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

Report of Inspection

CO Report No. 50-133/69-3

Licensee:	Pacific Gas and Electric Company License No. DPR-7 Category C
Date of Inspection:	August 28-29, 1969
Date of Previous Inspection:	May 12-14, 1969
Inspected by:	R. T. Dodds 9/26/69 R. T. Dodds Reactor Inspector
Reviewed by:	G. S. Spencer 9/29/69 G. S. Spencer Senior Reactor Inspector
Proprietary Information:	None

SCOPE

Type of Facility:	Boiling Water Reactor
Power Level:	240 Mwt (70 Mwe)
Location:	Humboldt Bay, Eureka, California
Type of Inspection:	Routine - Unannounced
Accompanying Personnel:	None
Scope of Inspection:	Review of facility records, observe reactor operation and tour of reactor facility.

Safety Items - No significant safety items were identified during the visit.

Noncompliance Items - Two items of noncompliance were noted for which a Form AEC-592 was sent to the licensee.

1. The Division of Reactor Licensing was not notified in writing within 30 days that both suppression chamber vacuum relief valves were stuck shut and failed to open when tested.

2. The suppression chamber vacuum relief valves were tested and left set to open at a higher vacuum than that allowed by the technical specifications (Section K.2.c.).

Unusual Occurrences - The following three items were considered occurrences and substantial variances from the design specifications contained in the application or the technical specifications and, therefore, should have been reported in writing to DRL within 30 days. (Note: This opinion is not shared by the licensee.)

1. Control rod F-5 continued to drift after being selected and moved during the drive friction tests on June 16, 1969. When scrammed or inserted the rod would latch, however, it drifted again when additional checks were made. The rod was replaced during the outage (Section F.2.).

2. Control rod A-5 continued to drift after being selected and moved during a reactor startup on July 21, 1969. It was subsequently scrammed and has not exhibited any further tendency to drift (Section F.2.).

3. The drive mechanism for the motor-operated feedwater isolation valve failed during the reactor startup following the refueling outage (Section H.2.).

Other Significant Items

1. There have been no scrams since the previous visit in May, 1969. The reactor was shut down during the period June 14 to July 20, 1969 for the annual refueling outage. Shortly after the startup following the refueling outage, the reactor was shut down for 16 hours to repair the motor-operated drive for one of the two feedwater isolation valves.

At the time of the inspection, the reactor was being operated at 210 Mwt - 65 Mwe. The off-gas release rate at 210 Mwt was approximately 26,000 uCi/sec. Analysis of the off-gas activity data indicated that there were no fuel leakers in the core. The refueled core now contains 132 Type-II elements and 52 Type-III elements (contain gadolinium as a burnable poison). (Section C.)

2. The new generator load control system demonstrated its ability to bypass steam automatically to the condenser without scramming the reactor during a system upset that caused the line frequency to increase to 62 Hz. (Section H.1.)

3. Section F.6. and L. contain information relating to facility modifications that were made to increase safety control system and emergency cooling system reliability.

Management Interview - The results of the visit were reviewed with Messrs. Raymond and Weeks at the completion of the visit and with Mr. Carroll, Supervising Steam Generation Engineer - General Office, by telephone on September 2, 5 and 18, 1969.

They all concurred with the two items of noncompliance involving the vacuum breakers for the suppression chamber. Mr. Weeks stated that the valves would be retested and properly set within two weeks. Unfortunately, these items had been overlooked by Humboldt management during its review of outage test results prior to reactor startup.

It is PG&E's official position that the three occurrences discussed above were not reportable in accordance with provisions C.4.(a) and/or C.4.(b) of the license since they did not "prevent a nuclear system from performing its safety function" and that they do not represent a "substantial variance from design specifications". Therefore, no special reports have been submitted. However, in accordance with CO Headquarters direction and in an attempt to clear the air, these items will be described in detail in the appropriate semiannual report and will be specifically referenced in the cover letter for the report. The inspector informed PG&E that he would encourage the Commission to provide PG&E with written guidance concerning the reporting requirements of the license as they relate to these and similar-type occurrences.

