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REPORT ON THE OPERATION OF HUMBOLDT BAY POWER PLANT
UNIT NO. 3, COVERING THE PERIOD OF
FEBRUARY 16 THROUGH AUGUST 16, 1963

References: 1. Docket No. 50-133, License No. DPR-7, Pacific Gas and Electric Company.

2. "Initial Core Loading and Critical Testing of the Humboldt Bay Power Plant Unit No. 3", dated April 23, 1963.

3. "Power Operation Testing of the Humboldt Bay Power Plant Unit No. 3", dated June 25, 1963.

This report describes the first six months of operation of Humboldt Bay Power Plant Unit No. 3 (February 16 through August 16, 1963) and is submitted in compliance with Provisional Operating License No. DPR-7, Paragraph 5.

The initial loading and low level critical test program for the reactor began on February 15 and was completed on March 22. This program has been described in Reference (2). Preparations for power operation were completed on April 10. This work included the rearrangement of the reactor core for operation without peripheral control rods as described in Appendix II of Reference (2). The power operation test program which has been described in Reference (3) began at that time and was completed on May 12. Operation of the Unit continued until May 17 when a scheduled outage occurred to correct steam leakage across the horizontal joint of the high pressure turbine and to improve neutron shielding in the drywell lower head. The Unit was returned to service on June 9 and continued to operate until June 21 when another scheduled outage took place. The principal work during this outage was the removal of 36 of the initial 76 poison curtains, gamma scanning of approximately half the fuel assemblies in the core, inspection of the reactor vessel head flange to determine the cause of leakage past the inner "O" ring, and installation of additional neutron and gamma shielding in the area under the drywell. This outage work was completed and the Unit was returned to service on July 31. On August 1, the Unit was placed in commercial operation status.

Specific aspects of the operation of the Unit as required by Paragraph 5 of the Provisional Operating License are discussed below.

A. Hours of Use of the Facility:

During the report period, the reactor was brought critical a total of 281 times (206 of these occurred during the initial loading and critical testing program) and was critical a total of 1,250 hours. Reactor on the line hours (periods when the reactor was critical and supplying steam to the turbine or to the condenser through the bypass valves) totaled 1,060 hours and turbine service hours (periods when the turbine generator was paralleled to the system) totaled 977 hours.

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B. Electric Output of the Facility:

The capability of the turbine generator is 52.0 gross and 50.2 Net MWe when the reactor is operating at the present license power limit of 165 MWt. During the report period gross electrical generation totaled 43,781 MWe hours. Station use on days the turbine generator was in service totaled 2,164 MWe hours resulting in a net electrical generation of 41,617 MWe hours.

C. Shutdowns of the Facility:

During the power operation test program which was completed on May 12, three scheduled outages for the purpose of modification or maintenance of plant equipment took place. Thirteen scrams occurred during the test program and are discussed in Reference (3). Only one of these was a real scram; of the remainder, seven were planned as part of the test program and five were spurious in nature. During the "shakedown" period prior to placing the Unit in commercial operation, four outages occurred; two of these were scheduled outages and two were forced outages resulting from reactor scrams.

Between August 1 when the Unit was placed in commercial operation and August 16, the end of the report period, one brief scheduled outage occurred.

Each of the above outages is described in detail in Appendix I.

D. Levels of Radioactivity Measured on the Site and at Off-site Monitoring Stations.

1. On-site radioactivity measurements - Radiation levels within the controlled area of the Unit were thoroughly investigated during the power operation test program in order to establish the adequacy of plant shielding. As described in Reference (3) dose rates met design criteria when extrapolated to 230 MWt with the exception of two areas in the caisson access shaft. Shielding modifications in these areas are now complete and dose rates are within the design limit. The more significant radiation survey results which have been obtained to date are given in Appendix II.

Contamination levels within the controlled area of the Unit have been very low although a few minor instances of floor contamination have been encountered. These areas were isolated and decontaminated. Maintenance has been performed on a variety of equipment associated with nuclear steam supply and turbine plant systems. Contamination has presented no particular problems in this work.

