Outage - 95 Hours, 3 Minutes
4. Discussion - This was a scheduled outage to perform the necessary semiannual license requirements on the control rod drives. Power descent was controlled and deliberate to a cold shutdown mode. Big Rock Point established an international record for length of operation of a BWR facility without power interruption. The plant had been in continuous operation since April 16, 1973 generating 198 consecutive calendar days at a unit capacity factor of 90%.

B. TYPE - FORCED

1. Unit Off Line - 12/8/73 0600
2. Unit On Line
3. Length of Outage (Plant Still Shut Down at End of Reporting Period)
4. Discussion - The unit was forced out of service due to a packing gland leak on an instrumentation root valve. Off-gas release rates were in the region of 10,000 μCi/sec (unit output was 53 MWe(g)) when the plant was shut down for repairs. The method of shutting down was a controlled deliberate shutdown to a cold shutdown mode. Valve packing was replaced; however, the outage schedule was extended to repair the emergency condenser. For details, please reference Section V G and VI A(5) of this report. At the end of the reporting period, the unit was off the line in the cold shutdown condition.

V. SAFETY-RELATED MAINTENANCE

Note: Dates contained in this section generally refer to the weekly period when the maintenance was performed.

A. REACTOR PROTECTION AND CONTROL SYSTEM INSTRUMENTATION

1. Neutron Monitoring Channel No 1 - 11/29/73 - Upscale drift in the output signal of this picoammeter was traced to defective contacts in the picoammeter range switch. Immediate repairs consisted of exercising the range switch between the "125%" and "Test Trip" positions to eliminate resistance in the range switch contacts. Subsequent repairs consisted of the same action with the picoammeter removed from service so that more switch positions (4) could be "wiped." The contacts in the range switch are in the feed-back circuit of the picoammeter and increased resistance in the feed-back loop tends to increase picoammeter output. This switch will be cleaned and inspected at the next refueling outage.

Failures of this type are considered to be within the design limitations of the equipment. The Technical Specifications and plant design provide for the temporary removal for maintenance of one power range flux monitor from service without compromising safety.

2. Neutron Monitoring Channel No 2

a. 9/13/73 - The picoammeter for this channel was replaced following reports of a rise in recorder trace level. The indication was still present following picoammeter replacement and was traced to the range switch for this channel. Exercising of the range switch alleviated the problem. This switch will be cleaned and inspected during the next refueling outage. Failures of this type are considered to be within the design limitations of the equipment.

b. 12/31/73 - The high-voltage power supply for this neutron monitoring channel was replaced with a spare unit following erratic flux level measurement at the most sensitive positions of the picoammeter range switch. Bench testing of the failed unit resulted in replacement of three marginal electron tubes. This failure occurred while the reactor was in "cold shutdown" for plant maintenance. Failures of this type are considered to be within the design limitations of the equipment.

3. <u>Neutron Monitoring Channel No 3</u> - 7/12/73 - The picoammeter in this channel was replaced with a spare unit on July 11 following small variations of 2%-4% on the neutron flux recorder trace. Bench testing of the unit removed revealed no problem and operation of the channel remained normal. The problem is now attributed to the range switch feed-back circuit contact resistance (similar to that observed in the other two power channels) and this switch will be inspected and cleaned at the next refueling outage.

Failures of this type are considered to be within the design limitations of the equipment. Technical Specifications and plant design provide for the temporary removal of one power range flux monitor from service without compromising safety.

4. <u>Neutron Monitoring Channel No 4</u>

a. 8/9/73 - The Log N-Period amplifier in this channel was repaired following reports of erratic period measurement while at power. Repairs consisted of replacement of a defective (gassy) electron tube in the period amplifier circuit. This type of failure is considered to be within the design limitation of the equipment. The Technical Specifications and plant design do not require this instrument to be in service when reactor power is above 5% of rated power.

b. 11/8/73 - The high voltage power supply for this channel was replaced with a spare unit following system response failure during instrumentation checkoff on November 3, 1973.

In performing the response check, it was noted that the Long N-Period indicator readings would increase when the compensation voltage was increased. This was first diagnosed as a defective chamber and the chamber and coaxial cables from the chamber to the chamber drive head were replaced. When this did not correct the problem, it was determined that the high-voltage power supply was defective and the unit replaced with the spare. Inspection of the defective power supply revealed that the unit was connected as a positive-positive supply instead of positive-negative as required. This unit was installed during a previous failure on June 4, 1973, while the reactor was at power and could not be checked for chamber response.

As a result of this error, which has been discussed by the Plant Review Committee and reported an abnormal occurrence (AO-13-73), appropriate steps have been taken to verify polarity of replacement power supplies when normal testing methods are not possible.

5. Neutron Monitoring Channel No 5 - 8/16/73 - The Log N-Period amplifier in this channel was repaired following reports of erratic period measurement. Repairs consisted of replacement of a defective electron tube in the period amplifier circuit. Failures of this type are considered to be within the design limitations of the equipment.

The Technical Specifications and plant design do not require this instrument to be in service when reactor power is above 5% rated power.

6. Neutron Monitoring Channel No 6 - 12/13/73 - The coaxial cable connector at the chamber location was repaired on this system following response failure after plant shutdown on December 8. The coaxial cable clamp on the chamber had loosened, placing strain on the cable and allowing the center wire to withdraw. The connector was repaired and the cable secured to preclude a future problem of this nature. This type of failure is considered to be within the design limitations of the equipment. Technical Specifications and plant design provide for removal of one start-up channel for maintenance during plant shutdown conditions.

7. Neutron Monitoring Channel No 7 - 12/31/73 - The high-voltage power supply for this channel was repaired following loss of count rate indication. Inspection of the supply indicated the voltage had dropped to approximately 300 volts (normally 850). Repairs to the supply consisted of electron tube replacement. This type of failure is considered to be within the design limitations of the equipment.

Technical Specifications and plant design provide for removal of one start-up channel for maintenance during plant shutdown conditions.

8. Reactor Protection System Sensors

a. 11/8/73 - Recalibrated the high reactor pressure scram and high condenser pressure scram bypass sensors to a more conservative set point. The high condenser pressure scram bypass sensors were found to operate at a less conservative set point than required and were reported as an abnormal occurrence (AO-12-73). The calibration of the reactor steam drum low water level scram sensors and the high condenser pressure scram sensors were checked at the request of the Operations Department. All sensors checked normally and were within required limits.

All testing of the sensors was performed with the reactor in cold shutdown condition.

b. 12/13/73 - Calibration checks were performed on the high condenser pressure scram bypass sensors to determine if any instrument drift had occurred since prior calibration (11/8/73). All sensors were within calibration specifications and operated normally.

Testing of the sensors was performed with the reactor in cold shutdown condition.

B. RADIOACTIVE EFFLUENT MONITORING SYSTEMS

1. Air Ejector Off-Gas System

a. Maintenance Related to Off-Gas System Integrity Testing

11/8/73 - Off-Gas Isolation Valve - The off-gas isolation valve, CV-4015, was replaced with a newly procured valve designed to close tightly enough to isolate the off-gas system. This work was performed with the reactor in cold shutdown.

12/31/73 - Off-Gas System - The following components were inspected and repaired with the reactor in the cold shutdown condition in preparation for integrity testing of the off-gas piping:

(1) After Condenser Drain Isolation Valve, CV-4030 - Inspection revealed scale on the valve internals and imperfect seating. The valve was cleaned, the seat and disc were lapped, the packing was replaced and the valve was test operated and returned to service.

