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PACIFIC GAS AND ELECTRIC COMPANY

DOCKET NO. 50-133

REPORT ON THE OPERATION OF HUMBOLDT BAY POWER PLANT UNIT NO. 3, COVERING THE PERIOD JANUARY 1, 1973 THROUGH JUNE 30, 1973

This report describes the six-month operating period for Humboldt Bay Power Plant Unit No. 3 from January 1 through June 30, 1973, and is submitted in accordance with Section IX.H.1.b. of the Technical Specifications.

A. Operations Summary

Operating Cycle 8 was in progress at the beginning of the report period with the Unit operating at reduced load of 170 MWt, 52 MWe. The load reduction was the result of a 13 MWe curtailment taken on December 18 so that reactor pressure could be reduced from 1100 to 1020 psig to minimize primary system leakage. The Unit was shut down on January 14, 1973 (Outage 73-1, 4.8 days) to correct this problem, which was found to be the result of packing leakage on the reactor vent valves. The primary system leakage rate had increased to 0.10 gpm when the decision was made to shut down the Unit. The Unit was returned to service at rated power of 210 MWt, 65 MWe, on January 19, 1973, following the repair work.

Following the Unit startup on January 19, water again began to accumulate slowly in the lower drywell. Analyses indicated that the accumulation was probably condensed reactor steam and it was concluded that the reactor vent valves were again leaking at the stem packing. Lower drywell head water accumulation surveillance was increased and the leakage rate calculated daily and plotted. The leakage rate increased from 0.004 gpm to 0.17 gpm on June 12 when the Unit was again shut down.

End-of-life coastdown with all control rods out was begun on May 22. By the end of the report period, the Unit's capacity was reduced to 57 MWe due to fuel depletion.

On June 12, the Unit was separated from the system to conduct routine turbine overspeed trip tests, to conduct reactor operator AEC license examinations, and to correct the primary system leakage problem (Outage 73-2, 3.07 days). The overspeed test was successfully completed on June 12. The reactor was brought critical five times on June 13 for practice and AEC license examination demonstrations. On the last demonstration, a reactor short period trip occurred as the reactor was being manuevered at low power when a control rod double-notched on a withdraw signal. The source of primary system leakage was verified to be the reactor vent valve packing and the valves were again repacked. The Unit was returned to service on June 15. For a few days following startup, zero leakage rate was measured, then a small amount of water accumulation in the lower drywell head was again noted. The rate of accumulation had increased to 0.02 gpm by the end of the report period. Again, the reactor vent valve packing is believed to be the source of this leakage. The different types of packing suitable for this service are being studied to determine whether a more reliable packing is available.

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1. Changes in Facility Design

All 10 CFR 50.59(a) changes in the facility design are described in Section A.6 below. All 10 CFR 50.59(b) changes made in the facility design during the report period are described in Appendix I.

2. Significant Performance Characteristics

The off-gas release rate normalized to rated power increased from approximately 12,000 to approximately 19,000 μCi/sec during the report period. This increase correlated with the shift in the distribution of fission products in the gaseous radwaste from pure recoil toward an equilibrium mixture and indicates some small quantity of fuel cladding failures. A program of dry sipping to locate and remove the failed fuel will be conducted during the regularly scheduled fall refueling outage.

During the report period, 40 new fuel elements for the next fuel loading were received from Exxon Nuclear Company and inspected. All of the new fuel was found to be within specifications except for one element, which was returned because of minor damage to a spacer (damage apparently occurred in transit).

The reactor cleanup pump motor failed on May 20. This failure was considered to be an unusual event as described in the Technical Specifications and was therefore reported to the Commission in our letter of June 18.

3. Changes in Operating Methods Which Were Necessitated by (1) and (2) or Which Otherwise Were Required to Improve the Safety of Facility Operations

None.

4. Results of Surveillance Tests and Inspections Required by the Technical Specifications

These tests and inspections are discussed under the Routine Operational Testing section of Appendix II.

5. Results of Any Periodic Containment Leak Rate Tests Performed During the Report Period

No integrated leak rate tests were performed during the report period. Results of the continuous leakage rate monitoring program are summarized in Appendix II.

6. Brief Summary of Those Changes, Tests, and Experiments Requiring Authorization from the Commission Pursuant to 10 CFR 50.59(a)

On January 25, Proposed Change No. 45 was submitted requesting an increase in the maximum permissible reactor vessel hydrostatic test pressure from 575 psig to 800 psig at a vessel metal temperature of 120°F. On February 13, Change No. 43 was issued by the Commission authorizing this proposed change for an interim period not to exceed one year. On March 2, Change No. 44 was issued by the Commission which authorizes changes to the Technical Specifications to incorporate reporting requirements consistent with current regulatory practice. Amendment No. 3 to the Facility License was also issued deleting from the license those record keeping and reporting requirements now included in the Technical Specifications.

On May 17, Proposed Change No. 46 was submitted requesting authorization to use Type IV - Batch 1 Exxon fuel for refueling the reactor at the beginning of Operating Cycle 9. This proposed change had not been approved by the Commission by the end of the report period.

No other 10 CFR 50.59(a) changes, tests, or experiment requests were made during the report period.

7. Changes in the Plant Operating Staff for Those Positions Which Are Designated as Key Supervisory or Technical Personnel

On March 1, Mr. O. E. Sundquist, a Shift Foreman, was transferred to the Diablo Canyon site. His position will eventually be filled by promoting a Senior Control Operator with an AEC Senior Operator's License. In the meantime, his position is being covered by the relief Shift foreman.

Mr. F. M. Reinhart, Engineering Trainee at the Humboldt Bay Power Plant, took his AEC Reactor and Senior Reactor Operator License examinations in June. He has received his Senior Operator License.

As a matter of information, on June 3, Messrs. C.A. Bartlett (Senior Control Operator), R.L. Ewing (Control Operator) and T. J. Martin (Control Operator) transferred to the Diablo Canyon site. Their positions were filled by upgrading plant personnel. Also, three Auxiliary Operators took their AEC Reactor Operator License examinations in June. All three men have received their licenses.

B. Power Generation

During the report period, gross thermal and electric power generation totaled 829,094.4 MWt hours and 249,482 MWe hours, respectively. Station use totaled 8,484 MWe hours, resulting in a net electric generation of 240,998 MWe hours.

During the report period, the reactor was brought critical seven times and was critical a total of 4,168 hours. Reactor on-the-line hours (periods when the reactor was critical and supplying steam to the turbine or to the condenser through the bypass valves) totaled 4,158 hours. Turbine service hours (periods when the turbine-generator was paralleled to the system) totaled 4,155 hours.

The Unit's capacity and availability factors for the report period were .884 and .957, respectively.

A histogram of thermal power versus time is presented in Appendix III.