A comprehensive program for measuring airborne radioactivity has been in effect since the initial receipt of fuel assemblies on the site. With the exception of one problem concerning the release of non-condensable gases from the reactor vessel steam space to the refueling building during reactor startups, (this problem was corrected by repairing a defective weld in a vent line and increasing ventilation flow from the associated radwaste collection tank) no significant air-borne radioactivity problems have been encountered during operation or maintenance of the Unit.

2. Off-site radioactivity measurements - Gamma dose rates in the environs have been measured at thirty off-site environmental monitoring stations using stray radiation chambers and the experimental film packs since June, 1962. (Prior to that time five monitoring stations were in service). A general upward trend in background gamma dose rate began early in 1962 and has continued to date. A typical value for background dose rate prior to early 1962 is 2.0 mr per week. Dose rate has gradually increased to the present typical value of 4.5 mr per week. This increase in background can be accounted for by fallout from the increased activity in nuclear weapons testing during this period and agrees with trends observed in other environmental radiological measurement which are being made by the Company and the State of California. No discernible change in the established dose rate pattern has been observed since the Unit began power operation in April, 1963.

E. Levels of Radioactivity in Principal Systems Established By Chemical Analyses.

The comprehensive program of chemical and radiochemical analyses which has been in effect since the commencement of power operation has shown that levels of radioactivity in principal plant systems are consistent with the experience of other boiling water reactor plants. Appendix III presents a summary of typical results to date.

F. Routine Releases, Discharges and Shipments of Radioactive Materials.

1. Release of Gaseous Radioactive Waste - The routine release of gaseous radioactive wastes has been monitored by the air ejector off-gas and stack gas monitoring systems. The calibration of these monitors for noble and activated gases has been checked by periodic analyses of "grab" samples on a multichannel gamma scintillation spectrometer. The average noble and activated gas release rate during periods of operation was 50 μc per second. The peak release rate occurred during a change in power level and was 500 μc per second. The current license limit for the annual average release rate is 50,000 μc per second.

The particulate and halogen filters which are part of the stack gas monitoring system have been removed weekly for counting. Weekly average particulate plus halogen release rates have ranged between 6×10^{-6} and 2×10^{-3} μc per second. The license limit for the annual average release rate (based on a conservative permissible concentration of 3×10^{-10} μc per cc) is 0.18 μc per second.

The monitoring systems associated with the emergency condenser and liquid radwaste system vents to atmosphere showed that no detectable releases of radioactive gases occurred during the report period.

2. Discharge of Liquid Radioactive Waste - Each batch of liquid radioactive waste has been sampled and analyzed for its concentration of radioactivity prior to release to the plant discharge canal. All radioactivity in liquid waste was either in solution at the time of discharge or was filtered prior to discharge. Radioactive wastes contributed by the operation of the Unit have resulted in concentrations in the plant discharge canal of less than 1.0×10^{-8} µc per ml when averaged over all seven consecutive day periods. The average concentration during the report period was 0.16×10^{-8} µc per ml. Analysis of weekly composite samples from the plant effluent canal and monitoring by the liquid waste discharge monitor have confirmed that no unaccounted release of radioactive waste occurred during the report period. These results are in compliance with the liquid radioactive waste discharge limits contained in the license.

3. Off-site Shipments of Radioactive Materials - Eight off-site shipments of radioactive material have been made to date. These are described in Appendix IV.

G. Principal Maintenance Performed and Principal Changes Made in the Facility with the Reasons therefor.

1. Principal Maintenance Performed - The principal maintenance performed during the report period is described in Appendix V.

2. Principal Changes Made in the Facility - The principal changes made in the unit prior to April 23, 1963 are described in Appendix I of Reference (2). Changes made between April 23 and the end of the report period are described in Reference (3) and in Appendix VI of this report.

