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

March 14, 1966

CO REPORT NO. 133/66-1

Title: PACIFIC GAS AND ELECTRIC COMPANY (HUMBOLDT BAY)
LICENSE NO. DPR-7
DATE OF VISITS: February 6, 7, 8, 9 and 16, 1966

By: A. D. Johnson, Reactor Inspector

SUMMARY

The Humboldt Reactor facility was visited for the purpose of conducting a routine inspection and to review, in particular, information associated with PG&E's decision to continue operation of the reactor following the discovery that there had been a sudden loss of reactivity. The loss of reactivity was believed to have been caused by a substantial reduction in core coolant flow due to the accumulation of scale in the fuel assemblies. In addition, the licensee's General Office in San Francisco was visited to discuss the more significant safety implications and the licensing and regulatory considerations associated with the anomalous performance of the Humboldt reactor.

The reduction in coolant flow is estimated to be approximately 15%. The licensee has evaluated the cause and effect of the flow reduction. From these evaluations, the licensee has concluded that present conditions do not involve a significant safety problem. However, the "Rated Power" of the reactor has been temporarily reduced by PG&E from 177 Mwt to 130 Mwt as a result of flow reduction.

Three items of noncompliance were noted. They are as follows:

1. The licensee failed to immediately report in writing the reduction in normal core coolant flow. A report is required by Section 3.C of the Facility License, DPR-7.

2. Corrective action was not initiated when the pH of the primary coolant exceeded the operating limit on a number of occasions. Corrective action is required by Section IV.B.5 of the Technical Specifications.

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3. An employee received 3.760 rem during the fourth quarter of 1965. This exposure is in excess of the limit specified by 10 CFR 20.101.

Other items of interest noted during the visit include the following:

1. Successful preoperational tests were performed on all systems modified in accordance with the provisions of Change No. 17 to the technical specifications.

2. The interior of the reactor vessel was inspected. No evidence of pitting or cracking was noted on any of the surfaces examined.

3. The emergency diesel generator failed to start on November 25, 1965. The licensee has increased the frequency and scope of the unit's preventive maintenance program.

4. No indication of the occurrence of additional fuel failures has been observed since operations were resumed on December 1, 1965.

5. An extensive review of the Humboldt radiation safety program indicates that it meets the requirements of the license and complies with the provisions of 10 CFR 20.

DETAILS

I. Scope of Visit

The Pacific Gas and Electric Company's Humboldt Bay Reactor facility, Unit #3, Eureka, California, was visited on February 6, 7, 8 and 9, 1966 by A. D. Johnson, Region V, Division of Compliance. Mr. A. G. Johnson, Radiation Specialist, Region V, Division of Compliance, accompanied the writer on February 8 and 9, 1966. Section X of this report was written by A. G. Johnson.

The visit included: (1) an analysis and evaluation of reactor operating parameters pertinent to the recently experienced reduction in core coolant flow; (2) a review of changes to the facility which have been authorized by Change No. 17 to the technical specifications (changes performed in conjunction with the Core II A refueling outage involving zircaloy clad fuel elements); (3) a routine inspection of the facility, and (4) an extensive review of the facility radiation protection program.

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In addition to items 1, 2, 3 and 4 above, this report includes the significant results of a meeting between Compliance personnel and management representatives of the Pacific Gas and Electric Company which was held in the licensee's General Office in San Francisco, California on February 16, 1966. The purpose of the meeting was to discuss the more significant safety implications and the licensing and regulatory considerations associated with the recent core coolant flow reduction at the Humboldt reactor. The following personnel participated in this meeting.

Pacific Gas & Electric Company

H. Braun	-	Senior Vice President, Electrical Operations
F. Matthews		Manager, Steam Generation

Division of Compliance

L. Kornblith, Jr.	-	Assistant Director for Reactors
R. H. Engelken	-	Senior Reactor Inspection, Region V
A. D. Johnson	-	Reactor Inspection, Region V

Because of the extensive scope of the information in this report and the wide range of material covered, the report has been divided into several main sections. They are indexed as follows:

Section II.A: Operating History
Section II.B: Core Coolant Flow Reduction
Section II.C: Change No. 17 to the Technical Specifications
Section II.D: Routine Facility Inspection
Section II.E: Health Physics Review
Section III: Exit Interview
Section IV: Meeting with PG&E General Office Personnel

Principal contacts during the visits were as follows:

H. Braun	-	Senior Vice President, Electrical Operations
F. Matthews	-	Manager, Steam Generation
J. Carroll	-	Senior Steam Engineer, General Office
D. Nix	-	Plant Superintendent
W. Raymond	-	Assistant Plant Superintendent

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E. Weeks	-	Technical Supervisor
J. Shiffer	-	Nuclear Engineer
G. Allen	-	Radiation Protection Engineer
J. Barr	-	Shift Foreman
O. Cole	-	Senior Control Operator
W. Dilbeck	-	Control Operator

II. Results of Visit

A. Operating History

A review of facility records and discussions with licensee personnel provided the following brief chronological history of operations and core flow.

December 1, 1965	-	Startup for fuel cycle 2A. Predicted core flow was 11.5×10^6 lbs/hr at 165 Mwt.
December 1-7	-	Reactor power varied from 0 to 168 Mwt for required equipment adjustments, irradiation of flux wires for calibration of incore flux monitors and total core flow measurements. The flow was determined to be 11.3×10^6 lbs/hr at 165 Mwt.
December 7, 1965 to January 5, 1966	-	Reactor power was maintained at 70 Mwt or less.
January 5	-	The reactor power was increased to 175 Mwt for core flow measurements. The flow was approximately 11.3×10^6 lbs/hr at 175 Mwt.
January 14	-	The reactor was shut down for maintenance purposes.
January 19	-	Completed maintenance outage - reactor startup commenced.
January 21	-	With the reactor power at 70 Mwt and operating under equilibrium conditions, the controlling four rods indicated an unaccountable loss of reactivity equivalent to 0.6 to 0.8% $\Delta k/k$. This was believed to be the result of a reduction of core coolant flow.

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January 25	- The core flow was determined to be 9.6×10^6 lbs/hr at 168 Mwt. No instability or other anomalies were noted during the brief period of 168 Mwt operations.
January 26	- Symmetrically located control rods were "calibrated" by noting the change in power (Mwe) associated with equal incremental movements of opposing rods. The purpose of these checks was to observe possible asymmetric effects in core flow or other thermal/hydraulic anomalies.
January 28	- A feed water shutoff transient test was conducted at 130 Mwt. No abnormal behavior of the reactor was noted.
January 28	- The On-Site Committee reduced the facility's "rated" power level from 177 Mwt to 130 Mwt. The base load of operations was to be 70 Mwt.
January 28 to February 9	- Reactor power was maintained at 70 Mwt.
January 31	- Licensee informally notified Region V of the reduction in the core coolant flow.