(2) Air Ejector Off-Gas Drain to Radwaste Isolation Valve, CV-4035 - Inspection revealed the valve seat and disc to be in "fair" condition. The seat and disc were lapped and the valve was test operated and returned to service.

b. 11/8/73 - Off-Gas Monitor

(1) Purge Valve Bypass - A bypass line was added around the three-way purge valve (SV RL 25) to facilitate cleaning of this valve while the plant was on the line.

(2) Purge Valve Repair - The three-way purge valve (SV RL 25) and the two-way purge valve (SV RN 25) were disassembled and inspected (both valves were quite scaly and dirty). The valves were cleaned and reassembled and returned to service.

The work was performed with the reactor in the cold shutdown condition.

c. Off-Gas Filter Changes

11/8/73 - The off-gas filter and demister was replaced.

12/31/73 - The filter was replaced.

Both filter changes were made with the reactor in the cold shutdown condition.

2. Stack Gas Radiation Monitoring System

a. 7/19/73 - Several components in this system were checked following erratic operation of the single isotope channel. Replacement of the differential discriminator provided some improvement due to a higher output signal level. However, the major source of the problem was discriminator shift on the log count rate meter. This was corrected by recalibration of the discriminator. The coaxial cable between the differential discriminator and the log count rate meter was also replaced with a new cable of shorter length to reduce signal attenuation.

b. 7/26/73 - The spare differential discriminators (2) were bench tested and calibrated. Several marginal electron tubes were replaced and minor adjustments were performed on the regulated power supplies and "E" dial span controls.

c. 9/6/73 - The differential discriminator in this system was replaced with a spare unit following instability in the single isotope channel. The instability was evident only during the daily calibration procedure, at which time minor changes were required in the detector polarizing supply to maintain system calibration. Repairs to the failed unit consisted of electron tube replacement.

d. 9/13/73 - The linear amplifier was removed and repaired in this system following reports of erratic indication. Repairs consisted of replacement of defective electron tubes and alignment following failure of the automatic scan feature.

The failures above are considered to be within the design limitations of the equipment. Removal of this system from service is permitted by the Technical Specifications provided repairs are made promptly and the system is returned to service. The off-gas monitors provide backup for this monitoring system.

3. Liquid Process Monitoring

a. 9/13/73 - Discharge Canal Liquid Process Monitor - The linear count rate meter in this channel was replaced with a spare unit following reports of full scale failure. Subsequent bench testing resulted in repair of a defective internal power supply socket (broken solder connection). A failure of this type is considered to be within the design limitation of the equipment.

b. 10/4/73 - Canal Sample Pump - Failure of the canal sample pump to pump design capacity was corrected by replacement of the pump impeller and casing.

c. 12/6/73 - Discharge Canal Liquid Process Monitor - The detector high-voltage supply cable for this channel was repaired following loss of reading on the linear count rate meter. This occurred immediately after monthly detector calibration and was the result of moving the detector during the calibration process. The high-voltage connector was repaired and the system returned to service.

Removal of this system from service is permitted by the Technical Specifications provided repairs are promptly made and the system returned to service.

C. CONTAINMENT SPHERE ISOLATION SYSTEM

1. 10/18/73 - Sphere Ventilation System - Leaking fittings were tightened in the nitrogen lines associated with SV 9152 in the emergency operating system for the supply ventilation valves.

Performing leak tightness adjustments on low pressure gas tubing fittings without removing the gas system from service is within the scope of acceptable maintenance practices. This approach was utilized in this case and the safety of the sphere ventilation system was therefore not compromised.

D. EMERGENCY POWER SYSTEM

1. Emergency Diesel Generator

a. 10/4/73 and 10/11/73 - Investigation of a low battery charger reading on the emergency diesel battery system disclosed insufficient voltage and current output from the charger while on "fast charge." This was corrected by replacement of the batteries and the battery charger rectifier.

Plant operating requirements permit removing the emergency diesel from service for periods in excess of 30 minutes with the approval of the Plant Superintendent. In each of the cases noted above, the Plant Superintendent's approval was obtained for removing the diesel from service only for the time specifically required for final hookup, troubleshooting and replacement, respectively.

b. 12/13/73 - The control panel indicating lights for this unit were replaced with new sockets and lenses to improve reliability and visibility of alarm indication.

The original panel lamps were of three different varieties, some of which had screw type bases and were susceptible to vibration (and loss of indication). The new sockets have bayonet bases and are sufficiently bright to be seen in most room locations.

The lamp sockets provide local alarm indication only for a common trip and annunciator scheme on the diesel generator. Replacement of the lamp sockets involved low-voltage wiring and no special precautions were required to provide for reactor safety.

2. 125 V D-C Power System

a. 8/16/73 - A 2-1/2 amp d-c ground on the 125 V d-c motor control center ground test station was eliminated by replacement of low accumulator pressure scram unit No D174.

b. 12/20/73 - Intermittent fluctuations and flickering of the control room d-c emergency lighting disclosed the HGA relay contacts in lighting panel 5L to be burned up. The contacts were replaced returning the system to service.

Continued availability of the battery system was assured by removing the system from service only as required for replacement of failed parts.

E. FIRE PROTECTION SYSTEM

1. <u>Diesel Fire Pump</u> - 10/18/73 - The starting batteries for the diesel fire pump were relocated for safer accessibility.

The electric fire pump provided primary fire system supply potential while the diesel fire pump batteries were relocated.

F. LIQUID POISON SYSTEM

12/20/73 - Investigation of air leakage from operator diaphragm joint on poison system discharge valve CV-4020 disclosed inadequately tightened flange bolts. The diaphragm was replaced as a preventive maintenance measure and the flange bolts were properly tightened. This maintenance activity was conducted with the reactor in cold shutdown.

G. EMERGENCY CONDENSER SYSTEM

1. <u>12/13/73 - Emergency Condenser</u> - Hydrostatic testing of both tube bundles in the emergency condenser disclosed an inability of the north tube bundle to achieve test pressure. Investigation disclosed three leaking tubes in this bundle. Dye-penetrant inspection indicated the tube leaks to be in the seal weld joining the tube end to the tube sheet. Upon disassembly of this tube bundle, the inlet-outlet water box baf le plate was observed to be warped or bowed. The findings of the inspection on the north tube bundle resulted in the decision to conduct a similar inspection on the south tube bundle.

2. <u>12/20/73</u> - Further examination of both tube bundles disclosed the required repairs to be as follows:

a. North Tube Bundle

(1) Modify inlet-outlet water box baffle plate.

(2) Repair three leaks and two indications at the tube-to-tube sheet welds.

b. South Tube Bundle

(1) Modify inlet-outlet water box baffle plate.

(2) Repair eight indications at the tube-to-tube sheet welds.

Southwest Research was contracted to engineer and repair the baffle plates while Consumers Power qualified procedures and a welder to conduct the tube repair.

3. 12/27/73 - The old baffle plates were removed by air arc from the north and south water boxes. The water boxes were ground out and made ready for the installation of the new baffle plate. New baffle plates were installed in both water boxes as described in Section VI A(5) - Facility Change C-238.

4. Tube Sheet Welds - A mockup of the tube sheet was fabricated with 1/4 inch 304 stainless steel overlay, 20 holes drilled and tubes installed and welded, to qualify the weld procedure and then the welder to that procedure.