C. Shutdowns of Facility

A total of two outages, both scheduled, occurred during the report period. The outages are described in detail in Appendix IV. The first outage was taken to determine and correct the source of leakage into the lower drywell. The second outage was for the purpose of conducting routine turbine overspeed trip tests, AEC reactor operator license demonstration examinations, and determining and correcting the source of leakage into the lower drywell. The total outage time during the report period was 7.85 days.

D. Maintenance

The principal corrective maintenance performed during the reporting period is described in Appendix V.

E. Changes, Tests and Experiments

Changes in the facility design and/or procedures as described in the FHSR which were made during the report period without prior Commission approval, pursuant to 10 CFR 50.59(b), are described in Appendix I.

Significant tests and experiments carried out during the report period were as follows:

1. Core Thermal Hydraulic Measurements
2. Containment Continuous Leakage Rate Monitoring
3. Routine Operational Testing
4. Control Rod Drive B-3 Reactor Trip Insertion Test
5. Control Rod Drive Hydraulic Return to the Reactor Test

These tests are described in detail in Appendix II.

F. Radioactive Effluent Releases

1. Gaseous Effluents

The monitoring systems associated with the emergency condenser and liquid radwaste system vents to atmosphere showed that no detectable releases of radioactive gases occurred during the report period. Therefore, all detectable gaseous radioactive waste releases were made via the 250 foot stack as discussed below.

a. Noble and activated gases - The release of gaseous radioactive wastes has been monitored continuously by the air ejector off-

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gas and stack gas monitoring systems. The calibration of these monitors for noble and activated gases has been checked by periodic analyses of "grab" samples on a multichannel gamma scintillation spectrometer. The releases of the radioactive noble and activated gases during each month of the report period are summarized in Appendix VI, Table I. The average release rate during each month was considerably less than the annual average release rate limit and peak release rates during the report period were considerably below the instantaneous release rate limit at which isolation of the off-gas discharge line occurs. The current license limits are 50,000 μCi per second for an annual average release rate and 500,000 μCi per second for an instantaneous release.

b. Halogen and particulate - The halogen and particulate filters, which are part of the stack gas monitoring system, have been removed weekly for counting. The release of halogens and particulates during each month of the report period are summarized in Appendix VI, Table I. The average release rate during each month was below the annual average release rate limit. The license limit for the annual average release rate (based on a conservative permissible concentration of 3×10^{-10} μCi per cc) is 0.18 μCi per second.

2. Liquid Effluents

The activity in each batch of liquid radioactive waste was either in solution at the time of discharge or the batch was filtered prior to discharge. Radioactive waste contributed by the operation of the Unit resulted in an average concentration in the plant discharge canal during the report period of 1.150×10^{-8} μCi per ml. The concentration for all discharges was less than 1.0×10^{-6} μCi per ml when averaged over all seven consecutive day periods. Waste discharge operations during the year were carried out in a manner such that the annual average concentration was less than 1.0×10^{-7} μCi per ml. This is in accordance with the liquid waste discharge requirements issued by the State of California North Coastal Regional Water Quality Control Board and with 10 CFR 20 requirements. Analysis of weekly composite samples from the plant effluent canal and monitoring by the liquid waste discharge monitor confirmed that no unaccounted release of radioactive waste occurred during the report period. The quantity of each of the principal radionuclides released monthly as liquid waste during the report period is specified in Appendix VI, Table II.

G. Solid Radioactive Waste

During the report period, 459 cubic feet of low level solid radwaste totaling approximately 1.1 Curies of activity and 165 cubic feet of high level solid radwaste totaling approximately 1.8 Curies was packaged. The total amount of solid radwaste packaged was 624 cubic feet, totaling

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approximately 2.9 Curies. Packaged low level radwaste was stored in the low level radwaste storage building and high level radwaste was stored in the underground solid waste storage vaults.

Solid radwaste shipments are included in Appendix VII, which summarizes all off-site shipments of radioactive material during the report period.

H. Environmental Monitoring

The gamma dose rate as measured by Victoreen ionization chambers at several of the 36 off-site environmental monitoring stations located in the proximity of the plant has been slightly higher than background during the report period. Background for the period of 1/2/73 to 7/2/73 was measured as 45.4 ± 2.8 millirem. The maximum dose measured during this period was 99.8 ± 3.3 millirem (54.4 millirem above background) at station 33, which is located on plant property at the site boundary. The dose measured at station 14 was 80.2 ± 3.3 millirem (34.8 millirem above background) for the same period. These doses are to be compared to the maximum annual off-site dose limit of 500 millirem contributed by the plant.

Appendix VIII contains the data from the plant's environmental monitoring program for the period July 1 to December 31, 1972. The types of media sampled include marine flora and invertebrates, bottom sediment, milk, ground water, air particulate, and gamma radiation dose measurements with dosimeters. This appendix shows that with the exception of the gamma radiation dose measurements discussed above, the levels of radioactive materials in environmental media indicate that public exposures will be less than 1% of those that could result from continuous exposure to the concentrations listed in 10 CFR 20, Appendix B, Table II.

I. Occupational Personnel Radiation Exposure

During the report period, radiation levels within the controlled area of the Unit during power operation essentially have been unchanged from those previously reported.

Contamination levels within the controlled area of the Unit have remained low. No significant spills of radioactive water or long-term airborne activity problems have been encountered during the operation or maintenance of the Unit. Special control measures have been utilized in specific areas for special maintenance or operation involving contaminated equipment or floors.

Film badge exposure results for the exposure period between January 15, 1973 and July 14, 1973 for the plant employees and all other personnel assigned to work in a radiation area (as defined by 10 CFR 20.202b.2) and who also received at least 100 mRem whole body exposure were as follows:

Whole Body Exposure Range - mRem	Duty Function[1]				
	Plant Operations	Plant Maintenance	Radiation Monitoring &Chemistry	Plant Management	Other PG&E Personnel & Non-PG&E Personnel
100 - 499	12	4	1	1	10
500 - 1249	12	8	1	3	2
1250 - 2499	5	7	3	2	6
2500 - 4999	2	0	0	0	0
>5000	0	0	0	0	0
TOTAL	31	19	5	6	18
Max.Exp.-mRem	2810	2050	1620	2300	1800
Avg.Exp.-mRem	993	995	1173	1161	770
Total Man-Rem	30.790	18.905	5.865	6.965	13.990

[1] Defined in Company's letter of 6/8/73 to Mr. D. J. Skovholt.

In addition, film badges were issued to a total of 66 individuals (plant employees, employees temporarily assigned to the plant, or visitors) who received less than 100 mRem whole body exposure during the report period.

APPENDIX I

PRINCIPAL CHANGES MADE IN FACILITY PURSUANT TO 10 CFR 50.59(b)

The changes described below were completed during the January through June 1973 report period. These changes were reviewed by the On-Site Review Committee as required by Section IX-C of the Technical Specifications. None of these changes were found to involve a change in the Technical Specifications or an "unreviewed safety question" as defined in 10 CFR 50.59(c).