B. Core Coolant Flow Reduction

The inspector reviewed records and discussed the cause of the abrupt change in core coolant flow which was noted on January 21, 1966, following reactor startup from the recent maintenance outage. Discussions were held with Messrs. Nix, Raymond, Week, Shiffer and Carroll. Results of the discussions were as follows:

1. **Fuel Inspection**

During the refueling cycle for fuel cycle 2A (September 20 - December 1, 1965), inspection of several fuel assemblies showed that a significant amount of scale had built up on the fuel cladding.* It was estimated that the buildup

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on the more severely affected elements was 8 - 10 mils thick. A photograph of fuel assembly A-147 showed that a significant amount of the scale had spalled from the cladding. The scale was retained in the assembly by the wire fuel element spacers. It appeared from the photograph that approximately 1½ to 2 inches of scale residue was resting on the spacer.

During the previous inspection visit, the inspector was informed of the effect that the scale buildup on the fuel clad had had on the flow characteristics of the fuel assemblies at that time. However, the fact that the scale had spalled from the cladding and had become trapped in the fuel assemblies was not made known to the inspector. Also, the scale deposits were not evident in the photographs available to the inspector at that time.

2. Flow Tests

A series of flow tests was made on five or six of the assemblies which had been left in the core for fuel cycle 2A to determine the effect that the trapped clad scale had on coolant flow characteristics. As a result of these tests the fuel assemblies were classified "Clean", "Moderately Dirty" or "Dirty". The licensee and G-E both concluded that the amount of scale in a particular fuel assembly was dependent upon the radial peaking factor of the assembly's core position during fuel cycle 1-B.

The classifications and corresponding radial peaking factors were as follows:

Classification	Radial Peaking Factor
Clean	<1 and new fuel
Moderately Dirty	>1 but <1.2
Dirty	>1.2

The flow tests were made at flows of 100 and 170 gallons per minute. The differential pressures in inches of water recorded for these conditions were as follows:

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	100 gpm	170 gpm
Clean	5.7 in. of water	13.1 in. of water
Moderately Dirty	6.8 in. of water	16.4 in. of water
Dirty	7.8 in. of water	18.7 in. of water

According to Mr. Weeks, the scale remained in the fuel assemblies during and after the above flow tests.

The present core contains 10 Dirty and 33 Moderately Dirty fuel assemblies. These fuel assemblies are located in the center section of the core. Prior to the fuel cycle 2A startup, G-E evaluated the flow relationships of the mixed fuel assemblies in a given fuel cell. These evaluations were performed on a computer utilizing the G-E "COFFI" code. Two cases were evaluated. One case was for a fuel cell with two Moderately Dirty (M.D.), one Dirty (D) and one Clean, or new Type II-Zircaloy cladding (C). The other case was for a cell composed of two "C", one "D" and one "MD". A review of the data showed that the ratio of flow through a specific channel to that of the average channel flow at 165 Mwt was as follows:

	Case I (2-M.D. 1-D 1-C)	Case II (2-C, 1-D, 1-M.D.)
Moderately Dirty	.99	1.08
Dirty	.94	.92
Clean (Zr clad)	.89	.87

The above values are for core areas with a radial peaking factor of 1.3. This is the area of most interest. Case I represents actual fuel cell loadings in the present core. Case II was done for comparison purposes. The licensee concluded from the above tests and computer evaluations that despite the scale deposition and resulting flow reduction in Type I fuel, the thermal-hydraulic characteristics of the new Type II fuel would limit the reactor power level on the basis of MBOR. The MBOR in the hottest channel for overpower conditions at 165 Mwt was calculated to be 3.3.

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3. Power Schedule Prior to Flow Reduction

Prior to reactor operation on December 1, 1965, the licensee had decided to operate Unit #3 at 20 Mwe and be on call from the company's power dispatcher for a "rated" power level of 52 Mwe for this fuel cycle. According to Carroll the reduced power operation was planned in an attempt to increase the lifetime of the stainless steel clad fuel (Type I). In addition, because of the condition of the Type I fuel, it was planned to raise reactor power to approximately 165 Mwt at least monthly to check the core flow performance. It was also planned to observe control rod positions closely since reactivity changes would be an indication of a change in total core flow.

4. Mechanism of Flow Reduction

The flow reduction has been attributed to the spallation of cladding scale which then lodged in the fuel assemblies to form a flow restriction. The principal constituents of the scale have been identified as Cu, Fe, Zn and O_2. The molecular composition was identified as X Fe_2O_4, where X is either Cu or Zn. The crystalline structure was identified as a spinel which has a continuous buildup rate, whereas the normal magnetite (Fe_3O_4) will only build up to a thickness of a few mils. Because of the different thermal properties of the clad and the scale, the scale cracks and spalls from the cladding during thermal cycles. Also, when this material is alternately subjected to aerated water and deaerated water, a loss of hardness occurs. The source of the Cu and Zn has been identified to be the corrosion and erosion of the tube bundles in the Admiralty feed water heaters. The licensee plans to replace the tube bundles in these heaters with stainless steel bundles during the next refueling outage. Further information concerning this problem is contained in Section II.D.c. of this report. The tentative date for the next refueling outage is August, 1966.

5. Indication and Evaluation of Additional Flow Reduction

On January 21 following the reactor startup from the maintenance outage which began on January 14, control rod positions at 70 Mwt indicated a loss of reactivity. The loss was noted to be between 0.6 and 0.8% $\Delta k/k$ and could not be definitely explained. However, the licensee felt that accelerated scale spallation, which resulted in further reduction of core coolant flow, probably accounted for the loss of reactivity. By way of verification, the following test program was carried out between January 25 and January 28, 1966.

a. Core Flow Measurements

The reactor power was increased to 168 Mwt on January 25, 1966. The total flow at that time was determined to be 9.6×10^6 lbs/hr. The reactor feed water flow was varied while observing the nuclear instrumentation. No abnormalities were indicated by nuclear or other instrumentation.

b. Feed Water Transient Test

A feed water shutoff transient test was performed on January 28, 1966. The reactor power was 130 Mwt. High-speed Sanborn recorder traces were made of two of the picoammeter neutron flux signals. These picoammeters are located 120° apart. The Sanborn traces showed uniform response of the neutron flux during the transient test. No abnormal behavior of the reactor was indicated by the Sanborn recordings.

c. Control Rod "Calibration"

Calibration of symmetrical control rods in core 2-A was performed on January 26, 1966. The purpose of these checks was to determine if nonuniformly distributed flow existed in different regions of the core. To accomplish this, the assumed symmetrical rods C-6, D-1 and A-3 were calibrated by incrementally moving the control rod through its entire stroke and noting the change in electrical generation. The total worths of rods C-6, D-1 and A-3 were 5.09, 4.99 and 5.06 Mwe, respectively. According to Carroll, the approximate equal reactivity worths of these rods, showed by these checks, indicated that the core coolant flow was uniform in different sections of the core.