A. Personnel Contacted

Personnel contacted during the visit included the following:

W. Raymond	-	Assistant Plant Superintendent
E. Weeks	-	Plant Engineer
D. Backens	-	Mechanical Foreman
T. Maul	-	Power Production Engineer
G. Boots	-	Chemical Engineer

B. Administration and Organization

1. Operating Organization

Appendix A is an updated copy of the organization chart for Humboldt. There have been no substantial changes in the organization in the past year other than an increase in the number of licensed senior reactor operators (five this summer).

2. Nuclear Engineers

Mr. Shiffer, the principal Nuclear Engineer, is currently at Rochester Gas and Electric's Ginna power reactor facility for the purpose of obtaining startup experience on a pressurized water reactor. He did not leave Humboldt until after completion of the recent refueling outage. The day-to-day chores of the Nuclear Engineer are being performed by Messrs. T. Maul and T. Rapp, Power Production Engineers.* They have been in training at Humboldt since July 1 and December 1, 1968, respectively. Mr. Rapp was previously assigned to the Humboldt plant from April to December, 1965 for training in nuclear power plant technology. According to Mr. Weeks, Plant Engineer and Senior Reactor Operator, he has the capability and experience to provide the needed technical direction and assistance during Mr. Shiffer's absence.

3. Facility Audit

According to Messrs. Weeks and Raymond, Messrs. Carroll, Supervising Steam Generation Engineer, and Scherrer, Senior Steam Generation Engineer, both from the General Office, separately audited the facility during the refueling outage. Carroll and Scherrer were at the facility on June 19-21 and June 25-26, respectively. Their inspections covered a review of the facility records for the past

*See Section I of Humboldt Bay Power Plant semiannual report dated February 19, 1969 for a discussion of the experience and duties of Messrs. Maul and Rapp.

six months operation. Reports of the results of their visits were submitted to Mr. Mathew, Manager, Steam Generation Department.

Another Steam Generation Engineer from the General Office has been scheduled to audit the overall operation at Humboldt this fall. He will audit the "mechanics" of operation - procedures, station orders, housekeeping and conventional safety.

Based on these discussions, it appears that the facility has been audited by members of the General Office in accordance with the requirements of Section IX-B.8. of the technical specifications.

C. Operations

Reactor operation was reviewed by reading logbooks, reports and from discussions with Messrs. Raymond and Weeks. The reactor was shut down during the period June 14 to July 20, 1969 for the annual refueling outage. There have been no scrams since the previous visit. However, the reactor was shut down for 16 hours shortly after the startup following the refueling outage, to repair the motor-operated drive for one of the two feedwater isolation valves.

At the time of the visit, the reactor was being operated at 210 Mwt - 65 Mwe. The off-gas release rate at 210 Mwt was approximately 26,000 uCi/sec. Analysis of the off-gas activity data indicated that there were no fuel leakers in the core.

Principal work performed during the refueling outage included (1) the removal of the last 3 Type-I elements (stainless steel clad) plus 37 Type-II elements; (2) insertion of 52 Type-III elements (contain gadolinium as a burnable poison) which brings the total fuel loading to 184 elements; (3) three strings of incore monitors (three fission chambers in each string) were replaced, and (4) several modifications were made to improve the reliability of the reactor safety system (see Section F.).

E. Primary System

1. Primary System and Pressure Vessel Inspection

The reports and results of the primary system inspections, that were performed during the refueling outage, were reviewed and discussed with Messrs. Raymond and Weeks. The reactor vessel flange, flange stud bolts, and head nozzles (two 3" nozzles and four 4" nozzles including safe ends and transition welds) were ultrasonically tested on June 19-20, 1969 by Messrs. Burke and Friedrick, technicians from PG&E's Department of Engineering Research. Mr. Gail Allen, Radiation Protection Engineer, and Mr. J. Stevens, Hartford Insurance Company, observed the nondestructive testing. No significant discontinuities were identified.