5. 12/27/73 - Emergency Condenser Outlet Valve MO-7053 - The valve was disassembled for inspection and nondestructive testing. The gate and seals were cleaned and reground before reassembly. This inspection was required because of slight leakage experienced while conducting the hydrostatic test on the tube bundle.

All repairs were made with the reactor in the cold shutdown condition.

H. REACTOR CLEAN-UP SYSTEM

1. Clean-Up Pump

a. 9/27/73 - The clean-up demineralizer pump was replaced during this report period due to failed windings. The replacement pump had been completely rebuilt following a winding failure which had occurred approximately one year prior to this date. The change out was again performed in accordance with applicable Quality Assurance requirements. The failed pump is scheduled for immediate replacement and a facility change has been approved to convert the pump from a welded-in installation to a flanged arrangement to simplify pump maintenance. (Facility change work was not yet initiated.)

This work was performed during a period when the reactor was in cold shutdown.

b. 11/15/73 - The failed clean-up system pump which was replaced was completely disassembled and decontaminated. The pump casing, bearing housings and end cover plate were salvageable and were returned to the manufacturer for rebuilding. The rebuilt pump will be stored as a spare.

I. PRIMARY COOLANT SYSTEM

1. Reactor Recirculating Water Pump

a. 7/26/73 - A leaking flexible cooling waterline to the No 2 reactor recirculating water pump thrust bearing was replaced during a load reduction this report period. The leak was caused by normal deterioration of the flexible hose material.

This work was performed during a power reduction with the No 2 recirculating loop isolated, thus assuring personnel and reactor safety.

b. 8/16/73 - A failed flexible cooling waterline to the 3/4 inch heat exchanger for the No 1 recirculating pump was replaced.

This work was likewise performed during a power reduction with the No 1 recirculating loop isolated.

2. Reactor Recirculation Pump No 1 - 11/8/73 - Consumers Power Company Region Electric Laboratory replaced a defective thermal overcurrent relay (149 TMC - X phase) in the No 1 reactor recirculation pump motor control scheme. The alarm setting on the defective relay could not be properly adjusted to the required setting.

The relay was replaced while the plant was in cold shutdown for other testing.

J. CONTROL ROD DRIVE SYSTEM

1. Control Rod Drive Pumps (CRD)

a. 8/23/73 - Improper operation of the CRD pump discharge low-pressure alarm was traced to a plugged reference line to the switch. The line was dismantled, flushed and returned to service.

The repair was made during reactor operation with the CRD system at operating pressure. However, the reference line provides sensing for alarm and indication only and had no significant effect on plant operation as backup indication of the CRD discharge pressure is available.

b. 9/27/73 & 10/4/73 - Investigation of pressure pulsations on the No 1 CRD pump disclosed that one discharge valve seat and one suction valve seat were cocked in the pump casting allowing leakage between the casting and valve seats. The gouged out areas of the pump casting were repaired with Devcon Plastic Steel and the valve reseated.

The above repairs on the CRD pumps were made with the reactor at power. One of the two CRD pumps may be removed from service and still maintain normal operational status.

2. Control Rod Drives and Instrumentation - 11/8/73 - The following repairs were made to the CRD position probes:

C-4 - Aligned "02" switch position.

C-5 - Replaced a defective reed switch for the "03" position. The reactor was in the cold shutdown condition during the above probe repairs.

3. Control Rod Drive Scram Accumulators

a. 7/26/73 - E-4 Accumulator - A failed gas side accumulator burst disc was replaced on the E-4 CRD accumulator. Failure was determined to be due to gas erosion.

b. 8/9/73 - E-3 Accumulator - The nitrogen O-ring seals on CRD E-3 accumulator were replaced to stop leakage.

c. 8/30/73 - C-4 Accumulator - A leak from the joint between the accumulator halves was corrected by replacement of the O-ring and backup rings utilized at that point.

d. 8/30/73 - Replaced a defective gauge on C-4 accumulator following reports of false readings during charging of this accumulator.

e. 9/13/73 - D-4 Accumulator - A leak from the joint between the accumulator halves was corrected by replacement of the O-ring and backup rings used as seals at that point.

f. 9/27/73 - A-4 Accumulator - Leakage from the A-4 accumulator was corrected by replacement of seals between the accumulator halves.

g. 12/20/73 - F-2 Accumulator - Leakage from the F-2 accumulator was corrected by replacing the bladder and the O-rings and backup rings on both the water and gas sides of the accumulator.

The above repairs were conducted with the reactor at operating pressure. Under these conditions, the primary hydraulic source for CRD scramming comes from the reactor vessel. This design feature permits repair of an accumulator without affecting reactor safety.

In each case, only one accumulator was removed from service and only for the time required to perform the corrective maintenance.

3. Accumulator Drain Valves

a. 8/30/73 - C-4 and C-5 Accumulator Drain Valves - Leaks from the waterside drain valves were repaired by relapping the valve discs and seats.

b. 9/13/73 - D-4 Accumulator Drain Valve - Leaks from the waterside drain valve were repaired by relapping the valve disc and seat.

The above repairs were made with the reactor at operating pressure. Under these conditions, the primary hydraulic source for CRD scramming comes from the reactor vessel. This design feature permits repair of an accumulator without affecting reactor safety.

In each case, only one accumulator was removed from service and only for the time required to perform the corrective maintenance.

4. CRD Pump Relief Valves

a. 7/12/73 - Control Rod Drive Pumps - Excessive leakage from the No 1 CRD pump relief valve necessitated its replacement with a rebuilt valve. The newly installed valve was set to relieve at 1900 psig.

b. 10/25/73 - Control Rod Drive System - The No 2 CRD pump relief valve was replaced due to excessive leakage.

c. 11/18/73 - No 1 CRD Pump Relief Valve - The valve was repaired and reset due to excessive leakage.

K. FEED-WATER SYSTEM

1. Reactor Feed Pumps - 12/13/73

No 2 Reactor Feed Pump - Failure of the pump to start through use of the control room hand switch resulted in the following inspections:

a. Control Circuit - The feed pump breaker was tested and cleaned. Inspection revealed the trip coil linkage to be binding slightly preventing resetting of the coil after a pump trip.

b. Auxiliary Oil Pump - Indications of low after-filter discharge pressure (8.5 psig) on the No 2 auxiliary oil pump were corrected by readjusting the relief valve to relieve at 10.1 psig (approximate correct setting). Low discharge pressure prevented the associated pressure switch from closing, completing the feed pump start circuit.

Either of the conditions in a. or b. above would have prevented the pump from properly starting. Corrections of these problems have returned the No 2 feed pump to service.

The above repairs were made during the forced outage of 12/8/73, during which the plant was in the cold shutdown condition.

L. STEAM DRUM

1. Steam Drum Level Sensing Root Valves

a. 11/8/73 - Minor leakage from the steam drum east end level element bottom reference line valve was corrected by tightening the packing.

b. 12/18/73 - The failed packing on the east end instrument root valve was replaced, restoring the instrumentation system to service. The west end instrument root valve packing was also replaced as a preventive maintenance measure.

The above maintenance activities were performed with the reactor in cold shutdown and the steam drum drained.