6. Formal Review and Corrective Action by Licensee

The inspector met with members of the On-Site Review Committee on February 6, 1966. The Committee was in unanimous agreement that the apparent loss of reactivity

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noted on January 21, 1966 did not present an immediate safety problem. The Committee felt that the cause for the loss of reactivity was reasonably well known and understood. On January 28, 1966 the Committee decided that it would be "politically expedient" to limit the "rated power" of the unit to 130 Mwt until the General Electric Company had completed its computer evaluation of the situation. According to Mr. Carroll, G-E had concurred with PG&E's decision concerning the safety aspects of the problem. Mr. Carroll stated that some of the computer evaluations made since the noted reduction of core flow indicate that the channel flow in the assemblies of concern was approximately 38,000 to 40,000 lb/hr. With these flow rates the minimum burnout ratio at 125% of 165 Mwt was calculated to be 3.2. Based on test data obtained during the uprating test program in November, 1964, a channel flow rate of 35,000 lb/hr would result in a burnout ratio of 1.53 at a reactor power level of 288 Mwt.

In addition to the above, the facility management issued a memorandum on January 25, 1966 to all reactor operators which reads as follows:

"As you all know, we have experienced a loss in reactivity and core flow since the outage on 1/20. We are still investigating the possible causes of this problem, but cannot, as yet, state definitely where the trouble lies. Although the evidence to date indicates that operation of the plant can be safely continued, there is no assurance that further deterioration will not take place. Therefore, any unusual reactor phenomena which you observe should be reported to the Nuclear Engineer immediately. The type of things which should be watched for are sudden changes in reactivity (either up or down), the appearance of instabilities in pico traces and feedwater flow traces, sudden changes in off-gas activity, etc. We also want to guarantee that if a gradual deterioration is taking place, it will not be masked by higher than normal rod withdrawal. Therefore, under equilibrium xenon conditions, a rod withdrawal of more than 2 notches in any one day must not be made without prior approval from the Nuclear Engineer. Furthermore, in one week, the number of notches which are withdrawn should not exceed 5 without approval from the Nuclear Engineer."

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A copy of the above notice was also posted in the console logbook.

7. **Discussions with Shift Personnel**

The inspector discussed recent operations with Mr. Barr, a Shift Foreman, and Messrs. Cole and Dilbeck, reactor operators. These personnel indicated that they had observed no unusual behavior of the reactor or of any of its components, such as the control rod, since the startup for fuel cycle 2-A. They indicated a possible exception to this observation was that the controlling rods were further withdrawn after equilibrium conditions were reached at 70 Mwt on January 21, 1966, than they had been at 70 Mwt prior to the reactor shutdown on January 14, 1966. The inspector's review of the operating records, picoammeter recording and the gas monitor recording supported the operators' comments.

8. **License Notification**

Mr. Carroll informally notified the writer on January 31, 1966 of the core coolant flow reduction which had been observed on January 21, 1966. On that date he was advised that it appeared that an immediate report of the flow reduction and an evaluation of its safety implications was required by the facility license. Carroll contended that the flow reduction was not a safety problem and that the flow had not decreased to a point where it was "significantly different" than that specified in the technical specifications. (Table V-2, Section V of the Technical Specifications lists a design total core flow of 11.3×10^6 lbs/hr for the present core loading vs a measured flow of 9.6×10^6 lbs/hr at 168 Mwt on January 25, 1966.)

After learning of Mr. Carroll's position concerning the reporting requirements of the license on January 31, R. H. Engelken, Senior Reactor Inspector, Region V, discussed the matter with Carroll several times during the next few days. At the conclusion of these discussions, a substantial difference of opinion concerning the reporting requirements still existed between the PG&E and AEC representatives. Carroll summarized his position substantially as follows:

a. The loss of reactivity and apparent reduction in flow did not constitute a safety problem at reduced reactor power levels.

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b. The reporting provisions of the license are only applicable in situations which clearly involve potential hazard to the health and safety of the public.

c. The technical personnel who write the reports were too busy writing other reports and doing other more essential work.

d. One of the reasons for deciding not to submit an official report was to determine "how serious the Commission was when they promised minimum regulation."

Engelken stated his position as follows:

a. The observed loss of reactivity and the apparent reduction in core flow were significant deviations from specified performance and, as such, should be reported in accordance with the provisions of the license.

b. From the limited information available it was not clear that the anomalies in reactor performance did not constitute a significant safety problem or an "Unreviewed Safety Problem" as defined in 10 CFR 50. It was felt that this conclusion was particularly valid because the licensee's theories concerning the cause of the anomalous performance had not been verified by inspection of the core.

c. The abnormal performance of the reactor, in the opinion of the inspector, his supervision and representatives of the Division of Reactor Licensing, was clearly the type of problem that would be of substantial interest and use to the Commission in their continuing evaluation of the safety of the Humboldt reactor. This being the case, it was quite clear that the reporting requirements of the license were intended to include reporting abnormalities of this nature.

Following these discussions and the licensee's continued reluctance to submit a written report, the Division of Reactor Licensing requested a full report by PG&E in a letter from Dr. R. L. Doan dated February 8, 1966. The licensee responded by submitting a full report on February 24, 1966.

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The failure of the licensee to make an immediate report in writing to the Commission was in apparent noncompliance with Section 3.C. of the facility license. This section requires that the licensee make an "immediate report in writing of any indication or occurrence of a condition relating to the operation of the facility which might adversely affect the health and safety of the public or facility personnel" and also "any substantial variation disclosed by operation of the facility from design specifications contained in the Technical Specifications."

C. Change No. 17 to Technical Specifications

1. Preoperational Tests

The inspector reviewed procedures and results of tests of the new or modified systems required by Change No. 17 to the Technical Specifications. The systems are as follows:

a. Reactor vent system.

b. Low pressure core flooding system.

c. Cleanup system demineralizer and regenerative heat exchanger bypass system.

d. High pressure core flooding system.

The above systems were checked for proper operation using detailed procedures and check lists. This was to assure that the systems were installed and would operate as described in the amendment application for Change No. 17. The checkoff sheets were noted to have been completed and the systems were approved by the On-Site Review Committee for operation prior to the reactor startup for fuel cycle 2-A on December 1, 1965.

The inspector reviewed code certifications of materials and the xrays of the welds for the installation of the reactor vent valves in the primary steam lines. According to Raymond, he and Mr. Bachans, the Maintenance Foreman, had reviewed all specifications and drawings to assure that the applicable code requirements were fulfilled in connection with the installation of the reactor vent valves. Mr. Raymond stated that all of the specifications for changes involving

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the primary or containment systems were completed as required by the provisions of Section III of the ASME Boiler and Pressure Vessel Code for Nuclear Vessels. Mr. Backens showed the inspector the radiographs and temperature recordings of stress relieving of the welds for the weldolets installed in the main steam line for the reactor vent valves. The inspector noted from the radiographs that no flaws were evident in the welds. Also, the welds were stress relieved at 1000°F. The "soak" was maintained for approximately 80 minutes according to the temperature trace. According to Raymond all welds and changes requiring ASME Code approval were inspected and approved by Mr. G. D. Rice of the Hartford Insurance Company. The State of California accepts Mr. Rice's inspections for assuring pertinent code compliance.