The vessel and other primary system components were visually inspected by Mr. G. Allen, PG&E, and Mr. F. Mayo, Hartford Insurance Company, on July 7, 1969. The plant borescope was inoperative so the visual inspection was performed with binoculars. The water was reported to be clear with very little suspended material. Nozzles N-2, N-3 and N-4 were visible. The deposit of reddish-brown material around them was smooth with no irregularities. The vessel cladding was covered with a uniform reddish-brown deposit. A mark was noted in the deposit on the southwest side of the vessel. This mark did not appear to be deep or sharp. It appeared that a tool or some other object had scraped along the surface during the core loading. No other anomalies were observed. It was pointed out by the licensee that the lower nozzles could not be examined without the borescope.

The primary and associated process system piping were thoroughly inspected during the hydrostatic pressure test (1150 psig) of the vessel and primary system prior to startup. This examination involved a detailed inspection of each flange, all piping flange welds, all accessible piping, vessel nozzles, valves (motor-operated and manual), control rod drives and the control rod drive hydraulic system. No leakage to atmosphere was evident during the hydrostatic test. The primary system leakage rate was determined to be substantially less than the 46 gpm output of the test pump as evidenced by the fact that the bypass control valve was 75% open and the manual bypass valve was one-half turn open. Mr. Raymond personally assisted in the examination of the piping during the test.

2. <u>Steam Safety Valves</u>

As per past practice, two of the four steam safety valves were changed out during the refueling outage. The valves installed in their place had been tested and set to open at 1235 and 1250 psig. The other two valves were tested and set to open at 1250 and 1260 psig a year ago. The limits are 1220 to 1270 psig.

3. <u>Minimum Vessel Temperature</u>

The nil ductility temperature of the pressure vessel was reported to be 15°F. The temperature during the hydrostatic test following the refueling outage was 100°F, substantially above the licensed limit of NDT plus 60°F, i.e., 75°F.

F. Reactivity Control and Core Physics

1. Core Shutdown Margin

Three Type-I and 37 Type-II fuel elements plus 12 dummy elements were removed from the core during the refueling outage. They were replaced with 52 Type-III elements that are equipped with some rods which contain gadolinium as a burnable poison. Subcriticality checks and shutdown margin demonstrations were made during and after the refueling outage in accordance with the requirements of sections V.B.2.a. and V.B.6. of the technical specifications. (Section V.B.2.a. - reactor subcritical with the strongest control rod fully withdrawn and the strongest adjacent central rod withdrawn seven notches -- four sets of rods were used for the demonstration at core loadings of ¼, ½ and fully loaded. Section V.B.6. - subcritical checks prior to and following reactivity increment addition from loading fuel assemblies.)

2. Control Rod System

a. Control Rod Drifting

During the drive friction tests conducted on June 16, 1969 control rod F-5 continued to drift after being selected and moved. The rod was subsequently exercised, scrammed and timed. This temporarily cleared the problem. However, the rod drifted several more times during subsequent testing. It was pointed out by Mr. Raymond that drifting occurred only after the rod had been selected and control rod withdrawal had been initiated by the operator. Further the rod latched and did not drift after insertion had been initiated. The drifting was slower than normal withdrawal and was observed by the operator in each instance.

The drive for F-5 was replaced with one of the two spare drives during the refueling outage. The examination of the drive mechanism did not disclose any anomalies. However, it was suspected that the collet fingers might be weak, therefore, the collet assembly was sent to the General Electric Company for inspection and testing. The results of G-E's examination were not yet available.