VI. CHANGES, TESTS, EXPERIMENTS

A. FACILITY CHANGES PERFORMED PURSUANT TO 10 CFR 50.59

1. Facility Change C-214

This change added sealing water to the shaft seal on the diesel generator cooling water pump. The seal water system utilizes a head tank supplied from either the service water system or the diesel generator water pump. The head tank will provide 1 gpm water at approximately 4.4 psig for 24 minutes which will be adequate to provide sealing water to the pump shaft seal until the diesel is started. The 4.4 psig seal water pressure will be sufficient to permit early detection of pump shaft packing problems and eliminate pump failure due to packing air leakages. The addition of the seal water system does not constitute a change in the designed function of the diesel generator as described in the FHSR and the Technical Specifications; nor is the designed safety of the diesel generator impaired since the seal water system will assure adequate cooling of the diesel through proper functioning of the water pump.

2. Facility Change C-228

This change covered the plugging of the drain line from each reactor feed pump base. The original design provided for draining of accumulated waste (oil and/or water) to either the radwaste or turbine sumps. Valves were provided to close the drains off entirely or to divert the flow to either of the sumps mentioned above. During normal operations, these drain lines were valved closed. When they were used, they drained accumulated waste from the pump bases to the turbine sumps and from there to the clean waste receiver tanks. The use of these drains thus provided the potential of introducing oil into the clean waste system. Since the normal accumulation of waste on the pump bases is of a small enough volume to allow easy removal by hand, the potential of introducing oil into the clean waste system was eliminated by plugging the drain lines. A safety evaluation determined that this change did not constitute a change in the designed function of the feed pumps as described in the FHSR and the Technical Specifications; nor is the designed safety of the feed pumps impaired since accumulated waste cannot enter into the feed pump mechanisms or the components of any other equipment in the vicinity.

3. <u>Facility Change C-230</u>

This change facilitated construction of a 12-foot diameter by 20-foot high tank for storage of radioactive materials removed from the spent fuel pool while it was being relined. Safety analyses pursuant to 10 CFR 50.59 were conducted to examine the effects of flooding and exposure should the tank rupture. The analyses assumed complete tank rupture and concluded: (1) there would be no additional hazards to safety related equipment if the tank contents were limited to 1217 cubic feet of water; and (2) the resultant exposures at the nearest site boundary would remain within applicable requirements if the field at the top of the tank were limited to 10 R/hr. Since these criteria established that failure of the tank would not present a significant change in the hazards considerations described or implicit in the FHSR, the tank was so utilized.

4. <u>Facility Change C-231</u>

This change facilitated the addition of a bypass line around the three-way air purge valve on the off-gas monitoring system. The bypass line will allow inspection or repair of the three-way valve during normal plant operation. (In the past, the effectiveness of purge in removing excess moisture has been hindered through the introduction of dirt into the three-way valve.)

A safety evaluation concluded that this change does not constitute a change in the designed function of the off-gas monitoring system as described in the FHSR and the Technical Specifications; nor is the designed safety of the system impaired since malfunctions can now be quickly corrected.

5. <u>Facility Change C-238</u>

The change replaced the original baffle plates on the emergency condenser inlet-outlet water boxes. The original baffle plates had been warped in service and were replaced with the following redesign: Each new baffle plate consists of a center section which is bolted to a narrow ledge welded horizontally around the three interior sides of inlet-outlet water box; and, which when in position on the tube bundle butts up against a fourth ledge welded horizontally to the tube sheet. The redesign of the baffle plates considered flow induced and thermal response loadings on the plates and bypass flow around the plate (the fourth ledge replaced a flexitallic gasket on the original design). The new design concluded that the flow induced loading and the bypass flow to be relatively insignificant and the highly flexible design of the new baffle plates to adequately withstand the thermal response loadings. It was thus concluded that the baffle plate modification does not constitute a change in the designed function of the emergency condenser as described in the FHSR and the Technical Specifications; nor is the designed safety of the system impaired since the new baffle plates will withstand the loadings imposed during the intended service. In support of these conclusions, a baseline test of the emergency condenser north tube bundle will be conducted when the system is returned to service.

B. TESTS PERFORMED PURSUANT TO 10 CFR 50.59(b)

1. Liquid Poison System Explosive Valve Firing Circuits

a. 11/8/73 - During the scheduled shutdown for semiannual testing, the firing circuits were tested for operability utilizing 2-ampere fuses in place of the explosive valves. The test failed due to the incompatibility of the test equipment with the circuit design. Further test results are described below.

b. 12/20/73 - Component checks and operability tests were performed following plant shutdown. Resistance measurements indicated that no problems existed in the relay contact surfaces or fuse clip resistance.

Firing tests were performed using various size fuses in the explosive valve test devices (simulators). Firing with the 2-ampere fuses was marginal as a time lag is encountered upon firing (the parallel 2-ampere fuses approach total circuit current capability).

Firing with the 1-ampere fuses was acceptable. The total time required to open all fuses was 0.19 second (including time required for control relay closure).

Firing with the 0.5-ampere fuses was nearly instantaneous. The total time required for fuse opening was approximately 0.070 second, including the 0.050 second required for relay closure. Firing of the 1-ampere fuses indicates the circuit is reliable in meeting the 1-ampere maximum firing current criteria of the explosive valves.

c. 12/27/73 - In order to verify an adequate safety margin above the 1-ampere maximum firing current criteria of the explosive valves, an operability test using 1.5-ampere fuses was performed on this system on December 22. All circuits worked as designed and the total firing times for Circuits A and B were 0.550 and 0.650 second, respectively.

The reactor was in cold shutdown during all testing on the poison system and no additional precautions were required to provide for reactor safety.

All tests were performed using approved written procedures. Prior reviews of these procedures were performed to ensure that these tests were consistent with Technical Specifications and did not involve an unreviewed safety question per 10 CFR 50.59.

2. Steam Drum Level Tests

a. 7/25/73 - A special test was conducted to determine if variable reactor recirculation pump flows could affect steam drum levels at either end. The test results showed that there had not been any subtle or small changes in relative pump flows which would account for the change in drum tilt over the past years.

b. 10/31/73 - A special test was committed to in order that data may be gathered to evaluate any change in the reactor steam drum tilting phenomena as a function of power. Steam drum elevations were measured for both ends of the drum at power levels of 10 MWe(g), 30 MWe(g), 50 MWe(g) and 70 MWe(g). There was little noticeable change over the full range of power settings. The data are tabulated below:

Plant Output	Yarway West RE06A	Yarway West RE20A	Yarway East RE06B	Yarway East RE20B	Bailey East Recorder	Bailey West Indicator
69 MWe(g)	+3	+3	0	-1	0	+4
50 MWe(g)	+3	+2.5	0	-1	0	+4
30 MWe(g)	+3	+3	0	-1	0	+4
10 MWe(g)	+3	+2.5	0	-1	-.5	+3

This test will continue to be run at regular six-month intervals to verify the posture of the drum.

This test was performed using approved and written procedures. A prior review of this test determined that this test was consistent with Technical Specifications and did not involve an unreviewed safety question per 10 CFR 50.59.

3. Reactor Recirculation Pump Interlock Tests

11/3/73 - A special test was conducted to fulfill the operating requirements of Technical Specifications, Section 6.1.5(q). The logic associated with the pump starting circuits with the various settings of the valving was checked. All systems logic checked out as required. One deficiency, the annunciator on No 2 recirculating "pump trip," did not alarm. The circuit was repaired and test-operated satisfactorily.

In conjunction with the above tests, the valves (pump suction, pump discharge and discharge bypass) were timed over their entire opening and closing strokes. The pump discharge valves were found outside the limits specified in the Technical Specifications as indicated in our letter to the AEC dated December 6, 1973. The gear train ratio in the limitorque valve operators will be modified to bring the timing within limits.