1. Off-Gas Filter Vault Sump Pump

During a period of unusually heavy rainfall in January, ground water leaked into the off-gas pipe trench via a guard pipe and accumulated in the off-gas filter vault area. The leak into the trench was stopped by plugging the guard pipe. To prevent a recurrence of this problem, a Burns Model S-23 non-submersible sump pump capable of pumping approximately 50 to 60 gpm was installed in the off-gas filter vault in order to pump any accumulation of ground water and rain water to the base of the stack which in turn drains to the TBDT. The pump is started and stopped automatically by a float switch which senses water level in the filter vault. The pump is powered from a local outlet in the base of the stack.

2. Condensate Demineralizer Regeneration Room "High Radiation Area" Alarm

The "high radiation area" local and remote alarm initiation device was moved from the entrance gate of the condensate demineralizer resin regeneration tank cubicle to the entrance door of the condensate demineralizer room (the cubicle is inside of the room). This change increases the area controlled by this alarm and was required because on occasion the accessible area outside the entrance gate of the resin regeneration tank cubicle becomes a high radiation area during resin regeneration operations. Plant procedures governing access to high radiation areas were always in effect during the relocation of the alarm.

3. Relocation of Radwaste Discharge Process Monitor Detector

The scintillation detector for the liquid radwaste discharge process monitor was moved from the north wall inside the radwaste building to the roof of the radwaste building in order to place it in an area of lower background radiation. As a result of this move, the background radiation at the detector has been reduced by a factor of approximately 10, thereby increasing the sensitivity of the monitoring system. The function of this monitoring system was not changed by this modification and no liquid radwaste discharges were made during the period of relocation.

4. Test Switch for Diesel Fire Pump Backup Starting Battery

A test switch was installed in series with the undervoltage coil, which senses the voltage of the normal starting battery for the diesel fire pump. This switch allows the simulation of an undervoltage condition on the normal starting battery, which initiates the transfer of the diesel fire pump starting circuit to the backup starting battery. The installation of the switch permits the operational testing of the backup starting battery. The operation of the diesel fire pump is in no way affected by this modification since the pump remains fully operational whether the test switch is in the "normal" or "test" position. Also, the diesel fire pump was always fully operational during the period of the modification.

APPENDIX II

DISCUSSION OF SIGNIFICANT TESTS

This appendix discusses significant tests performed during the report period and presents an analysis of the test results.

1. Core Thermal-Hydraulic Measurements

   During the report period, two flux wire irradiations were performed for incore calibration and critical heat flux ratio (CHFR) calculations. Results of the CHFR calculations indicate operation well above the 1.5 minimum CHFR limit at 125% overpower.

2. Containment Continuous Leakage Rate Monitoring

   During the report period, the continuous leakage rate monitoring system was in service whenever the Unit was in power operation. After the Unit startup on January 23, and until the end of the report period, the average drywell leakage rate was 0.026 W/O per day or less at a nominal gauge pressure of 20 inches of water, while the average suppression chamber leakage rate was 0.008 W/O per day or less at a nominal gauge pressure of 10 inches of water.

   These leakage rates are considerably below the operational leakage rate limit given in the Technical Specifications and compare favorably with leakage rates previously observed.

3. Routine Operational Testing

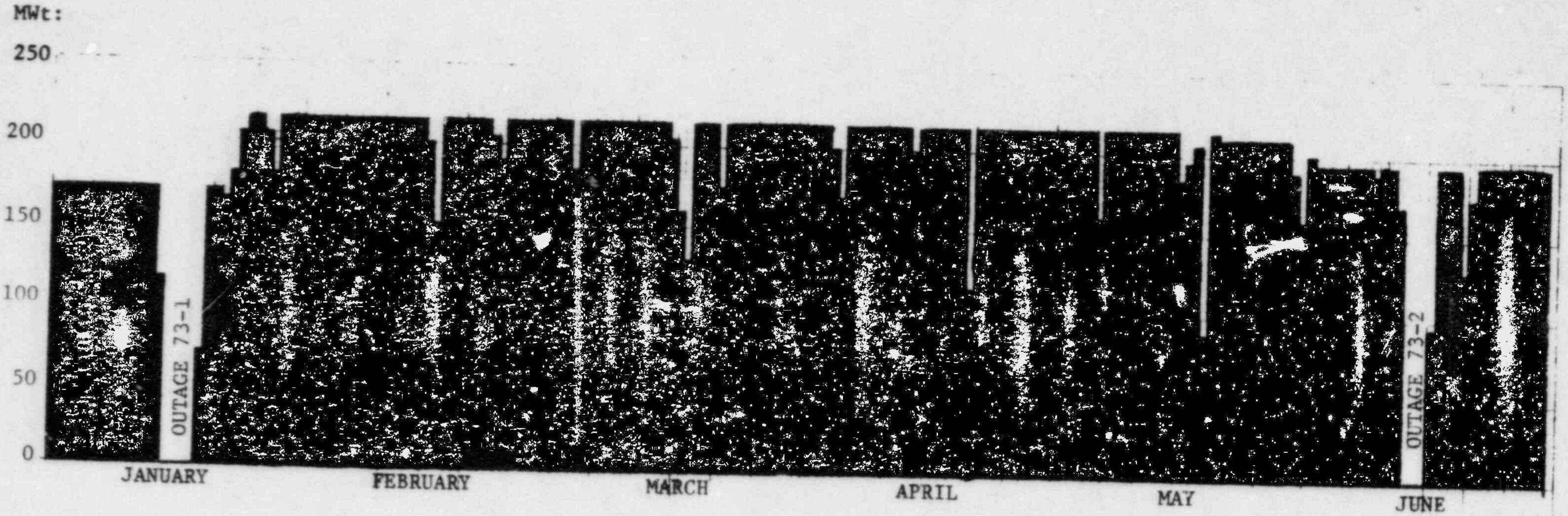
   All routine daily, monthly, and quarterly operational tests listed in Table IX-I of the Technical Specifications have been performed in accordance with established procedures. The results of these tests revealed no significant problems with the nuclear safeguards systems during the report period.

4. Control Rod Drive B-3 Reactor Trip Insertion Test

   Following the replacement of control rod drive B-3 during Outage 73-1, the drive was tested using our existing reactor trip insertion test procedure. The drive performed satisfactorily and was well within the Technical Specification limits on drive insertion time following a reactor trip.

5. Control Rod Drive Hydraulic Return to Reactor Test

   The modification to minimize the thermal stress on nozzle C-2 by rerouting the 100^{o}F control rod drive hydraulic return from nozzle C-2 to the feedwater line had previously been successfully tested up to 55 MWe (refer to 7/1/72 through 12/31/72 Semi-Annual Appendix V), but had not been tested at full load. The full load test was conducted on March 13, 1973 using an approved test procedure. The control rod drives functioned properly during the test and the modification is now being used as the normal control rod drive hydraulic return to the reactor.