Drive A-5 started to drift once during the reactor startup on July 21, 1969. It was subsequently scrammed and has not exhibited any further tendency to drift. The drifting occurred after the rod had been selected and rod withdrawal initiated by the operator. The drift rate was slower than the normal withdrawal rate and was immediately observed by the operator. A-5 is normally fully withdrawn during operation. The control room position indicator has been tagged to call the operator's attention to the possibility of a rod drift problem.

b. Rod Drive Performance Tests

The control rod drive system "as left" was tested during the refueling outage and found to meet the requirements of section V.B.2. of the technical specifications by performance of the following tests:

(1) Withdrawal and insertion time tests

(2) Drive friction tests

(3) Notch latch and unlatch operation tests (O.K. except some position indication lights needed replacing)

(4) Scram time tests (cold scram times were 0.94 - 1.13 seconds, limit $\leq$ 2.5 seconds. Hot scram times in January, 1969, were 1.1 - 1.37 seconds)

c. Rod Following Checks

The review of the Nuclear Engineer's rod following check list disclosed that control rod poison section rod following was verified prior to the refueling outage and during the subsequent approaches to criticality that were made following the refueling outage. The checks were performed in accordance with the requirements of Section V.B.2.d. of the technical specifications.

d. Miscellaneous

Drive D-6 had to be replaced because of "O" ring seal leakage problems discovered during the hydrostatic pressure tests following the refueling outage. Mr. Raymond believed that this was the source of leakage to the drywell (about 50 gallons per month) that started about January, 1969. No increase in the water level of the drywell lower head has been noted since the outage.

3. Liquid Poison System

Chemistry tests conducted on March 11, 1969 (required semi-annually) showed there was 124 pounds (limit $\geq$ 100 pounds) of boron in solution, the solution concentration was 17.6% (limit $\leq$ 40%), and the solution saturation temperature was 77° F (solution temperature controlled at 100 ± 5°F, limit $\geq$ 5°F above saturation temperature). The records for the past four months showed that the nitrogen pressure to the tank has been maintained above 1300 psi. The injection system valves have been exercised monthly. The condition and tests of the poison injection system have complied with the requirements of Section V.B.3. of the technical specifications.

4. Power Coefficient

The power coefficient of the new core was checked during the startup following the refueling outage and verified to be negative from at least 115°F to operating temperature, in accordance with requirements of Section V.B.4. of the technical specifications.

5. Fuel Burnup

Mr. Weeks provided the following information about maximum fuel burnup at the time of the refueling outage:

Type-I

Average	-	8,066 Mwd/T (limit - 10,000 Mwd/T)
Maximum	-	10,785 Mwd/T (limit - 14,000 Mwd/T)

Type-II

Average	-	9,341 Mwd/T (limit - 17,000 Mwd/T)
Maximum	-	12,609 Mwd/T (limit - 23,000 Mwd/T)

6. Reactor Protection System

Several modifications were made to the master scram solenoid valve system to facilitate system testing and increase reliability.

a. Filters were installed ahead of each pressure regulator.

b. Both pressure regulators were overhauled, one at a time.

c. Pressure gauges with gauge savers were installed on each set of master scram solenoids between the solenoid outlet and the manual outlet shut-off valve.

d. Test switches with guarded pushbuttons were installed in the hot line of each of the four scram solenoids between the transfer switch and solenoid.* The modifications, which were made in accordance with the provisions of 10 CFR 50.59, were installed and tested in accordance with written procedures that had been reviewed and approved by the On-Site Review Committee.

Parallel or auxiliary relays were installed in the isolation valve control system in the refueling building high-differential pressure protection system to preclude a single component failure from negating safety action. A manual trip switch was also installed in the isolation valve control system. These modifications, made to improve system reliability, were also reviewed and approved in accordance with the provisions of 10 CFR 50.59.

The following systems were satisfactorily tested prior to and during the refueling outage in accordance with the requirements of Section VI.B.1., 3. and 4. of the technical specifications. The "as left" condition of the setpoints and the special features were noted to comply with Tables VI-1 and 2.

Reactor safety system scram sensor and circuits; master reactor switch; and control rod withdrawal permissive system.