2. Code Compliance

The changes made under Change No. 17, which required code compliance, are listed below:

a. The removal and capping of the two 14-inch vents from the dry well to the suppression chambers.

b. The approximate 4-foot extension of the reactor safety valve discharge lines into the suppression pool.

c. Installation of the reactor vent valves and the 4-inch containment penetration for the air supply lines of the reactor vent valves.

3. Inerted Containment Atmosphere

The inspector reviewed the procedures used to inert the pressure suppression containment atmosphere in November, 1965, and in January, 1966. The procedures appeared to be adequate to assure a uniform gas in the inerted areas. The sampling procedure to measure the percentage of oxygen in the gas appeared to provide for representative sampling of the containment atmosphere. The initial inerting of the containment atmosphere was completed on November 25, 1965. The oxygen concentration of the atmosphere was 3.1%. Nitrogen in the amount of 160,000 ft^3 was used to obtain this concentration. Following the January 14, 1966 outage,

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the dry well atmosphere was again inerted. This required the use of approximately 60,000 ft^3 of nitrogen to reduce the oxygen content to approximately 1%. Section III.B.5 of the Technical Specifications specifies that the oxygen content of the containment atmosphere shall be less than 5% by volume during reactor operating periods.

D. Routine Facility Inspection

1. Water Chemistry

a. The inspector reviewed the results of the daily reactor water analyses for the month of January, 1966. They were as follows:

	January Values	Table IV-2 of Tech. Specs. Operating Limit	Absolute Limit
Conductivity	$<$ 0.5 (μmho/cm^3)	1.0 (μmho/cm^3)	5.0 (μmho/cm^3)
pH	6.4 to 8.7	5.5 to 7.5	4 to 10
Chloride ion, ppm	$<$ 0.01	0.1	0.5
Gross radioactivity	$<$ 2 uc/ml	none	50 uc/ml
Total Boron ppm	$<$ 0.04	5	100
Turbidity ppm SiO_2	$<$ 1.0 except for 1/21 when a value of 1.5 was recorded.		

The minimum, average and maximum pH values for the four quarters of 1965 were recorded as follows:

	Minimum	Average	Maximum
First Quarter	6.0	7.21	8.1
Second Quarter	5.2	6.93	7.7
Third Quarter	6.0	7.0	7.6
Fourth Quarter	6.5	7.2	8.5

During the month of January, 1966, the pH of the reactor water was recorded as being higher than the operating limit of 7.5 specified by Section IV.B.5 of the Technical Specifications during 16 of the 26 days of power operation. The records indicated that the longest period of time the pH was above 7.5 was 4 days (1/26-1/29). During this period the values ranged from 7.6 to 8.7.

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Section IV.B.5 requires that if the pH operating limit of 7.5 specified in Table IV-2 is exceeded operations of the reactor may continue but corrective action shall be initiated. Mr. Shiffer stated that no specific corrective action had been initiated to control the pH of the reactor water. The failure to take corrective action to reduce the pH below 7.5 is in noncompliance with Section IV.B.2 of the facility's Technical Specifications. Shiffer explained that the Humboldt process systems have no direct means for controlling the pH. He added that they have interpreted the applicable technical specification to mean that if the pH remained above 7.5 for approximately a week, investigation would be started to determine a means for lowering the pH. One method for lowering the pH is through increased reactor cleanup flow through the primary system demineralizers. Mr. Carroll indicated that they had planned to request a change to the Technical Specifications to raise the operating limit to 8.5 but had decided to delay the request until the submission of other documents necessary for their application for a 40-year license.

Mr. Carroll said that in his opinion the scale noted on the Type I fuel assemblies would not be affected by the pH values that have existed during recent operations.

After review of the procedure used for determining the pH, the inspector stated that the recorded values were probably lower than the actual because of the samples exposure to air for about 15 minutes prior to measuring the pH value. Mr. Shiffer indicated that the accuracy of the pH determination had not been questioned to date. He indicated that the procedures would be examined to see if improvements in accuracy could be obtained.

b. Uranium Concentration

A one litre sample of reactor water was analyzed for uranium on January 21, 1966. The sample was filtered and the uranium content in both the filtrate and the residue was determined. The results of the analysis was as follows:

Filtrate	-	4.3 ugms/liter
Residue	-	0.3 ugms

According to Shiffer, these values are slightly lower than those noted prior to the end of operation of fuel cycle 1-B.

c. Feed Water

The inspector reviewed a report dated August 25, 1965 by Messrs. Osborn and Lim, of the G-E Vallecitos Chemistry Department. This report indicated the results of an extensive effort to determine what effect the Admiralty feed water heaters could have on the content of Fe, Cu and Zn noted in the reactor water. The results of the report indicated that the corrosion product pickup, as the reactor water passed through the heaters at a flow rate of 1100 gallons per minute and a temperature of 260°F, was as follows:

		Inlet to Heaters	Outlet from Heaters
Zn	soluble	2.0 ppb	18.0 ppb
	insoluble	0.7 ppb	1.9 ppb
Cu	soluble	0.3 ppb	14.0 ppb
	insoluble	0.9 ppb	15.0 ppb
Fe	soluble	6.0 ppb	8.0 ppb
	insoluble	25.0 ppb	22.0 ppb

2. Containment and Confinement Leak Rate Tests

a. Integrated Containment Leak Rate Test

The licensee submitted a report dated January 21, 1966 on containment leak rate testing at the facility. The inspector reviewed the raw data obtained during the integrated leak rate test of the containment system during November, 1965. It appeared to the inspector that the results submitted in the licensee's report to DRL were consistent with the raw data obtained during the tests.

b. Confinement Leak Rate Tests

Section III.B.3 of the Technical Specifications establish the methods and frequency for determining the air in-leakage rate for the refueling building. The inspector's review of the pertinent records indicate that the licensee has complied with the requirements of this specification. Four in-leakage tests of the refueling building were made in December, 1965. The average negative pressure recorded for these tests was approximately 0.4 ± 0.06 inches of water with an in-leakage of 134 cfm. The license limit is a minimum of 0.25 inches of water with an in-leakage of 134 cfm.