HUMBOLDT BAY POWER PLANT UNIT NO. 3
AVERAGE MEGAWATTS THERMAL (MWt) vs. TIME

APPENDIX III

APPENDIX IV

DESCRIPTION OF OUTAGES*

OUTAGE 73-1

Outage Period: 1657 hours on 1/14/73 to 1135 hours on 1/19/73.
Duration of Outage: 4 days, 18 hours, 39 minutes (4.78 days)
Type of Outage: Scheduled

Description of Outage: The Unit was separated from the system to determine the source of leakage into the lower drywell. Water had begun to slowly accumulate in the lower drywell following the last Unit startup on October 23, 1972. Radiochemical analysis indicated that the probable source of this water was condensed reactor steam. A daily plot of the water level in the lower drywell indicated only a slight accumulation until November 9, 1972 when the rate of accumulation started to increase. When the rate of accumulation reached about 0.03 gpm, the Unit load was reduced to 52 MWe and reactor pressure was reduced from 1120 psig to 1020 psig in an attempt to reduce the leakage rate. Some immediate improvement was noted, however the leakage rate continued to increase to 0.1 gpm on January 9, 1973, when the decision was made to shut down the Unit to repair the leak. On January 14, 1973, the Unit was shut down by normal insertion of all control rods even though the leakage rate had decreased to 0.07 gpm in the few days just prior to the shutdown. The reactor was cooled to ambient temperature at the normal cooldown rate.

The lower drywell head was removed at 0900 on January 15, 1973 for visual inspection. Water was observed to be dripping from the upper portion of the drywell; therefore, at 1130 the shield plug and the upper drywell head were removed for visual inspection. A 550 psi hydrostatic test was conducted on the reactor vessel, and leakage was observed at the stem packing of both reactor vent valves. No other leakage was observed. The vent valve packing (Anchor Amflex 900 AH) was found to be completely deteriorated.

Both vent valves were repacked with John Crane 177 AI pressed to chevron form. Neither valve leaked under a 550 psi hydrostatic test performed on January 18, 1973.

The control rod drive located in cell B-3, which was not latching properly, was replaced with a spare drive. The replacement control rod drive was tested and performs satisfactorily.

No. 2 reactor safety valve was disassembled, cleaned and lapped to correct weeping. This condition apparently had existed since a reactor excursion on November 1972. Repairs were satisfactory.

No. 3 and No. 4 drywell fan motors were removed because of grounded stators. Inspection revealed water in the windings so both stators were baked dry and rechecked for grounds. The No. 4 stator was satisfactory, but the No. 3 stator required rewinding. Both motors were cleaned and received new bearings. Following repairs, the motors were reinstalled and operate properly.

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*The principal maintenance performed and principal changes made during these outages are further described in Appendices IV and V. Tests performed during these outages are described in Appendix VI.

The H.P. turbine 8th stage extraction line orifice flange had a steam leak. The flange was disassembled, but no cuts were found on the flange faces. The flexible gasket was replaced. The flange no longer leaks.

The radwaste process monitor detector was relocated from inside the radwaste building to the roof of the building to reduce background radiation at the detector.

Following the repair work, on January 19, 1973, the reactor was brought critical at 0524 hours, and the Unit was paralleled to the system at 1136 hours.

OUTAGE 73-2

Outage Period: 2133 hours on 6/12/73 to 2315 hours on 6/15/73
Duration of Outage: 3 day, 1 hour, 42 minutes (3.07 days)
Type of Outage: Scheduled

Description of Outage: The Unit was separated from the system to conduct a main turbine overspeed test, for AEC license examinations, and to determine the source of leakage into the lower drywell.

The turbine overspeed trip test was satisfactorily conducted on June 12, 1973. The Unit was then shutdown by normal insertion of all control rods. The reactor was brought critical five times on June 13, 1973 for practice and the AEC exam demonstrations. Following the last reactor critical, the reactor tripped on short period when control rod D-6 double-notched on a withdraw signal. The reactor was cooled down at the normal cooldown rate and the shield plug and upper drywell head were removed to look for the source of the leakage into the drywell.

Following the Unit startup on January 19, 1973, water again began to accumulate in the lower drywell. Radiochemical analysis indicated that the water was probably condensed reactor steam. It was believed that the reactor vent valves were again leaking by the stem packing. The leak rate increased from about .004 gpm to almost 0.17 gpm on June 12, 1973 when the reactor was shut down.

The shield plug and upper drywell head were removed at about 1600 on June 13, 1973 to inspect the reactor vent valves for leakage. Visual inspection of the valve insulation revealed that the east valve had definite signs of leakage and that the west valve had slight signs of leakage. Both reactor vent valves were repacked with 3/16" John Crane 177 AI pressed to chevron form. From 2345 on June 13, 1973 to 0035 on June 14, 1973, an 800 psig hydrostatic test was conducted on the reactor. No leakage was visible at the reactor vent valves or elsewhere.

The seats on scram outlet valves No. 1, 7, 14 and 15 were lapped. All seating surfaces were confirmed tight by 360° bluing.

The air ejector root valve stem backseat was lapped and the valve was repacked to correct a steam leak into the pipe tunnel.

Following the repair work, on June 15, 1973 the reactor was brought critical at 1834 hours, and the Unit was paralleled to the system at 2315 hours.

APPENDIX V

PRINCIPAL CORRECTIVE MAINTENANCE PERFORMED

Maintenance performed during the report period consisted mainly of routine maintenance such as lubrication, valve and pump packing and inspection, and servicing of mechanical, electrical and instrument equipment. Specific items of corrective maintenance include the following:

1. Instrument Maintenance

   a. Nuclear Instrumentation - On January 8, the operation of picoammeter No. 1 became "noisy" and it was replaced with the spare. Subsequent repair consisted of vacuum tube replacement and a complete checkout of the picoammeter. On April 23, the picoammeter No. 1 high voltage positive power supply dropped 500 V due to a bad regulating tube The tube was replaced.

   b. Area, Isolation, and Process Monitors - On March 23, it was noticed that the south refueling building isolation monitor indication was decreasing. The indication decrease was caused by a deteriorating vacuum tube. This condition was corrected by replacing the faulty tube. On June 22, the refueling building isolation monitor at the top of the access shaft failed and isolated the refueling building (also sounded evacuation horn and started gas treatment system). The problem was due to a resistor failure which was corrected by replacing the resistor.

      On May 21, the lower area monitor power supply failed due to a bad vacuum tube. It again failed on June 3 due to a degraded resistor in the regulating circuit. Repairs were made in each case by replacing the faulty parts.