7. Reactor Instrumentation

a. Incore Flux Monitors

Three new strings of incore flux monitors (three monitors per string) were installed during the refueling outage. There were 17 operable incore flux monitors at the time of the visit (12 required for operation above 175 Mwt).

b. General

Instrument performance was observed during the tour of the facility and appeared to be satisfactory. No items of concern or of noncompliance were identified.

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*Appendix B shows the modification to the master scram solenoids.

8. Minimum Critical Heat Flux Ratio

Incore calibrations using incore flux wires were made at a thermal power level of 212 Mwt on July 30, 1969. A rated level of 225 Mwt was used to evaluate MCHFR for the 125% over power condition. The MCHFR at 281 Mwt (i.e., 225x1.25) was 2.93 (limit - 1.5). The peak heat flux for the 125% over power condition was calculated to be 271, 094 BTU/hr-ft^2 (limit - 495,000 BTU/hr-ft^2).

H. Power Conversion System

1. Generator Load Control

The first "test of the pudding" for the new generator load control modification* occurred on May 21, 1969. A system disturbance caused the line frequency to increase to approximately 62 Hz. The unit load was reduced by the new frequency control scheme that automatically bypasses steam to the condenser if line frequency increases to 60.5 Hz or greater. The controller decreased generator output from 57 Mwe to 3 Mwe at a rate of 1.5 Mwe/sec. There was a momentary reactor power "flux spike" of about 12% when the incident first occurred.

According to the minutes of the On-Site Review Committee, all equipment responded in a normal manner during the occurrence. Mr. Raymond stated that, without the new controller, the reactor would have scrammed because of high reactor pressure.

2. Feedwater Isolation Valve

At 1318 hours on July 20, 1969, during power ascension following the refueling outage, it was discovered that the manually controlled, motor operated, feedwater isolation valve was not fully open and could not be opened any further with the motor operator. The position indicator in the control room showed the valve to be fully opened since its indication is derived from drive gear movement rather than valve movement. Feedwater flow was limited to 350,000 lb/hr until the valve was manually cranked open at 1500 hours. The following day the reactor was shut down when it was discovered that the valve could not be closed with a "strong back" after the removal of the manual crank which connects with the motor-operated gear train.

*See Section H.1. of CO Report No. 50-133/69-3

The inspection of the drive mechanism disclosed that the bearing retainer portion of the housing cover for the valve stem worm gear was broken in several places and that the outer bearing race was also broken. It appeared that the bearing retainer had been broken for a long time. The break in the bearing race was new and was probably a direct result of the breaks in the housing. A new housing was machined at the site and then installed along with a new bearing race. The valve was tested for satisfactory operation prior to resuming operations The shutdown to repair the valve lasted about 16 hours.

Mr. Raymond pointed out that the reactor was not shut down immediately because it appeared that the valve could be closed manually. However, the reactor was promptly shut down after the manual motor operator was removed and it was determined that it was no longer possible to close the valve. Both Messrs. Raymond and Weeks pointed out that the backup isolation valve was operable and had just been satisfactorily tested during the hydrostatic pressure test following the refueling outage. With the exception of a break in the feedwater system, the feedwater isolation valves remain open to supply high pressure emergency coolant water during a loss of coolant type accident.

K. Containment

1. Refueling Building

The refueling building in-leakage and the ventilation system were functionally tested for satisfactory operation prior to, during and following the refueling outage in accordance with the requirements of Section III.B.8. and 9. of the technical specifications.