This test was performed utilizing an approved written procedure. Prior review of this procedure indicated that it was consistent with Technical Specifications and did not involve an unreviewed safety question per 10 CFR 50.59.

4. Off-Gas Isolation Valve Tests

12/29/73 - A special test was run following the installation of a new off-gas isolation valve. The purpose of the test was to determine the integrity of the off-gas piping system and to verify the holdup line volume. The initial test of the system showed leakage in the region of the absolute filter and the off-gas isolation valve. Following repairs (see Maintenance Section of this report), the integrity of the system was verified with a static holding test.

The volume of the system was also calculated to be as designed. At the close of the report period, isolating tests of the off-gas system were being formulated and will be conducted with the plant in service. This test was performed using a written and approved procedure. Prior review of this procedure indicated that it was consistent with Technical Specifications and did not involve an unreviewed safety question per 10 CFR 50.59.

VII. RADIOACTIVE EFFLUENT RELEASES

A. INTRODUCTION

Releases of radioactive material both to the atmosphere and Lake Michigan from January 1 to December 31, 1973 were well within the facility-licensed limits and the AEC's regulations, particularly Title 10, Code of Federal Regulations, Part 20.

B. GASEOUS EFFLUENT

Gaseous releases to the atmosphere totaled 224,500 curies of fission and activation gases. This corresponds to .17% of the licensed technical specification limit of 1 Ci/s. Particulate releases totaled 0.37 curie or 0.87% of the licensed limit while halogen releases were measured to be 4.7 curies or 27% of the licensed limit. *Gross alpha measurements on the particulate filter revealed that the release of alpha emitting nuclides totaled 1.6×10^{-6} curies. The tritium releases for the period totaled 85 curies of 7×10^{-5}% of limit based upon meteorological dispersion to the point of maximum ground concentration.

1. Gaseous Effluent Calculational Methods

A sample of off-gas is obtained weekly during power operation and analyzed by gamma spectrometry for **six noble gas radionuclides. Based upon the mixture of the six nuclides, a stack release rate, which includes a total of 22 noble gas radionuclides, is determined. The stack release rate is based on a 26-minute holdup time for off-gas plus a 1% contribution from the turbine sealing steam system utilizing a 2-minute holdup. The 1% turbine seal contribution has the same distribution of nuclides as the off-gas corrected for a 2-minute decay period. This is reflected in the monthly totals shown in Appendix A.

*Due to uncertainties in iodine collection efficiencies for various species and potential sample line plateout, the measured values will be arbitrarily tripled for reporting purposes. A detailed study is currently being made to empirically quantify (and significantly reduce) the appropriate correction factor.

**The six nuclides are: Kr-85m, -87, -88 and Xe-133, -135 and -138.

Activation gas releases are composed primarily of N-13. The rate of release is power-level dependent and is incorporated in the total monthly releases shown in Appendix A.

Particulate and halogen releases to the atmosphere are measured by counting particulate and charcoal filters weekly. These filters collect stack effluent continuously at a rate of 3 cubic feet per minute. Determination of release rates in this manner assumes radioactivity is continually being deposited uniformly throughout the week on the filters and, hence, a decay correction to the time of analysis is applied, depending on the half-life of the nuclide observed.

Table I, Appendix A, has been revised since the last Semiannual to properly distinguish between total particulates released and gross beta activity on the particulate filters and to correct an error in the reported percent of Technical Specifications limit for particulate releases. (See footnote Table I, Appendix A.) The net beta activity, as now reported in Appendix A, represents the unidentified portion of the total activity present on the particulate filters (ie, gross beta activity minus the identified isotopic activity). Unlike the individual isotopes, the net unidentified beta activity, due to the lack of a known half-life, has not been corrected for continuous deposition and decay.

Tritium releases to the atmosphere are calculated, based upon measurements made in the primary coolant and containment air and using identical concentrations for all releases as follows:

a. Off-Gas - A flow rate of 10 cfm containing 90% radiolytic gas by volume at primary coolant tritium to hydrogen ratio and at 100% relative humidity is used to determine tritium releases both in vapor and molecular form.

b. Turbine Sealing Steam - The measured flow rate at 100% relative humidity and primary coolant tritium to hydrogen ratio.

c. Containment Ventilation - The measured flow rate and measured containment building tritium concentration.

The results of these calculations are also shown in Appendix A.

C. LIQUID EFFLUENTS

Liquid waste releases totaled 2.65 curies of radioactive material. This release corresponds to 3.1% of Technical Specifications limits. Additionally, 19.7 curies of tritium were released corresponding to 0.006% of 10 CFR 20 permissible concentration in the discharge canal.

1. Liquid Effluent Calculational Methods

The release pathway to Lake Michigan for all liquid effluents is through the plant's condenser circulating water discharge canal. A flow rate of 49,000-53,200 gpm dilution for liquid effluents is obtained through the use of the condenser circulating water pumps, two at 24,500 gpm each and house service water pumps, two at 2,100 gpm each.

Each collected tank of liquid is sampled, analyzed for radioactive content, and discharged at a controlled rate to assure that permissible concentrations are not exceeded in the canal prior to dilution in Lake Michigan during the time of discharge. Each sample is analyzed by gamma spectrometry to identify as many of the component nuclides as possible. (See Appendix B for results.) Permissible concentrations in the canal are determined from the following:

$$\sum \frac{Ci_i}{MPC_i} \leq 1$$

where Ci is the concentration of the ith isotope in the canal at the given concentration measured in the tank diluted by the known canal flow rate.

Those isotopes not identified by gamma spectrometry but measured by a gross beta analysis are presumed to be Sr-90 and released on that basis. Periodic samples of the batches are then sent to the radiological environmental contractor and analyzed for Sr-90 and Sr-89. From concentrations of Sr-90 and Sr-89 found in the batches, the total curies released of these two isotopes is calculated and used in calculating the percent of applicable limit in Appendix B. The remaining unidentified isotopes are assigned an MPC of 3×10^{-6} μ Ci/ml per 10 CFR 20. Tritium released are based on average concentrations in both "clean" and "dirty" waste tanks.

D. SOLID WASTES

A total of 11,583,948 curies of radioactive material was shipped off site during the period covered by this report. Of the total, irradiated cobalt accounted for 11,000 curies, spent fuel 11,561,575 curies and solid radwaste 11,373 curies. See Appendix C.

VIII. ENVIRONMENTAL MONITORING

A. ENVIRONMENTAL SURVEY

Environmental levels of radioactivity as found in the vicinity of the plant were composed almost entirely of naturally occurring radioactive materials. In the vicinity of the circulating water discharge canal, radioactive material of plant origin was found. These materials occurred primarily in aquatic organisms. The levels of radioactive materials, however, were extremely low and are of no significance to the health and safety of the organisms or the public. Further, the levels of radioactive material found in the resident biological community are consistent with levels found in previous years and show no upward trend.

The environmental surveillance program includes continuous sampling of air for particulate and halogen activity at seven locations including background sample locations at Traverse City and Boyne City, Michigan, about 50 miles south-southwest and 20 miles southeast of the plant, respectively, to determine increased concentrations, if any, of radioactivity of plant origin.