      During the discharge of an analyzed batch of radwaste, it was observed that the radwaste process monitor was not responding properly. The problem was traced to the photomultiplier in the detector. Following replacement of the P-M tube, the monitor has responded properly to subsequent releases.

   c. Bypass Valves - During routine exercising of the bypass valves it was discovered that they would not operate. The problem was traced to a frozen servo-motor at the bypass valves. When an attempt was made to operate the valves, the operational amplifier overloaded and blew a control module fuse. The amplifier was overhauled and the servo motor replaced. The replacement of the servo motor is now scheduled for each refueling outage to reduce the possibility of a recurrence of this problem.

2. Electrical Maintenance

   For electrical maintenance performed during Outages 73-1 and 73-2, see Appendix IV.

3. Mechanical Maintenance

a. For mechanical maintenance performed during Outages 73-1 and 73-2 see Appendix IV.

b. Control Rod Drives - The control rod drive which had been removed from cell D-4 during Outage No. 72-8 (because of its tendencey to stick at the full in position) was overhauled. Several scored and broken carbon seals were found as well as a large amount of red oxide. The drive was cleaned and all necessary new parts were installed to restore the drive to a like-new condition. The drive was leak rate tested following established procedures. This drive was then installed in position B-3 during Outage No. 73-1 and has since performed satisfactorily.

The drive removed from cell B-3 during Outage No. 73-1 had experienced latching problems during operation. During disassembly, the collet piston was difficult to remove, requiring the use of a puller. Galling of the piston was noted after removal. Clearance measurements were taken and clearances were minimal. The drive vendor recommended that proper clearances be restored by machining of the housing bore and piston O.D. The drive was reassembled using all necessary new parts to restore the drive to like-new condition.

NDIX VI

REPORT OF RADIOACTIVE EFFLUENTS

TABLE I - AIRBORNE RELEASES

1st Half X 2nd Half _____ Year 1973

	NOTES	UNITS	JAN.	FEB.	MARCH	APRIL	MAY	JUNE	TOTAL
A. Noble Gases									
1. Nuclides Released		Curies							
Kr-89	2		0	42.9	30.2	17.2	11.8	3.5	105.6
Xe-137	2		11.9	293.5	230.0	116.8	78.6	17.6	748.4
Xe-138	1		5556.9	13128.9	13927.9	10876.6	9162.2	5269.7	57922.2
Xe-135m	2		2406.4	4969.9	5323.8	4332.1	3716.0	2218.6	22966.8
Kr-87	1		3116.9	4056.4	4976.9	4541.6	5690.0	5544.8	27926.6
Kr-83m	2		566.0	728.8	912.4	838.2	1042.1	1008.0	5095.5
Kr-88	1		2764.6	3581.5	4467.9	4177.5	5969.2	6416.0	27376.7
Kr-85m	1		763.9	953.1	1202.8	1137.1	1765.6	1996.4	7818.9
Xe-135	1		3686.8	4135.6	5222.0	5428.0	8824.0	10253.6	37550.0
Xe-133m	2		17.8	36.3	49.0	79.0	98.3	95.2	375.6
Xe-133	1		880.6	1052.0	1361.1	2799.9	2886.3	2299.7	11279.6
2. Total Activity Released	3	Curies	19,772	32,979	37,704	34,344	39,244	35,124	199,167
3. Average Release Rate	3	μCi/sec	7382	13,632	14,077	13,250	14,652	13,551	
4. Maximum Release Rate	3	μCi/sec	16,000	21,500	23,200	23,300	45,000	27,000	
5. Percent of Annual Average Release Rate Limit		%	14.76	27.26	28.15	26.50	29.30	27.10	25.47
B. Halogens									
1. Nuclides Released		Curies							
I-131	4		.004156	.007588	.007577	.007651	.007748	.05727	0.091990
I-133	4		.01892	.04611	.03432	.04827	.05201	.09588	0.29557
I-135	5		0	0	0	0	0	0	
2. Total Activity		Curies	.023076	.053698	.041957	.055921	.059758	.015315	0.38756
3. Average Release Rate		μCi/sec	8.62×10^{-3}	2.22×10^{-2}	1.57×10^{-2}	2.16×10^{-2}	2.23×10^{-2}	5.91×10^{-2}	2.48×10^{-2}

APP[illegible]IX VI

REPORT OF RADIOACTIVE EFFLUENTS, TABLE I - AIRBORNE RELEASES (CONTINUED)

1st Half X 2nd Half ____ Year 1973

C. Particulates	NOTES	UNITS	JAN.	FEB.	MARCH	APRIL	MAY	JUNE	TOTAL
1. Nuclides Released		Milli-Curies							
Ba/La-140	4		24.21	6.680	5.162	6.343	21.158	7.388	70.941
Cs-137	7		(0.064)	0.058	(0.064)	(0.364)	0.376	(0.364)	(1.290)
Cs-134	7		(0.003)	0.003	(0.003)	(0.037)	0.038	(0.037)	(0.121)
Sr-89	7		(1.5)	1.465	(1.5)	(2.4)	2.45	(2.4)	(11.62)
Sr-90	7		(0.005)	0.005	(0.005)	(0.005)	0.005	(0.005)	(0.030)
Co-60	7		(0.005)	0.004	(0.005)	(0.131)	0.136	(0.131)	(0.412)
Mn-54	7		(0.003)	0.003	(0.003)	(0.145)	0.150	(0.145)	(0.449)
Zn-65	7		(0.005)	0.004	(0.005)	(0)	0	(0)	(0.014)
2. Total β, γ Activity Released	8	Milli-Curies	(25.8)	(8.22)	(6.75)	(9.43)	(24.3)	(10.5)	(84.9)
3. Average β, γ Release Rate		μCi/sec	9.63×10^{-3}	3.40×10^{-3}	2.52×10^{-3}	3.64×10^{-3}	9.07×10^{-3}	4.05×10^{-3}	5.43×10^{-3}
4. Total Alpha Activity Released	9	μCi	0.130	0.084	0.011	0.043	0.011	0.015	0.294
5. Average Alpha Release Rate		μCi/sec	4.85×10^{-8}	3.47×10^{-8}	4.11×10^{-9}	1.66×10^{-8}	4.11×10^{-9}	5.79×10^{-8}	1.88×10^{-8}
D. Halogen and Particulate Percent of Annual Average Release Rate Limit	6	%	6.21	3.62	2.96	3.66	6.65	14.5	6.28
E. Tritium									
1. Total Activity Released	10	Curies	(0.152)	0.137	(0.152)	(0.139)	0.144	(0.139)	(0.863)
2. Average Release Rate		μCi/sec	$(5.65 \times 10^{-2}$	5.65×10^{-2}	5.65×10^{-2}	5.35×10^{-2}	5.35×10^{-2}	5.35×10^{-2}	5.52×10^{-2}

NOTES:
1. Measured nuclides, proportioanl to total activity released.
2. Estimated from distribution of measured nuclides, proportional to total activity released.
3. Determined from stack gas monitor readout. Calibration of stack gas monitor based upon off-gas sample.
4. Determined from analyses of cartridge/filter in stack gas sample lines which is changed weekly.
5. Determined quarterly from 1 halogen cartridge (see 4 above).
6. Based upon annual average release rate limit as defined in Section VIII-B-3 of the plant's Technical Specifications.
7. Determined quarterly from a composite of 4 consecutive filters (see 4 above).
8. Determined by summing nuclides in C.1.
9. Determined quarterly from 1 filter (see 4 above).
10. Determined quarterly.