2. Penetration Closure Testing

a. Access Penetrations

The double "O" ring seals on the top and bottom heads of the drywell and the access penetrations to the suppression chamber were satisfactorily leak tested at 75 and 25 psig, respectively, in accordance with Sectinn III.B.5.a. of the technical specifications.

b. Isolation Valves

The isolation valves were tested for proper operation and leak tightness (1150 psig hydrostatic pressure test) during the refueling outage in accordance with Section III.B.5. of the technical specifications. The subsequent malfunction of the manually controlled motor operated feedwater isolation valve is described in Part H.2. of this report.

c. Vacuum Breaker Testing

The operation tests of the suppression chamber and dry well vacuum relief valves that are required every two years were conducted on June 16, 1969. The dry well vacuum breakers were found to have been set and to function in accordance with the technical specifications. However, the suppression chamber vacuum relief valves did not open when subjected to a vacuum of up to 9 inches of water. Further, after repairing the valves, they were tested and found to operate smoothly at a vacuum of 5.0 and 5.4 inches of water. No additional tests were made on these valves even though the Section III.B.5.c. technical specifications require that the suppression chamber vacuum relief valves be set to operate at a vacuum of 2.0 $\pm$ 0.5 inches of water.

The licensee failed to report to DRL in writing within 30 days as required by Section C.4.(b) of Facility Operating License DPR-7 that both suppression chamber vacuum relief valves had stuck - a substantial variance from design specifications contained in the technical specifications.

Both of these items were an oversite on the part of management according to Mr. Weeks. He stated that the valves would be tested and properly set within two weeks. A Form AEC-592 that referenced the two items of noncompliance was sent to the licensee.

3. Containment System Inspection

The dry well and suppression chamber were visually inspected on July 7, 1969. No significant changes were noted from conditions observed a year ago. There was no apparent corrosion product buildup.

L. Emergency Core Cooling System

The fire pump control system was modified to improve the reliability of the low pressure core flooding system as follows:

1. Automatic sequential starting of the pumps on low fire header system pressure or whenever the low pressure core flooding butterlfy valve opens;

2. Alarms to indicate failure of the diesel engine to start;

3. A second set of diesel engine starting batteries with an under-voltage transfer scheme; and

4. Bypass switches across No. 1 and No. 3 fire pump motor contactor to provide a means of manually starting these pumps in the event of a contactor failure.

Tests of the following systems were reviewed with Mr. Weeks. The systems were tested and found to function in accordance with the requirements of the applicable technical specifications.

System Tested	Technical Specifications
High pressure core flooding system*	III.A.3.c. and IV.A.
Reactor vessel vent system	III.A.3.d. and IV.A.
Core spray system	III.B.2.
Low pressure core flooding system	III.B.7.b.
Reactor cleanup system	IV.A.4.
Emergency condenser	IV.A.3. and B.4.

M. Other Engineered Safeguards

The control rod thimble support was inspected by Messrs. Raymond and Backens on July 7, 1969. The shock absorbers were inspected for signs of oil leakage. None was apparent. Oil levels of the shock absorbers were normal. No binding or interference was observed between the drives and the support.

N. Emergency Power

The sources of electrical power and/or emergency power, a.c. and d.c., that can be used to supply power to the reactor were satisfactorily functionally tested in accordance with the provision of Section VI.7. of the technical specifications as follows:

1. Full load test of the automatic transfer scheme for 480 v.a.c. emergency power source (2 year test).

2. The ability of the d.c. system to supply the emergency short-term load for safe shut down of the reactor (2 year test).

3. The ability of the 2.4 KV bus of the reactor facility to automatically transfer from its house transformer to the Plant 60 KV bus (2 year test).