In addition, film badges and thermoluminescent dosimeters (TLD), placed at each of these locations plus six additional locations on the site property boundary, measure direct dose in the environment. Average monthly doses at the site, inner ring and background stations are compared and any difference, at the 95% confidence level, is reported using standard "F" and "t" tests. The results of these dosimeter analyses are given in Appendix D. While all the dosimeters record doses from natural occurring sources, the dosimeters on site can also be expected to receive doses from not only the plume but direct radiation from the plant. The site dosimeters showed, on an average, 0.74 ± 0.31 mR/mo above the background station dosimeters. During the same period of time, the inner ring of dosimeter stations did not show a dose rate above the background station dosimeters.

Air samples gathered continuously and analyzed weekly at the stations shown in Appendix D showed no difference, at the 95% confidence level, in the level of radioactivity measured at those stations close to the site and those remote from the site. Both particulate filters and carbon cartridges are used to measure potential concentration of radioactive materials resulting from plant operations. From the known meteorological dispersion conditions, the following maximum concentrations can be calculated:

Particulates (May) $(1.2\ \mu Ci/s) \times (0.013) \times (5.0 \times 10^{-14}\ s/cm^3)$

$= 7.8 \times 10^{-16}\ \mu Ci/cm^3$

*Halogens (March) $(1.2\ \mu Ci/s) \times (1.32) \times (5.0 \times 10^{-14}\ s/cm^3)$

$= 7.9 \times 10^{-14}\ \mu Ci/cm^3$

*Reflects measured values multiplied by three.

These compare to the minimum detectable activity values and normal background concentrations as follows:

Release	Maximum Calculated Concentration $\mu Ci/cm^3$	Minimum Detectable Activity $\mu Ci/cm^3$	Normal Background Activity $\mu Ci/cm^3$
Particulate	7.8×10^{-16}	1×10^{-14}	7×10^{-14}
Halogen	7.9×10^{-14}	2×10^{-13}	-

Hence, the negative data obtained in the program was expected.

Also, at the Big Rock Point Plant, daily composite condenser circulating water inlet and canal water discharge samples are taken and analyzed for radioactive content. In addition, a monthly composite of these samples is analyzed for radioactive content. These results are shown in Appendix D. Additional aquatic samples are taken and analyzed during the summer growing season and these results are also tabulated in Appendix D.

Based upon the liquid release of 2.06 curies of radioactive material (less tritium and noble gases) which results in an annual average concentration in the discharge canal of $2.0 \times 10^{-8}\ \mu Ci/ml$, the analysis of discharge canal water should indicate an increase of radioactive material in discharge canal water samples since the minimum detectable activity for gross beta measurements is about $5 \times 10^{-9}\ \mu Ci/ml$ or about four times lower than the average concentration discharged. The results shown plotted in Appendix D indicate an average of about $(1.2 \pm 0.94) \times 10^{-8}\ \mu Ci/ml$ for the year, which is in close agreement with the calculated concentration.

B. ENVIRONMENTAL DOSE CALCULATIONS

Levels of radioactive materials in environmental media indicate that public intake is well below 5% of that which could result from continuous exposure to the concentration values listed in Appendix B, Table II, 10 CFR Part 20.

1. Atmospheric Releases

In order to predict potential radiation doses resulting from gaseous releases, environmental transport and uptake factors must be known. A confirmation of these calculated doses is attempted then by measuring levels of radioactive materials in the plant's environmental surveillance program.

Currently, a computer model is used to calculate radiation dose resulting from plant releases of noble gases. The integrated population dose, out to 50 miles, for 1973 is shown on the following page. The computer model utilizes the following:

a. X/Q values for the five sectors are averaged over both stability class and wind frequency.

b. Doses are calculated for each of the 22 noble gas radionuclides and daughter products based on individual decay energies. Total dose is then the summation of the individual nuclide contributions.

c. The 1973 population is estimated from the 1970 Census of Population on a township basis corrected by the census-determined State of Michigan growth rate of 1.3% per year and includes transient population as 1/4 residents. The total estimated 1973 population resides 24 hours per day all year at the same location.

d. The actual mixture found during the weekly off-gas analysis is used for that week's releases and the total release is further corrected by daily measurements of off gas.

e. Site boundary doses are finite cloud shine doses. Semi-infinite cloud geometry is utilized to calculate doses after the plume reaches ground level.

f. No credit is taken for the meandering of the plume before it reaches the different annuli.

The maximum calculated radiation dose at the site boundary resulting from noble gas releases was 7.5 milliRems. The integrated dose to the population out to 50 miles was 6.0 person-Rems.

Doses from particulate, iodine and tritium releases as shown in Appendix A were negligible compared to that received from noble gases due to the conservative limits in the plant Technical Specifications.

2. Liquid Releases

In order to predict potential radiation doses resulting from the liquid releases, environmental transport and uptake factors must be known. A confirmation of these calculated doses is then attempted by measuring levels of radioactive materials in the plant's environmental radiation surveillance program.

The nearest municipal drinking water supply intake is located in Charlevoix, Michigan, which is generally upstream of the prevailing current flow in Lake Michigan at this location. However, since current patterns do occur that could, at times, carry the discharged water in the direction of Charlevoix, population dose based upon this flow is calculated in the next section of this report. A conservative dilution factor of 800 is taken from the point of discharge to the city of Charlevoix based upon the report, "Big Rock Point Hydrological Survey, Great Lakes Research Division, University of Michigan, Special Report No 9," by John C. Ayers, 1961.

In addition, the population dose is calculated to the entire population which receives its drinking water from Lake Michigan, based on a uniform concentration, resulting from plant releases, throughout Lake Michigan. Also, radiation dose to human populations can occur as a result of plant releases through the consumption of fish caught in Lake Michigan.

Utilizing the measured values of radionuclides released as shown in Appendix B, the following formula, and the standard man model, drinking water doses can be calculated as follows:

CALCULATED RADIATION DOSES FROM GASEOUS RELEASES

January 1, 1973 to December 31, 1973

(Person-Rems)

Distance (Miles)	Sector 1	2	3	4	5	Total
1-2						
Population	13	75	0	10	0	98
Population Dose	0.019	0.054	0.0	0.013	0.0	0.086
2-3						
Population	264	270	0	51	73	658
Population Dose	0.24	0.14	0.0	0.048	0.067	0.50
3-4						
Population	562	397	0	48	58	1,065
Population Dose	0.37	0.15	0.0	0.034	0.039	0.59
4-5						
Population	722	3,344	0	103	0	4,169
Population Dose	0.21	1.7	0.0	0.057	0.0	2.0
5-10						
Population	2,102	24	0	534	0	2,660
Population Dose	0.44	0.003	0.0	0.14	0.0	0.59
10-20						
Population	8,987	395	747	14,115	327	24,571
Population Dose	0.47	0.013	0.049	0.93	0.018	1.5
20-30						
Population	9,651	3,504	1,902	4,623	327	20,007
Population Dose	0.14	0.032	0.038	0.092	0.006	0.31
30-40						
Population	22,775	4,081	2,916	4,847	0	34,619
Population Dose	0.14	0.016	0.025	0.043	0.0	0.22
40-50						
Population	40,790	8,888	5,873	12,101	0	67,652
Population Dose	0.14	0.018	0.026	0.054	0.0	0.24
0-50						
Population	85,866	20,978	11,438	36,447	785	155,409
Population Dose	2.17	2.13	0.14	1.41	0.13	6.0
Site Boundary Dose (Rem)	6.5×10^{-3}	4.2×10^{-3}	-	7.5×10^{-3}	7.2×10^{-3}	-

$$D_a = \left(\sum \frac{Ci_i}{MPC_i}\right) \text{ (Limiting Dose Rem/Yr)}$$

where: D_a is the individual dose in Rem/yr,

Ci is the Average concentration in Lake Michigan of the individual nuclides measured, in μCi/ml,

MPC is the concentration of each nuclide measured required to produce the limiting dose at continuous intake in μCi/ml and limiting dose is the dose produced at continuous exposure to MPC concentrations.