N.M. Not Measured ( ) Numbers in parenthesis represent best estimate based upon quarterly determinations.

APPENDIX VI

REPORT OF RADIOACTIVE EFFLUENTS

TABLE II - LIQUID RELEASES

1st Half X 2nd Half ____ Year 1973

	NOTES	UNITS	JAN.	FEB.	MARCH	APRIL	MAY	JUNE	TOTAL
A. Volume of Waste Discharged		Liters	158,504	196,120	121,703	122,558	175,280	145,488	919,653
B. Volume of Dilution Water		Liters	1.53 x 10^{10}	1.43 x 10^{10}	1.54 x 10^{10}	1.26 x 10^{10}	1.40 x 10^{10}	1.57 x 10^{10}	8.73 x 10^{10}
C. Gross Radioactivity (β, γ)									
1. Nuclides Released		Milli-Curies							
Ce-144	1		6.78	9.01	11.37	10.89	28.59	23.22	89.86
Mn-54	1		23.06	22.25	7.10	26.25	1.79	3.10	83.55
Co-60	1		15.53	22.78	3.69	27.22	4.36	2.55	76.13
Zn-65	1		16.00	7.36	3.75	0.90	17.06	7.45	52.52
Cs-134	1		19.10	52.28	18.20	27.39	50.40	17.43	184.80
Cs-137	1		34.80	83.76	27.72	30.81	91.97	29.45	298.51
I-131	1		10.70	10.45	3.77	2.44	10.91	27.70	65.97
Co-58	1		4.71	3.72	1.04	0.17	0.74	0.01	10.39
Ba/La-140	1		2.82	22.78	0.00	0.00	0.08	0.52	26.20
Sr-89	2		5.20	14.10	4.90	4.01	7.38	2.56	38.15
Sr-90	2		0.98	2.69	0.95	1.21	2.22	0.77	8.82
Additional But Otherwise Unspecified Beta Emitters			0.00	9.73	3.44	0.44	0.00	0.03	13.64
I-133	1		0.00	0.00	0.89	0.39	1.98	1.46	4.72
2. Total Activity Released	3	Milli-Curies	139.68	260.91	86.82	132.12	217.48	116.25	953.26
3. Average Concentration Released		μCi/ml	0.913x 10^{-8}	1.824x 10^{-8}	0.564x 10^{-8}	1.048x 10^{-8}	1.553x 10^{-8}	0.740x 10^{-8}	1.092x 10^{-8}
4. **Maximum Daily Concentration**		μCi/ml	**7.190x 10^{-8}**	**11.270x 10^{-8}**	**4.710x 10^{-8}**	**22.970x 10^{-8}**	**15.949x 10^{-8}**	**9.667x 10^{-8}**	
5. Percent of limit for Gross β, γ Activity Released	4	%	9.13	18.24	5.64	10.48	15.53	7.40	10.92
D. Tritium									
1. **Total Activity Released**	2	Curies	8.213	10.162	6.306	4.806	6.873	5.705	42.065
2. Average Concentration		μCi/ml	5.37x 10^{-7}	7.11x 10^{-7}	4.09x 10^{-7}	3.81x 10^{-7}	4.91x 10^{-7}	3.63x 10^{-7}	4.82x 10^{-7}

APPENDIX VI

REPORT OF RADIOACTIVE EFFLUENTS, TABLE II - LIQUID RELEASES (CONTINUED)

1st Half X 2nd Half ____ Year 1973

	NOTES	UNITS	JAN.	FEB.	MARCH	APRIL	MAY	JUNE	TOTAL
3. Percent of Limit for tritium	5	%	0.054	0.071	0.041	0.038	0.049	0.036	0.048
E. Gross Activity (Alpha Only)									
1. Total Activity Released	2	Micro-Curies	14.27	17.65	10.95	13.85	19.81	16.44	92.97
2. Average Concentration		μCi/ml	9.32×10^{-13}	1.23×10^{-12}	7.11×10^{-13}	1.15×10^{-12}	1.41×10^{-12}	1.05×10^{-12}	1.06×10^{-12}
3. Percent of limit for alpha	6	%	0.003	0.004	0.002	0.004	0.005	0.004	0.004

NOTES:
1. Determined from batch analysis except for laundry wastes, which are determined from monthly composite sample.
2. Determined quarterly from composite of all wastes dumped during quarter. Numbers in parenthesis represent best estimate based upon these quarterly determinations.
3. Determined by summing gross radioactivity of specific nuclides and additional but otherwise unspecified nuclides activity.
4. Limit is 10^{-7} μCi/ml when averaged over each calendar year per North Coast Regional Water Quality Control Board.
5. Limit is 10^{-3} μCi/ml when averaged over each calendar year per North Coast Regional Water Quality Control Board.
6. Limit is 3×10^{-8} μCi/ml per 10 CFR 20.

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APPENDIX VII

OFF-SITE SHIPMENTS OF RADIOACTIVE MATERIALS

Off-site shipments of radioactive materials during the report period are listed below:

Shipment Number	Date of Shipment	Transfer License Number From	To	
1	1/2/73	DPR-7	Prime Contractor for AEC (10 CFR 30.12) Lawrence Radiation Laboratory Livermore, Ca.	32 liters of liquid radwaste containing MFP*, Co-60, Mn-54 < 30 μCi Total.
2	1/24/73	DPR-7	0017-59 (Calif.) General Electric Co. Vallecitos Nuclear Center	Activated control rod drive flange bolts for metallurgic examination Co-60, Mn-54, Fe-59, < 0.5 mCi Total.
3	2/7/72	DPR-7	0418-59 (Calif.) General Electric Co. Atomic Power Equipment Department San Jose, Ca.	Compensated ion chamber, Activated and Contaminated Cs-137, Mn-54, Co-60, Sc-46 Ag-110m, Cs-134, Co-58 < 0.6 mCi Total.
4	2/15/73	DPR-7	Prime Contractor for AEC (10 CFR 30.12) Lawrence Radiation Laboratory Livermore, Ca.	32 liters of liquid radwaste containing MFP, Co-60, Mn-54 < 300 μCi Total.
5	4/3/73	DPR-7	Prime Contractor for AEC (10 CFR 30.12) Lawrence Radiation Laboratory Livermore, Ca.	0.3 liters of liquid radwaste containing MFP, Co-60, Mn-54 < 1 μCi Total.
6	4/6/73	DPR-7	Prime Contractor for AEC (10 CFR 30.12) Lawrence Radiation Laboratory Livermore, Ca.	45 liters of liquid radwaste containing MFP, Co-60, Mn-54 < 50 μCi Total.
7	4/11/73	DPR-7	4-3766-1 Nuclear Engineering Company P. O. Box 638 Richland, Wash.	15 drums (55 gal.) containing solid radioactive waste < 3 Ci per drum. Co-60, Mn-54, Zn-65. 2.5 Ci Total.