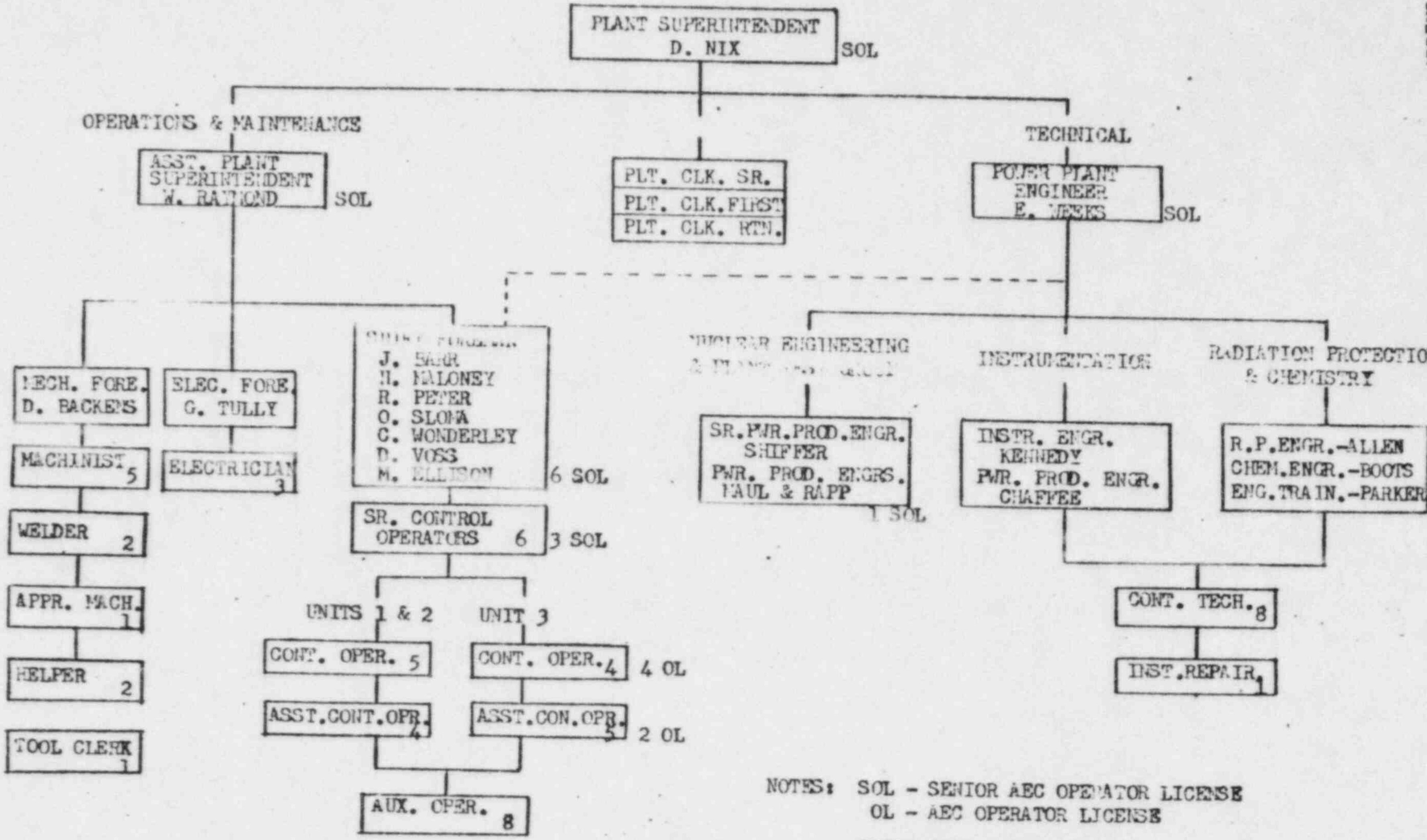
T. Facility Modifications

Changes that were made in accordance with the provisions of 10 CFR 50.59 are contained in Sections F.6. and L. of the report.

*See Part H.2. of this report for discussion of failure of feedwater isolation that occurred following outage.