3. Reactor Vessel Inspection

Messrs. Backens, Humboldt Maintenance Foreman, Rice, Hartford Insurance Company Inspector, and Chaffee, PG&E Power Production Engineer, inspected the reactor pressure vessel on October 18, 1965. The inspection was made from the inside of the vessel with the aid of a borescope. Specific nozzle areas inspected were as follows:

a. Emergency condenser return

b. Shutdown heat exchanger suction

c. Feed water inlet

d. Various liquid level and pressure sensing outlets

According to Backens, all areas inspected were found to be in good condition. There was no evidence of pitting or cracking of any of the surfaces examined. All surfaces inspected were coated with a uniform reddish-brown material which resembles the scale noted on the fuel assemblies.

4. Emergency Generator

The emergency diesel generator failed to automatically start when the normal power transferred to Load Center No. 5 on November 25, 1965. The diesel could not be started manually. The malfunction was found to be due to a bad battery and an improper setting of the distributor points. Mr. Raymond said that the diesel has been placed on an annual maintenance schedule which requires a complete tuneup and installation of a new battery. The system is routinely operated at least monthly.

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E. Radiation Safety Program

1. Health Physics Organization and Control

The radiation safety program at Humboldt Unit No. 3 was reviewed by A. G. Johnson, Radiation Specialist, Region V, Division of Compliance, on February 8 and 9, 1966. During this phase of the inspection the licensee was represented by Mr. E. Weeks, Technical Supervisor, Mr. J. Shiffer, Nuclear Engineer, and Mr. G. Allen, Radiation Protection Engineer. The latter two individuals currently report to Mr. Weeks who in turn reports to the Plant Superintendent, Mr. D. Nix. In general, Mr. Shiffer is responsible for controlling liquid and stack effluent, while Mr. Allen is responsible for radiation surveys, personnel monitoring, environmental monitoring, and other operational health physics problems. The two men are assisted by control technicians who divide their time on a rotating schedule between chemical control, radiation protection, and instrument technician duties. Control technicians report to the "Engineer-In-Charge" for each of the preceding areas of responsibility.

Radiation work procedures (RWP) and special work permits (SWP) are still used by the licensee to control personnel and work in radiation areas. SWP's are issued where work conditions may change over a short period of time or where a significant hazard may exist. Normally, SWP's are effective for a maximum of 3 consecutive shifts. RWP's are extended permits which are issued to control work in areas where conditions and hazards are known and not expected to change.

2. Personnel Monitoring

The licensee's personnel monitoring program centers around the use of film badges and self-reading pocket dosimeters. However, urinalyses for gross beta-gamma activity, and whole body counting have been used to check for internal deposition of isotopes. To date, no internal deposition has been detected.

Film badges are currently supplied by Radiation Detection Company, Mountain View, California, and are exchanged on an interval of once per month. Self-reading pocket dosimeters in use at the time of inspection included Landsverk, 0 - 200

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millirem range, and a group of new Stephens dosimeters with an extended range of 0 - 500 millirem. The licensee noted that the Stephens dosimeters were acquired following the high exposure of one employee during the fourth quarter of 1965. Approximately 65 full time PG&E employees are badged at the Humboldt facility. Film badges are worn continuously by each employee when present at the reactor. Pocket dosimeters are worn by individuals working in the controlled area and are maintained in a rack adjacent to the area entryway. Upon leaving the controlled area, dosimeter results are recorded by each individual on his exposure estimate card. Cards are reviewed daily by Mr. Allan when data are transferred onto each individual's master exposure estimate sheet. This sheet is designed to provide an up-to-date estimate of the individual's exposure so that appropriate controls can be administered.

A review of film badge records showed that data are maintained in a file of monthly reports from Radiation Detection Company. In the same general file Forms AEC-4, completed exposure estimate cards, and other personal history forms are maintained for each badged individual. The available information constituted the equivalent of Form AEC-5. Film badge results for the period September 15, 1965 through January 15, 1966 were reviewed, and data through December 15, 1965 are summarized in Appendix A. Exposures for the final month of the fourth quarter generally coincided with those recorded for the previous month (11/15/65 through 12/15/65), while the majority of fourth quarter exposures totalled in the range of 1200 to 2500 millirem. The maximum exposure for the fourth calendar quarter was 2700 millirem for Mr. E. Windbigler. Maximum exposures for the year 1965 were obtained by two senior control operators, N. C. Pene, 4655 millirem, and O. A. Cole, 4660 millirem.

The licensee was questioned regarding the 3.760 rem exposure to Mr. John Domino which occurred during the fourth quarter of 1965. Radiation safety control procedures stated by Mr. Weeks, and personnel exposure records for the fourth quarter confirmed information, including corrective actions, described in the licensee's overexposure report to the Commission dated December 20, 1965. The licensee acknowledged that personnel exposure in excess of 3 rem per calendar quarter was contrary to the requirements of 10 CFR 20.101.

(Continued)

3. Monitoring Instrumentation

A wide variety of conventional monitoring and analytical instrumentation was observed to be present at the Humboldt facility. Instruments are normally maintained and calibrated by the licensee, with calibration being performed using a 15 curie cobalt-60 or a 5 curie PuBe neutron source.

4. Radioactive Waste

Mr. Weeks stated that one disposal of solid radioactive waste was conducted during 1965. This was accomplished by transfer of material to California Nuclear Waste Disposal Company on December 17, 1965. The shipment involved 1770 cubic feet of waste with a total activity of 1038 millicuries. The licensee confirmed that high level waste containers were stored in one of three underground concrete pits until decayed to permissible shipping levels. Mr. Weeks stated that one solid waste shipment per year would be considered average for this facility.

Procedures and records for liquid effluent release were reviewed and significant data are summarized in Appendix B. Concentrations and volumes represent values obtained after mixing the collected radioactive liquids (rad waste) with the non-radioactive liquid effluent from the plant. Normally, the concentration of the radioactive liquid component is in the range of 10^{-8} uc/ml. The maximum daily liquid effluent concentration during the fourth quarter was 2.64×10^{-7} uc/ml on December 16, 1965. This value was a factor of approximately 10 higher than normal daily release concentrations recorded during the fourth quarter. The licensee confirmed that part of the liquid analytical procedure included a multichannel analyzer scan of each sample in order to identify the radioisotopes. Statements by the licensee and a review of past scans indicated major isotopes to be cesium-134, cesium-137, and zinc-65.

5. Environmental Survey Program

The Humboldt environmental survey program is essentially as described in CO Report No. 133/65-4. The licensee regularly operates 30 off-site environmental stations containing two stray radiation chambers (0 - 10 mr), and a dual film badge packet containing one standard range beta-gamma film and one low range environmental film

(Continued)

(alleged sensitivity down to 4 mr). The licensee also temporarily installed two additional off-site environmental stations (stations 31 and 32). These stations operated from August 17, 1965 through September 21, 1965, and were used to collect additional data downwind from the reactor.