* MFP - mixed fission products

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Shipment Number	Date of Shipment	Transfer License Number From	To	Radioactive Material
8	4/24/73	DPR-7 SNM 1078 ADM.1	SNM-960 General Electric Co. Vallecitos Nuclear Center Pleasanton, Ca.	Irradiated fuel rods. (Primarily MFP with smaller quantities of fissile material and activation products*.) < 30,000 Ci Total. (369 g fissile material.)
9	4/26/73	DPR-7 SNM 1078 ADM.1	SNM-960 General Electric Co. Vallecitos Nuclear Center Pleasanton, Ca.	Irradiated fuel rods. (Primarily MFP with smaller quantities of fissile material and activation products*.) < 30,000 Ci Total. (314 g fissile material.)
10	4/28/73	DPR-7 SNM 1078 ADM.1	SNM-960 General Electric Co. Vallecitos Nuclear Center Pleasanton, Ca.	Irradiated fuel rods. (Primarily MFP with smaller quantities of fissile material and activation products*.) < 30,000 Ci Total. (224 g fissile material.)
11	6/15/73	DPR-7	0418-59 (Calif.) General Electric Co. Atomic Power Equipment Department San Jose, Ca.	Contaminated solenoid valve. Co-60, Mn-54 < 0.05 μCi Total.
12	6/15/73	DPR-7	0418-59 (Calif.) General Electric Co. Atomic Power Equipment Department San Jose, Ca.	18 "Empty" fuel shipping containers. MFP, Co-60 Mn-54. < 10 μCi Total.
13	6/25/73	DPR-7	0418-59 (Calif.) General Electric Co. Atomic Power Equipment Department San Jose, Ca.	9 incore fission chambers. MFP, U-235, activation products. < 20 Ci Total. (0.24 g fissile material.)
14	6/28/73	DPR-7	SNM-1227 Exxon Nuclear Co. Richland, Wash.	8 "Empty" fuel shipping containers. MFP, Co-60 Mn-54. < 10 μCi Total.
15	6/29/73	DPR-7	SNM-1227 Exxon Nuclear Co. Richland, Wash.	7 "Empty" fuel shipping containers. MFP, Co-60, Mn-54. < 10 μCi Total.
16	6/29/73	DPR-7	SNM-1227 Exxon Nuclear Co. Richland, Wash.	1 new fuel assembly. U-235 < 0.02 Ci Total. (1,664 g fissile material.)

* Activation Products - typically Co-60, Mn-54 and Fe-59.

APPENDIX VIII

ENVIRONMENTAL MONITORING

Quarterly reports "Environmental Radiation Study in the Vicinity of Humboldt Bay Power Plant (HBPP), Eureka, California" contain the basic data from the Humboldt Bay Power Plant environmental monitoring program. The most recent Reports Nos. 46 and 47 are attached and include data from the summer and fall of 1972. These reports describe the sampling locations, total number of samples for each media sampled, as well as the associated measured levels of radioactivity. The types of media sampled are marine flora, marine invertebrates, bottom sediment, milk, domestic water, air particulate, and external radiation measurements with ion chambers.

Potential public exposure in the environs of the plant was calculated from dosimetry data and those sampling media that could result in exposure pathways to man. These sampling media include the aquatic species (gaper clams, Pacific oysters, rock crab), milk, domestic water, and air particulate.

It was shown that with the exception of the direct exposure measurements from ion chambers, the levels of radioactive materials in environmental media indicate that public exposures were less than one percent of those that could have resulted from continuous exposure to the concentrations listed in Appendix B, Table II, Part 20. For the ion chamber measurements, the exposure when extrapolated to annual rates for the station recording the highest reading, was 106.7 millirem/year, or 23.6 millirem/year above background. This measurement was well within the technical specifications limit of 500 millirem/year above natural background.

1. Potential Exposure from Aquatic Media

The aquatic dose model used was taken from ICRP Publication 2, Report of Committee II, "Permissible Doses for Internal Radiation." The samples of gaper clams and rock crabs, which can be taken by sports fishermen, were collected in the vicinity of the Plant discharge. Pacific oysters, which are taken commercially from Humboldt Bay, were collected from the North Bay at Station 65. It was assumed that an individual would consume 20 grams per day of each species. If particular isotopes were identified by the gamma scans of the samples, then these isotopes were used in the exposure evaluations. In cases where there was an unidentified residual activity in the gross beta-gamma measurements of if there were no isotopes identified, the measured gross beta-gamma activity was distributed according to the radionuclides in the plant liquid releases as reported in "Report on the Operation of Humboldt Bay Power Plant, July 1, 1972, through December 31, 1972," in Table II.

Table I summarizes the potential exposures from the ingestion of the three aquatic species, gaper clams, Pacific oysters, and rock crabs. These data showed that the ingestion of the above species would result in exposures to all organs of much less than 1 percent of the 10 CFR 20 limits.

2. Potential Exposure from Milk

The plant contribution to the radioactivity in milk and potential exposures were determined by two different methods. Since no specific radionuclides other than 40-K were identified in the milk samples, the first estimate of exposure assumed that the difference between the gross beta-gamma radioactivity measurement and the 40-K activity, as determined by atomic absorption spectroscopy, was of plant origin. Large errors are introduced in this method of analysis because essentially all of the measured radioactivity is 40-K. This model, like the aquatic model, assumes that the unidentified activity is distributed according to the isotopic distribution of measured plant airborne particulate releases. Using this assumption, the principal isotope in milk was 89-Sr from the chain 89-Kr 3.18m 89-Rb 15.2m 89-Sr 50.8d 89-Y stable.

The second model used the measured airborne release data from the plant and the annual average X/Q data to predict airborne concentrations. In this calculation, the proposed AEC model in Regulatory Guide 1.42, Appendix C, for transfer of radioiodine through the air-grass-milk chain was used to estimate the concentration of all isotopes as well as the iodines in milk. Again, the principal isotope of concern was 89-Sr from the above mentioned decay chain. The effective environmental decay constants of 4 days* for 89-Sr and 12 days for the iodines were used.