In calculating ingestion dose from the consumption of fish, an equation similar to the one used for drinking water dose is used except that a standard daily diet of 50 grams of fish flesh is used in contrast to the 2,200 ml of fluid consumed daily by the standard man. This, in effect, alters the MPC_i by 50/2,200 or 0.0227.

The calculation of individual doses, both from drinking water and consuming fish, are per the previous formula while integrated population doses in man-Rem are calculated utilizing the following parameters:

a. For drinking water, the individual doses are summed over the entire population that receives its drinking water from Lake Michigan with discharge canal flow appropriately mixed with the lake. This is approximately 10 million people of which approximately 7 million reside in the Chicago metropolitan area.

b. The population dose due to drinking water to Charlevoix residents is based on a population of 3,500 people.

c. *For fish consumption, the average concentration in Lake Michigan water, resulting from plant releases, is used with a bioaccumulation factor to determine the average concentration in fish.

d. Fish do not reside continuously in the discharge canal but migrate. This can be seen in the following table which compares the fish consumption dose based on the discharge canal water concentration

*ERG Special Report No 2, "Trace Element Distributions in Lake Michigan Fish: A Baseline Study With Calculations of Concentration Factors and Equilibrium Radioisotope Distributions," March 1973.

and the appropriate reconcentration factors to the fish consumption dose calculated from actual concentrations in fish caught in or near the discharge canal.

Population doses based upon drinking water from the Charlevoix municipal system were 0.05 person-Rem and total Lake Michigan drinking water consumption population dose was 1.9 person-Rems. The consumption of all of the Lake Michigan fish harvested resulted in a population dose of 0.37 person-Rem.

As a measure of total environmental impact, the radioactive liquid releases from the plant are averaged over the entire lake and then used to determine the population dose from fish caught throughout the entire lake and total water consumed from the lake.

Both of the dose calculations are conservative in that:

(1) Equilibrium is not obtained in the human body for most isotopes released.

(2) No credit is taken for precipitation and deposit in sediment or uptake by life forms other than fish which are seen to occur by the data shown in Appendix D.

(3) No credit is taken for radioactive decay which for I-131 is significant.

Results are shown in the following tables.

CONSUMERS POWER COMPANY
BIG ROCK NUCLEAR POWER PLANT
CALCULATED RADIATION DOSES

FROM LIQUID EFFLUENTS - POPULATION DRINKING WATER DOSE
1/1/73 TO 12/31/73

VECTOR	ISOTOPE	MPC	CRITICAL ORGAN	CURIES RELEASED	AVG CONCENTRATION IN LAKE MICHIGAN (UCI/ML)	CI/MPCI	(CI/MPCI)MPDI (MREM/YR)	POPULATION DOSE (MAN-REM)	POPULATION DOSE CHARLEVOIX,MICH (MAN-REM)
WATER	ZN-65	1.00E-04	WHOLE BODY	0.0008	1.57E-16	1.57E-12	7.85E-10	0.00001	1.93E-07
WATER	I-131	3.00E-07	THYROID	0.0594	1.24E-14	4.12E-08	2.06E-05	0.20612	5.06E-03
WATER	CS-134	9.00E-06	WHOLE BODY	0.2493	5.19E-14	5.77E-09	2.89E-06	0.02886	7.08E-04
WATER	CS-137	2.00E-05	WHOLE BODY	0.5197	1.08E-13	5.41E-09	2.71E-06	0.02707	6.64E-04
WATER	BALA-140	2.00E-05	G.I. TRACT	0.0004	8.96E-17	4.48E-12	6.72E-09	0.00007	1.65E-06
WATER	CO-58	1.00E-06	G.I. TRACT	0.0284	5.92E-15	5.92E-09	8.88E-06	0.08884	2.18E-03
WATER	CO-60	3.00E-05	G.I. TRACT	0.2153	4.49E-14	1.50E-09	2.24E-06	0.02243	5.50E-04
WATER	OTHERS	3.00E-07	WHOLE BODY	0.9669	2.01E-13	6.71E-07	1.81E-04	1.81287	4.45E-02
						TOTAL	WHOLE BODY	1.86880	4.58E-02
							THYROID	0.20612	5.06E-03
							G.I. TRACT	0.11134	2.73E-03
							BONE	0.0	0.0

(1) Average concentration in Lake Michigan = curies released/volume of Lake Michigan
Volume of Lake Michigan 4.8 x 10^{-15} liters.

(2) Based on a fluid intake of 1200 ml/day.

(3) Population taking its drinking water from Lake Michigan is approximately 10,000,000 people with 7,000,000 in the Chicago area.

(4) Using average concentration in discharge canal diluted by 800.

(5) 10 CFR 20 MPC for unknown mixture with certain isotopes not present.

(6) This compares to a background and medical radiation dose of 0.215 Rem/yr/person or 2.15 x 10^{6} man-Rems for the population taking its drinking water from Lake Michigan.

DATE 01/18/74

CONSUMERS POWER COMPANY
BIG ROCK NUCLEAR POWER PLANT
CALCULATED RADIATION DOSES

FROM LIQUID EFFLUENTS - FISH CONSUMPTION DOSE
1/1/73 TO 12/31/73

VECTOR	ISOTOPE	MPCI	CRITICAL ORGAN	BIOACCUMULATION FACTOR	AVG CONCENTRATION IN LAKE MICHIGAN (UCI/ML)	AVG CONCENTRATION IN FISH (UCI/G)	(CFI/MPCI)MPDI (MREM/YR)	POPULATION DOSE (MAN-REM)
FISH	ZN-65	4.40E-03	WHOLE BODY	900.	1.57E-16	1.41E-13	1.61E-08	0.00001
FISH	I-131	1.32E-05	THYROID	500.	1.24E-14	6.18E-12	2.34E-04	0.13913
FISH	CS-134	3.96E-04	WHOLE BODY	2360.	5.19E-14	1.23E-10	1.55E-04	0.09195
FISH	CS-137	8.80E-04	WHOLE BODY	2360.	1.08E-13	2.55E-10	1.45E-04	0.08623
FISH	BALA-140	8.80E-04	G.I. TRACT	365.	8.96E-17	3.27E-14	5.58E-08	0.00003
FISH	CO-58	4.40E-05	G.I. TRACT	330.	5.92E-15	1.95E-12	6.66E-05	0.03958
FISH	CO-60	1.32E-03	G.I. TRACT	330.	4.49E-14	1.48E-11	1.68E-05	0.00999
FISH	OTHERS	1.32E-05	WHOLE BODY	80.	2.01E-13	1.61E-11	3.30E-04	0.19579
					TOTAL	WHOLE BODY	6.30E-04	0.37398
						THYROID	2.34E-04	0.13913
						G.I. TRACT	8.35E-05	0.04960
						BONE	0.0	0.0

(1) Maximum Permissible Concentration for Fish = $MPC_1 \cdot (2200/50)$.