The licensee operates a (1) cfm Schmidt continuous particulate air sampler located at environmental station No. 11 on Humboldt Hill. The location of Station No. 11, 14 and several other off-site environmental stations is shown in Appendix C of the above referenced report. Stations 31 and 32 were located essentially on a line between Stations 11 and 14. Air sample filter papers are changed once per week, while stray radiation chambers are checked at two week intervals, and environmental film packets are exchanged on a monthly basis.

The licensee collects a wide variety of marine and sediment samples for data required by the Regional Water Pollution Control Board, and has obtained several vegetation samples for iodine-131 analysis. Mr. Shiffer stated that the most recent vegetation samples collected September 27, 1965 showed no detectable iodine activity.

Maximum and minimum values obtained on stray radiation chambers for the period June 22, 1965 through September 14, 1965 have been reported in CO Report No. 133/65-4. Results for the one week interval between September 14 and September 21, 1965 were generally consistent with previous data and showed Station 14 to be highest with a reading of 10+ mr on each chamber.

Following reactor shutdown on September 21, 1965 chamber results dropped to within the range of background readings. No significant increase in chamber results has been recorded since reactor startup on December 1, 1965. Specific values observed in the period between startup and January 18, 1966 showed chamber values within normal background range, except for an occasional 1 mr above background which has occurred randomly throughout the 30 environmental stations. Environmental chamber results are especially interesting when compared to stack gas release rates for corresponding periods of time. As an example, an average stack gas release rate of 39,000 microcuries/second was calculated for the period September 1, through September 14, 1965, while stack gas release rates have averaged only about 5000 microcuries/second since the December 1965 startup.

(Continued)

While film packets have not shown exact agreement with data obtained from stray radiation chambers, the tendency to drop within the normal background range following reactor shutdown was observed. Maximum levels in the range of 27 to 30 mr/month during September 1965 were observed to drop to an average of 8 to 11 mr/month in the months following reactor shutdown. Film data since reactor startup had not been returned from the processor at the time of inspection.

Particulate air sample results obtained from the continuous air sampler at Station 11 were reviewed for the period September 21, 1965 through the end of 1965. Results generally were in the range of approximately 0.060 picocuries per cubic meter. The maximum level during this period was 0.115 picocuries per cubic meter on September 21, 1965, and the minimum level was 0.030 picocuries per cubic meter on September 28, 1965.

6. Radiation Survey Program

The licensee's radiation survey program currently consists of numerous different routine facility surveys, routine air surveys, and various special surveys as required. Routine surveys currently being performed include a once per shift operations survey of major radiation and potential contamination areas, a once per day "clean area survey" by radiation protection control technicians to detect possible spreading of contamination into the locker room, control room, and other adjacent clean areas, a daily radiation and contamination survey of significant locations in the controlled area by radiation protection control technicians, a once per shift air sample in the refueling building, and a weekly routine radiation survey for contamination and radiation levels at randomly selected areas. The latter survey normally includes the counting room, instrument lab, hot machine shop, office areas, and areas around the Unit No. 3 fence.

Results of routine radiation surveys conducted by operations personnel and by radiation protection control technicians were normally in agreement, and showed the expected range of readings from millirem/hour to rem/hour depending upon the location in the plant. In accordance with the exemption contained in license DPR-7, access to some high radiation areas

(Continued)

was controlled by a locked positive barrier in place of high radiation area control or alarm devices normally required by Part 20. It was noted, however, that some high radiation areas were equipped with Part 20 type high radiation area alarms.

Results of contamination surveys normally showed no spread of activity above normal background into the uncontrolled "clean" areas. Wipes of the step-off pad at the main access control point only occasionally showed contamination above background. In the two instances where contamination was observed on the step-off pad the maximum level did not exceed 1000 counts per minute. Mr. Allen stated that contamination was immediately removed and surveys of adjacent areas indicated no further spread. Results of weekly routine radiation and contamination surveys of randomly selected areas throughout the Humboldt facility showed essentially no activity above background in uncontrolled areas, and only occasional activity up to a maximum of 4,000 cpm in the hot machine shop and rad waste facility. Measurements of radiation levels around the Unit No. 3 fence showed a maximum of 2400 cpm, with the normal level being essentially background.

Special and routine breathing air surveys are conducted by the licensee. A Nuclear Measurements Corporation continuous air monitor is present in the refueling building and operates primarily as an indication of general air activity. The device is not calibrated to readily provide results in terms of uc/cc, but would indicate significant changes in air concentration. The licensee also conducts a once per shift air survey in the refueling building using the portable Schmidt air sampler. These samples are counted to determine uc/cc of air activity, and one sample each day is analyzed to identify the radioisotopes. Results of scans indicate that major activity is attributable to cesium-138 and rubidium-88. Activity levels have been in the range of 5.0 to 7.0 x 10^{-9} uc/cc, and no special restrictions on working times are currently in effect. Air samples have also been conducted in an effort to detect I-131. I-131 sampling in breathing air is not now routinely performed, but according to Mr. Allen they plan to take at least one sample each month beginning in the near future. Results of past I-131 samples in breathing air have shown essentially background levels.

(Continued)

Mr. Allen stated that a variety of special radiation surveys are also conducted by radiation control technicians. These surveys consist of all types of radiation measurements associated with the operation of the facility including personnel decontamination, equipment release, and establishing radiation dose rates. Measurement of radiation levels around the bottom of the reactor pressure vessel was given as an example of a special survey. Mr. Allen stated that measurements of this type have been made from time to time since initial startup, and were again checked during the recent reactor shutdown. According to Mr. Allen, radiation levels at the bottom of the reactor pressure vessel during earlier operations normally ranged between 500 and 1,000 mr/hr. However, radiation levels at corresponding locations during the recent outage showed in excess of 50 r/hr in the middle and between 5 and 7 r/hr at the outer edges. Radiation in the lower level of the dry well ranged between 250 and 300 mr/hr, with a maximum radiation exposure rate 3 to 4 times greater than this for a limited area directly under the reactor pressure vessel.

7. **Stack Effluent**

The air ejector off-gas monitoring system and the stack gas monitoring system were confirmed to be as described in Change No. 15, dated August 31, 1964, to the Technical Specifications of License No. DPR-7. At the time of inspection the licensee was using the ion chamber type detector in the off-gas monitoring system rather than the scintillation type detector which is also authorized. Detailed calibration of off-gas and stack gas monitoring systems was performed following the recent reactor startup. Calibration is also checked on a daily basis by laboratory analysis of a 14 ml off-gas sample. According to Mr. Shiffer, this daily sample is held for two hours and then counted on a single channel analyzer set to count all energies above 30 Kev. The count is corrected for decay due to holdup and calculations are made to determine the actual radioactivity concentration of the stack gas effluent. Results obtained in this manner are compared with information on the stack gas monitor recorder. Average stack gas output for the months of December 1965 and January 1966 have been in the range of 5000 microcuries/second. Average particulate concentrations have been approximately 1.0×10^{-3} microcuries/second, while halogen concentrations have averaged approximately 6.0×10^{-2} microcuries/second.