HUMBOLDT BAY POWER PLANT

ORGANIZATIONAL CHART

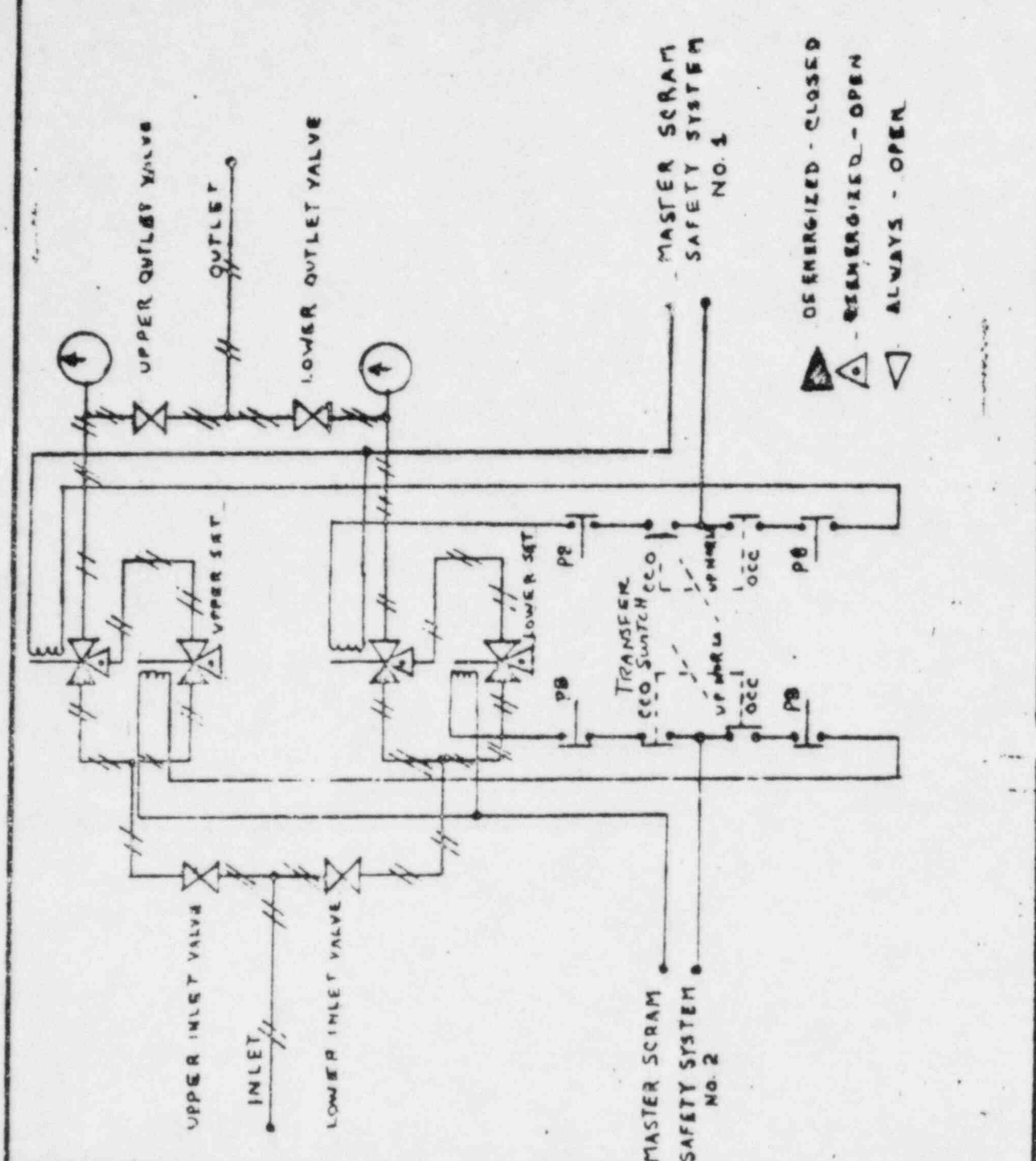


NOTES: SOL - SENIOR AEC OPERATOR LICENSE
OL - AEC OPERATOR LICENSE

TOTAL PERSONNEL = 78

7/29/69

APPENDIX B



APPROVED BY		CHG.	DATE	DESCRIPTION	ON	BY	CK.	APP'D

DEPT. BY
DSGN.
DR.
CH.
O.K.
DATE
5-7-69

MODIFICATION
MASTER SCRAM SOLENOIDS

PACIFIC GAS AND ELECTRIC COMPANY
SAN FRANCISCO, CALIFORNIA

DRAWING LIST
SUPERSEDES
DRAWING NUMBER