Table 2 compares net radioactivity levels and potential exposures from Model 1 and Model 2. The exposure estimates from Model 2 are believed to be more realistic and are in the range of typical background radioactivity in milk. Typical background data for activity in milk is compiled in the report, "Radiation Data and Reports, Volume 13, No. 12, December 1972," from the EPA's Pasteurized Milk Network. The data show measurements as Humboldt of an annual average of 12 pCi/ℓ for September 1971 to August 1972. These background measurements at Humboldt are typical of California as the whole and similar to the environmental radioactivities based on Model 2 which used effluent data.

Thus, the procedure for the determination of radioactivity in milk in Model 1 overestimates the activity and potential public exposures. Activities and exposures calculated in Model 2 show that these activities are within the range of local background measurements as compiled by the Environmental Protection Agency.

3. Potential Exposure from Domestic Water

The potential exposure resulting from domestic water was also considered. The data from water samples collected from wells supplying the plant, which are shown in Table 3 of the attached reports, show a range of gross beta-gamma activity of 1.0 to 1.2 pCi/ℓ. Although no recent data exist on the radioactivity of domestic water in the Eureka area, the above values may be compared with data collected in 1970. These data[1] from well-water in Crescent City, approximately 80 miles north of Eureka,

*Krieger, Herman L., "Effective Half-Times of 85-Sr and 134-Cs for a Contaminated Pasture," Health Physics, Volume 17, pp 811-824.

and treated water in Eureka, show yearly gross beta-gamma activity averages of 1.0 and 7.0 pCi/ℓ, respectively. It should be noted that because of the low beta-gamma activities, the water samples are not routinely gamma-scanned. Thus, when comparing the gross beta-gamma activity measurements in domestic water, it can be seen that the levels of radioactivity are well within local background radiation levels.

4. Potential Exposure from Air Particulate

The fourth potential exposure pathway to man was via inhalation of airborne particulates. Data From Table 6 of the Environmental Radiation Study Reports were averaged for each quarter and exposures calculated assuming that the unidentified isotopes were distributed according to the measured particulate releases from the plant during the corresponding period. The principal contributor to the exposure was assumed to be 89-Sr. The data shown in Table 3 results in exposures well below 1 percent of the 10 CFR 20 exposure limits. It should be noted that although all the particulate activity was assumed to be 89-Sr, the measured air particulate activities during this half of 1972 were well within the range of activities as measured by the network of air particulate stations in the State of California.[2] Thus, the exposure resulting from the above assumption and shown in Table 3 is well below 1 percent of the 10 CFR 20 exposure limits.

5. Potential Exposure for External Radiation

As seen in Figure 1, there are currently 30 dosimetry stations in the vicinity of the plant. Ionization chambers, which are typically read on a bi-weekly basis, are presently being utilized for dosimetry. Table 4 in the attached environmental radiation study reports, presents the bi-weekly dosimetry measurements from all stations with Stations 2 and 5 representing background in that they are assumed to be completely removed from the influence of the plant.

In order to test for statistically significant difference between stations, two statistical tests, a two-way classification, and a 95 percent confidence limit least significant difference test, were made using the average bi-weekly dosimeter readings from each station. Using the above tests, it was determined that 10 stations were statistically significant above background, with Station 14 being the highest at 1.9 mrem per month above background. As can be seen in Figure 1, the stations above background are located near the plant and at low elevations. The exposures associated with the gaseous effluent from the plant are greatly dependent not only upon meteorological conditions and distance from the plant, but also upon the topography.

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[1] "Radiation Data and Reports," U.S. Environmental Protection Agency, Volume 13, Number 6, June 1972.

[2] Private Communication, State of California, Bureau of Radiological Health, 1973.

TABLE 1

ESTIMATED EXPOSURES FROM MEASURED ACTIVITY IN AQUATIC SPECIES

STATION NUMBER	SAMPLE DESCRIPTION	1972 QUARTER COLLECTED	EXPOSURE (mrem/quarter) WHOLE BODY	BONE	INTERNAL	THYROID
59	72429 Gaper Clam	3rd	0.026 ±0.018	0.039 ±0.026	0.051 ±0.034	1
59	72527 Gaper Clam	4th	0.015 ±0.002	0.030 ±0.004	0.021 ±0.006	0.043
65	72430 Pacific Oyster	3rd	2	-	-	-
65	72526 Pacific Oyster	4th	0.031 ±0.002	0.065 ±0.004	0.032 ±0.002	0.085 ±0.005
59	72528 Rock Crab	4th	0.067 ±0.019	0.065 ±0.019	0.034 ±0.009	0.085 ±0.023

[1] Thyroid is not a reference organ for thes isotopes identified in this sample.

[2] Radioactivity in this sample was not above the background 40-K activity.

TABLE 2

MODEL 1 - ESTIMATED EXPOSURES FROM MEASURED ACTIVITY IN MILK

Station No.	Sample Description	1972 Quarter Collected	Net Sample Activity pCi/ℓ	Exposure (mrem/quarter) Whole Body	Bone	Internal	Thyroid
6	72317 Milk	3rd	90±13	1.71±0.25	3.08±0.45	0.39±0.06	1
16	72318	3rd	150±49	2.84±0.93	5.13±1.68	0.64±0.21	
6	72410	3rd	88±30	1.66±0.57	3.01±1.03	0.38±0.13	
16	72411	3rd	0	2	2	2	
6	72517	4th	21±34	0.40±0.65	0.74±1.16	0.09±0.15	
16	72518	4th	97±39	1.84±0.74	3.32±1.34	0.42±0.17	

MODEL 2 - ESTIMATED EXPOSURES FROM PLANT RELEASE DATA

Station No.	Distance from Plant (Km)	X/Q (sec/m^3)	Sample Activity pCi/ℓ	Exposure (mrem/quarter) Whole Body	Bone	Internal	Thyroid
16	7.8	6.13×10^{-8}	7.25	0.14	0.25	0.03	
6	10.7	2.0×10^{-8}	2.36	0.05	0.08	0.01	

1 Exposure to thyroid was several orders of magnitude below the exposure to the other organs.

2 Radioactivity in this sample was not above the background 40-K.

TABLE 3

ESTIMATED EXPOSURES FROM MEASURED ACTIVITY IN AIR PARTICULATE

Station No.	Average of Weekly Air Particulate Activity pCi/m³	1972 Quarter Collected	Radiation Exposure per Quarter[1] Whole Body	Bone	Internal
3	0.34 ± 0.003	3rd	0.004 ± 0.0003	0.004 ± 0.0003	0.001 ± 0.0001
3	0.20 ± 0.002	4th	0.002 ± 0.0001	0.003 ± 0.0002	0.001 ± 0.0001

[1] Radiation exposures are assumed to be principally due to 89-Sr.