(Continued)

6. Facility Tour

Mr. Allen accompanied the inspectors during a tour of Humboldt Unit No. 3. Radiation measurements made by A. G. Johnson were found to coincide with values reported previously. At several points during the tour it was noted that certain locations were roped off and posted as high radiation areas. Mr. Allen stated, and it was verified by radiation measurements, that these levels were not normally high radiation areas, but that a potential for high radiation levels existed. It was further explained that most of the roped off areas had been evaluated and a decision made to erect appropriate shielding to decrease potential radiation levels or to install a positive locked barrier around the area.

The restricted area was described by Mr. Allen as being the fenced area and building which encloses Unit No. 3. The fence around Unit No. 3 is in addition to several other fences which surround the Humboldt Power Facility. Personal identification is required prior to obtaining access through the main entry gate into the facility, and several other gates in the fence were observed to be locked. It was observed that Form AEC-3 was posted at several locations around the facility, and other radiation area posting and labelling was in accordance with 10 CFR 20.203.

As part of the facility tour discussions were held with members of the plant operations group. Mr. Don Daily, Senior Control Operator, Mr. Ray Swensen, Senior Control Operator, and Mr. Ray Rumrill, Jr., Auxillary Operator, were each interviewed. During the interview each individual was asked to comment on radiation protection procedures and any particular problems in this area which he felt existed. Further, each individual was questioned regarding his participation in the operation's routine radiation surveys, his feelings regarding the quality of portable monitoring instrumentation, adequacy of the protective clothing and other protective equipment, availability of personnel monitoring devices, and management's general attitude with respect to radiation safety. Results of the questioning ineach case indicated no particular problems in the areas of radiation protection practices, management attitudes, or availability of appropriate equipment. It

(Continued)

was generally acknowledged by the individuals that radiation levels within the facility had increased since the beginning of operations, and that additional care was now required, but each felt that adequate efforts were being made to afford an appropriate degree of protection.

III. Exit Interview

The inspectors met with Messrs. Nix, Raymond and Weeks at the end of the visit. The significant items noted in the report were discussed. In particular, the writer reviewed the reporting requirements of the license and the Division of Compliance procedures following notification of an abnormal condition at a licensed facility.

The writer pointed out that it appeared that the continued operation of the reactor with the pH of the primary coolant above 7.5, without initiating corrective action, was in noncompliance with the Technical Specifications. Mr. Weeks indicated that this area of the operation would be reviewed to determine what action should be taken when the pH is determined to be out of limits. In addition he indicated that a request would be made to DRL to increase the operating limit from 7.5 to 8.5. Mr. Weeks further indicated that the procedures used for determining the pH will be reviewed to assure that the measurements are reasonably accurate.

Mr. A. C. Johnson reviewed his finding of the review of the facility's radiation safety program. The quarterly overexposure of the one employee was pointed out as being in noncompliance with the requirements of 10 CFR 20.101.

Mr. Raymond indicated that current plans were to continue operations at 70 Mwt for the remainder of this fuel cycle. Following G-E's evaluation of the reduction in coolant flow, if conditions permit, the "rated" power will be raised from the temporarily reduced level of 130 Mwt to 177 Mwt, the level originally established for that fuel cycle. Mr. Raymond stated that if further deterioration of the fuel is observed, evaluation of the circumstances would be made and appropriate action taken. Mr. Raymond indicated that they feel they understand the flow reduction problem and said that without the background information concerning the scale the reactor would have probably been shut down for investigation when the reactivity loss was noted on January 21, 1966.

IV. Visit to PG&E General Office

Messrs. Kornblith, Engelken and Johnson visited the Pacific Gas & Electric Company's General offices in San Francisco on February 16, 1966 to discuss the more significant safety implications and the licensing and regulatory considerations associated with the recent anomalous performance of the Humboldt reactor. These discussions were held with PG&E's Messrs. Howard Braun, Senior Vice President, Electric Operations, and Paul Matthews, General Manager, Steam Generation.

Mr. Engelken reviewed the organization and activities of the Atomic Energy Commission's Regulatory Program and, in particular, the roles of the Division of Reactor Licensing and the Division of Compliance. He told the group that until recently there had been no serious technical, operational or safety problems encountered with the operation of the Humboldt reactor. However, recent anomalies in the facility's operation and events associated with these difficulties indicated the existence of some differences in opinion regarding the significance of such occurrences and the reporting requirements of the license. Engelken pointed out that the reporting requirements were part of the Commission's continuing surveillance responsibility and were provided to keep the Commission informed of abnormalities and potential safety problems. Mr. Braun indicated that these requirements were reasonable. However, he indicated that PG&E's evaluation, along with G-E's, indicated that there were no immediate safety considerations connected with the recent flow reduction. He added that he felt the information was of interest and would have been included in their routine semi-annual report to the Commission. Mr. Matthews indicated that they needed guidance with regard to what is considered immediately reportable under the provisions of the license. Mr. Engelken stated that he had offered guidance on this problem but that the indications were that it had not been accepted.

Mr. Braun asked if the letter from DRL requesting a report was a punitive action. Mr. Kornblith assured Mr. Braun that the action was not punitive but was a means to gain information so that DRL could evaluate the safety significance of the occurrence.

Mr. Matthews indicated that they feel that they are competent to judge the safety of their operation and further, that when abnormalities occur, judgements must be made without consulting the AEC. He indicated that it was their policy to keep the Division of Compliance fully informed of conditions at the site. Mr. Engelken pointed out that during the refueling outage the Division of Compliance was informed of scale buildup on the fuel, but was not made aware of the spalling conditions. Mr. Braun stated that there had been no intent to withhold information, but that it is a matter of judgement of what has safety significance and what does not. In addition, he stated that they do not want to burden the inspector with insignificant material. Mr. Kornblith indicated that it would be a more satisfactory arrangement if the inspector were allowed to judge for himself what is significant.

(Continued)

Mr. Braun commented that "the current problems seem to be the age old problem of communication". Engelken stated that the problem appeared to be a difference in philosophy concerning the regulatory functions of the AEC, rather than simply a lack of communications.

The importance of an independent audit of the facility was also discussed. Mr. Matthews felt that the facility was adequately audited by Mr. Jay Carroll of the General Office staff. Mr. Engelken indicated that the participation of others, less intimate to the operation of the Humboldt plant, might enhance the objectivity of the audit.