(2) ERG Special Report No 2.

(3) Using 23,873,689 pounds of fish harvested from Lake Michigan in 1970. This number includes both commercial and sports catches as shown in Appendix D minus alewives which are not generally consumed.

(4) This compares to an average background and medical radiation dose of 0.215 Rem/yr/person or 1.3×10^5 man-Rems for the population necessary to consume the Lake Michigan fish catch at a rate of 50 g/day/person.

DATE 01/18/74

Big Rock Pt.
July - Dec 73 S. A Report
Grand Total
159.275

IX

Occupational Exposure
(7.2.2.1.2.a(1)(h))

Number of Persons Within Exposure Range

mRem Dose	7/ 2/73 - 7/29/73 (2.25)				7/30/73 - 8/26/73				8/27/73 - 9/23/73			
0-100	*Maint	3	Oper	8	Maint	5	Oper	8	Maint	3	Oper	7
2.25 2.55	Supv	15	Tech	7	Supv	16	Tech	6	Supv	16	Tech	5
2.75	Others	12			Others	15			Others	24		
101-500	*Maint	9	Oper	11	Maint	6	Oper	6	Maint	2	Oper	4
9.3 5.1	Supv	4	Tech	2	Supv	3	Tech	2	Supv	3	Tech	3
6.0	Others	5			Others	0			Others	8		
501-1250	*Maint	2	Oper	0	Maint	3	Oper	5	Maint	6	Oper	7
2.625 8.75	Supv	0	Tech	1	Supv	0	Tech	2	Supv	0	Tech	2
13.125	Others	0			Others	0			Others	0		
1251-2500									Maint	1	Oper	1
2.75									Supv	0	Tech	0
									Others	0		
Total Number of People Badged		79				77				92		

mRem Dose	9/24/73 - 10/28/73				10/29/73 - 11/25/73				11/26/73 - 12/30/73			
0-100	*Maint	5	Oper	8	Maint	2	Oper	10	Maint	2	Oper	10
2.8 3.15	Supv	16	Tech	5	Supv	16	Tech	5	Supv	15	Tech	2
2.85	Others	22			Others	30			Others	18		
101-500	*Maint	8	Oper	7	Maint	8	Oper	8	Maint	8	Oper	8
9.3 9.3	Supv	3	Tech	3	Supv	3	Tech	1	Supv	1	Tech	7
12.3	Others	10			Others	11			Others	17		

Others include office secretaries, General Office personnel, contract personnel, vendors, plant guards, information center personnel, Region repairmen other than from Traverse City and visitors.

*Maint includes Region repairmen from Traverse City.

Big Rock Pt
July - Dec 73 S. A Report
Grand Total
159.275

IX

Occupational Exposure
(7.2.2.1.2.a(1)(h))

Number of Persons Within Exposure Range

mRem Dose	7/ 2/73 - 2.25 7/29/73		7/30/73 - 8/26/73		8/27/73 - 9/23/73	
0-100	*Maint 3	Oper 8	Maint 5	Oper 8	Maint 3	Oper 7
2.25	Supv 15	Tech 7	Supv 16	Tech 6	Supv 16	Tech 5
2.55 2.75	Others 12		Others 15		Others 24	
101-500	*Maint 9	Oper 11	Maint 6	Oper 6	Maint 2	Oper 4
9.3	Supv 4	Tech 2	Supv 3	Tech 2	Supv 3	Tech 3
5.1 6.0	Others 5		Others 0		Others 8	
501-1250	*Maint 2	Oper 0	Maint 3	Oper 5	Maint 6	Oper 7
2.625	Supv 0	Tech 1	Supv 0	Tech 2	Supv 0	Tech 2
8.75 13.125	Others 0		Others 0		Others 0	
1251-2500					Maint 1	Oper 1
2.75					Supv 0	Tech 0
					Others 0	
Total Number of People Badged	79		77		92	

mRem Dose	9/24/73 - 10/28/73		10/29/73 - 11/25/73		11/26/73 - 12/30/73	
0-100	*Maint 5	Oper 8	Maint 2	Oper 10	Maint 2	Oper 10
2.8	Supv 16	Tech 5	Supv 16	Tech 5	Supv 15	Tech 2
3.15 2.85	Others 22		Others 30		Others 18	
101-500	*Maint 8	Oper 7	Maint 8	Oper 8	Maint 8	Oper 8
9.3	Supv 3	Tech 3	Supv 3	Tech 1	Supv 1	Tech 7
9.3 12.3	Others 10		Others 11		Others 17	

Others include office secretaries, General Office personnel, contract personnel, vendors, plant guards, information center personnel, Region repairmen other than from Traverse City and visitors.

*Maint includes Region repairmen from Traverse City.

Number of Persons Within Exposure Range (Contd)

mRem Dose	9/24/73 - 10/28/73		10/29/73 - 11/25/73		11/26/73 - 12/30/73	
501-1250	*Maint 3	Oper 3	Maint 1	Oper 10	Maint 5	Oper 0
11.375 14.875	Supv 0	Tech 2	Supv 0	Tech 2	Supv 1	Tech 1
17.5	Others 5		Others 4		Others 13	
1251-2500			Maint 0	Oper 0	Maint 0	Oper 0
5.5 15.125			Supv 0	Tech 2	Supv 1	Tech 0
			Others 2		Others 10	
Total Number of People Badged	100		105		119	

The number of persons that received more than 2500 mRem during 1973 was 33 and the major causes were as follows:

1. Lining of the spent fuel pool with stainless steel.
2. Refueling shutdown.
    a. Head removal and replacement.
    b. Steam drum spool piece flange.
    c. Insulation removal and cleanup.
    d. Weld inspection.
    e. Clean-up demineralizer drain valve.
    f. Changing of rod drives.
    g. Recirculating pump overhaul.
    h. Primary leak detection system.
    i. Limitorque motor operator adjustment.
3. Routine maintenance.
4. Routine plant surveillance and inspection.

Others include office secretaries, General Office personnel, contract personnel, vendors, plant guards, information center personnel, Region repairmen other than from Traverse City and visitors.

*Maint includes Region repairmen from Traverse City.

X. RADIOACTIVE LEVELS IN PRINCIPLE FLUID SYSTEMS

	Minimum	Average	Maximum
A. Primary Coolant			
Reactor Water Filtrate[(a)] µCi/ml	1.5×10^{-2}	9.5×10^{-2}	2.9×10^{-1}
Reactor Water Crud[(c)] µCi/ml/Turbidity Unit	2.9×10^{-3}	1.0×10^{-1}	2.5×10^{-1}
Iodine Activity µCi/ml	5.0×10^{-4}	3.1×10^{-2}	2×10^{-1}
B. Reactor Cooling Water System			
Reactor Cooling Water[(a)] µCi/ml	4.4×10^{-3}	2.9×10^{-2}	4.4×10^{-2}
C. Spent Fuel Pool			
Fuel Storage Pool[(a)]	1.2×10^{-4}	8.8×10^{-4}	2.9×10^{-2}
Fuel Pool Iodine	3.0×10^{-7}	9×10^{-6}	2×10^{-4}

[(a)] A counter efficiency based on a decay scheme consisting of one gamma photon per disintegration at 0.662 MeV used to convert count rate to microcuries. All count rates were taken two hours after sampling.

[(b)] Based on efficiency of Iodine 131 two hours after sampling.

[(c)] Based on APHA turbidity units and 500 ml of filtered sample.