APPENDIX A

RADIATION EXPOSURE SUMMARY
9/15/65 through 12/15/65

Millirem Exposure	PERIOD 9/15-10/15 Extra Workers	Plant Workers	10/15-11/15 Extra Workers	Plant Workers	11/15-12/15 Extra Workers	Plant Workers
over 2000	0	0	4	0	0	1
1700-2000	0	0	3	2	0	0
1500-1699	0	2	3	2	0	0
1200-1499	0	7	3	6	0	1
1000-1199	0	10	4	3	0	3
800-999	0	5	3	1	0	5
600-799	1	8	5	6	1	4
400-599	6	4	3	14	3	15
200-399	13	13	4	18	5	14
10-199	15	15	8	15	11	17
10	0	0	0	0	3	8

APPENDIX

SUMMARY OF LIQUID EFFLUENT RELEASE

Month	Total Volume of Liquid Effluent Released*	Average Concentration of Radioactivity in Liquid Effluent
August 1965	4308.33×10^6 gallons	8.130×10^{-9} uc/ml
September 1965	4452.47×10^6 gallons	1.085×10^{-8} uc/ml
October 1965	4537.47×10^6 gallons	1.486×10^{-8} uc/ml
November 1965	4416.71×10^6 gallons	3.320×10^{-8} uc/ml
December 1965	4589.46×10^6 gallons	2.282×10^{-8} uc/ml
January 1966	4590.83×10^6 gallons	4.010×10^{-9} uc/ml

*Total equals summation of total gallons of liquid rad waste and circulating water.

DRAFT
RTedesco/dj
4/12/66

Docket No. 50-133

Pacific Gas and Electric Company
245 Market Street
San Francisco, California 94106

Attention: Mr. Richard H. Peterson
General Counsel

Gentlemen:

We have reviewed your reports on Humboldt Bay Power Plant Unit No. 3 concerning fuel assembly cladding defect determination by use of a "dry sipping" technique, and flow reduction due to increased core flow resistance, both received under cover letter dated February 24, 1966, which were submitted in accordance with letters from the Commission dated December 16, 1965 and February 8, 1966.

As a result of our review of the "dry sipping" equipment and associated technique for its use, we believe that prior to subsequent use of thedevice, certain automatically actuated safety features should be provided to supplement existing procedural controls. In this regard, consideration should be given to providing an automatic chamber reflooding capability that would be actuated either on high ambient temperature in the chamber or else after a specified heat-up time. Appropriate alarm signals could precede these actions that would allow operator corrective action to be made. Further, the presence of responsible personnel to supervise the operation should be required at all times. In addition, any future consideration to the use of the 'dry sipper' should be contingent upon resolution of the possible contributory effects its use might have had on the current reduction-in-flow problem.

With reference to your report on core flow reduction, it appears that your prediction of margins to fuel damage is based on your understanding of gross core flow and individual fuel assembly flow. Since burnout is basically a local phenomenon involving single fuel pins and the surrounding coolant channels we believe that a significant uncertainty exists in assessing these margins. With the quantities of corrosion products present (1 1/2 inch layer on fuel spacer) and their mobility (from top of one spacer at no flow to bottom of next spacer with full flow), it is likely that individual coolant channels may be restricted by a much higher peercentage than would be predicted by using averages derived from total core flow or individual assembly flow.

The fact that no fuel damage has yet occurred could be explained by:

(1) Deposition is localized and film boiling exists, but because of the relatively low heat flux, no fuel damage results.

(2) Deposition is in fact reasonably uniform (in contradiction to the possibility suggested above.)

In view of the foregoing, we believe that the maximum reactor power level of the Humboldt Bay Power Plant Unit No. 3 should not exceed 130 Mwt pending completion of the study referred to on page 7 of your report. Subsequently, assuming acceptable results from your evaluation and no further reduction in core flow, we believe that the maximum power level during the current operating cycle with the present core loading could be increased to 175 Mwt.

We have recently been informed by the Division of Compliance that the coolant pH of the Humboldt Bay Power Plant Unit No. 3 has increased, at times, during 1965. The increase was beyond the operating limit specified in the

Technical Specifications and that required corrective action was not taken to reduce the pH to within the range of the operating limits. In view of the current reduction in flow possibly due to corrosion products, it is not apparent as to what effect the coolant pH contributed to the overall phenomena. In any case it is recommended that the coolant pH be maintained, by corrective action, within the operational limits given in the Technical Specifications.

Sincerely yours,

R. L. Doan, Director
Division of Reactor Licensing

MEMO ROUTE SLIP
Form AEC-93 (Rev. May 14, 1947)

- [ ] See me about this.
- [ ] Note and return.
- [ ] For concurrence.
- [ ] For signature.
- [ ] For action.
- [ ] For information.

TO (Name and unit): R. G. Page, SLR

INITIALS

DATE

REMARKS

SUBJECT: PACIFIC GAS & ELECTRIC CO.
DOCKET NO. 50-133

We do not concur with the attached letter. In view of the correspondence which has already been exchanged with the licensee regarding the core flow reduction problem, and since pH control does not appear to represent a safety problem, we do not feel the notice of violation should be sent. We do feel that the question of pH limits should be resolved. I note that the licensee's Amendment No. 27 to the application dated 4/19/66 proposes a change to the pH limits in the technical specifications. I have discussed this matter with Roger Boyd and it is my understanding that the question of pH limits will be resolved in DRL's review of Amendment No. 27.

Attachment:
Letter for Concurrence

cc: R. S. Boyd, DRL, w/o attachment
R. H. Engelken, CO:V

TO (Name and unit) | INITIALS | DATE | REMARKS

TO (Name and unit) | INITIALS | DATE | REMARKS

FROM (Name and unit): B. H. Grier, CO

BHG

PHONE NO.

DATE: 5/3/66

USE OTHER SIDE FOR ADDITIONAL REMARKS

U. S. GOVERNMENT PRINTING OFFICE : 1957—O-422007

SLR:RGP
50-133

Pacific Gas and Electric Company
235 Market Street
San Francisco, California 94105

Attention: Mr. Richard H. Peterson
General Counsel

Gentlemen:

This refers to the inspection conducted during February, 1966 of your activities authorized under Facility License No. DPR-7 for the Humboldt Bay reactor.

It appears that in one respect your licensed activities were not conducted in full compliance with Condition No. 3.a of the license, in that corrective action was not initiated when the pH of the primary coolant exceeded 7.5, the operating limit specified in Section IV.B.5 of the Technical Specifications. Your records indicated that during the month of January, 1966 the pH was higher than 7.5, but did not exceed 8.7, on 16 of the 26 days of power operation.

This notice is sent to you pursuant to the provisions of Section 2.201 of the AEC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations. Section 2.201 requires you to submit to this office, within twenty (20) days of your receipt of this notice, a written statement or explanation in reply including (1) corrective steps which have been taken by you, and the results achieved; (2) corrective steps which will be taken; and (3) the date when full compliance will be achieved.

Very truly yours,

bcc: Compliance Div., HQ (2)
Public Document Room
R.S.Boyd, DRL:R&PRSB
OGC

Eber R. Price, Director
Division of State and
Licensee Relations

AIR MAIL

OFFICE ▶	DRL:R&PRSB	DRL	DRL	CO	OGC	SLR
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DATE ▶						

SLR:EB
4-25-66

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