H. Description of Significant Tests Performed and the Results of the Test Analysis Completed During the Report Period.

Significant tests performed during the report period excluding test work previously reported in References (2) and (3) were as follows:

1. Reactor critical testing incident to the removal of 36 of the initial 76 poison curtains.

2. Gamma scanning of fuel assemblies for the purpose of determing core power distribution.

3. Subcritical core pulsed neutron tests.

4. Routine operational testing of the Unit as required by Table IX-1 of the Technical Specifications of the license and plant operating procedures.

Appendix VII consists of a discussion of these tests and the analysis of the test results which have been performed.

I. Summary of Meteorological Data Obtained At the Site

Meteorological data obtained to date is summarized in Appendix VIII of this report. Data was collected utilizing the original 200 foot meteorological tower from April to September, 1962. In mid-September, 1962 the instruments were relocated onto the new 250 foot tower. The details of the changes in the tower are described in Appendix II of Reference (2). A smoke generator and stack have been added to the new facility. Smoke release studies will be made after check-out and "debugging" of the automatic data logging system are complete.

APPENDIX I
DESCRIPTION OF OUTAGES*

Outages During the Power Test Program

Outage No. 1

Outage period - 10:34 a.m., 4-14-63 to 7:48 p.m., 4-15-63
Duration of outage - 33 hours 12 minutes (1.38 days)
Type of outage - Scheduled
Reason for outage - Miscellaneous minor maintenance was required at the completion of Phase I (initial nuclear heating) of the power operation test program. This included 1) repairs to the cleanup system flow control valve, 2) preventive maintenance on control rod selector valves, and 3) checks and adjustments to the refueling building ventilation system flow balance.

Outage No. 2

Outage period - 1:47 a.m., 4-24-63 to 2:15 p.m., 5-2-63
Duration of outage - 204 hours - 28 minutes (8.54 days)
Type of outage - Scheduled
Reason for outage - Miscellaneous maintenance and modifications were required at the completion of Phase III (testing at 125 MWt) of the power operation test program. The major work during this outage was 1) modification of the drywell air cooling system, 2) the repair of a small leak on one of the control rod drive vessel flanges, 3) additional tensioning of reactor vessel head flange studs 4) inspection and repair of turbine bypass valve nozzles and steam baffles inside the main condenser and 5) improvement of shielding in front of the suppression chamber man-hole openings.

Outage No. 3

Outage period - 9:10 a.m., 5-5-63 to 1:11 a.m., 5-7-63
Duration of outage - 40 hours - 1 minute (1.67 days)
Type of outage - Scheduled
Reason for outage - Miscellaneous maintenance and modifications were required at the completion of Phase IV (testing at rated power) of the power operation test program. The major work during this outage was 1) the performance of a turbine hot control valve clearance check, 2) the calibration of feedwater heater level controls and 3) the inspection and cleaning of water side of the main condenser to assure proper cleanliness for the 100 hour rated power demonstration run.

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* The principal maintenance performed and the principal changes made during these outages is further discussed in Appendices III and IV.

Outages During "Shakedown" Period Prior to Commercial Operation

Outage No. 4

Outage period - 4:32 p.m., to 5:43 p.m., 5-17-63
Duration of outage - 1 hour - 11 minutes (0.05 days)
Type of outage - Forced
Reason for outage - A reactor scram occurred as Unit load was being increased to rated power. Load had been previously reduced to inspect steam leakage at the horizontal joint of the high pressure turbine. At the time of the scram the reactor power was 135 MWt and electrical load was 41 MWe. Thorough investigation of plant conditions did not reveal any reason for the scram. It was concluded that spurious, simultaneous operation of two of the picoammeter trip circuits had occurred but that the relatively slow scram annunciator relays had not functioned. One previous spurious scram of this type occurred during the power operation test program. Plant and vendor personnel have completely overhauled all picoammeter trip circuits since the time of this scram. A number of weak components were replaced and a minor modification was made to the circuit to improve its reliability. No additional scrams of this type have occurred since that time.

Outage No. 5

Outage period - 9:29 p.m., 5-17-63 to 7:55 a.m., 6-9-63
Duration of outage - 538 hours 26 minutes (22.44 days)
Type of outage - Scheduled
Reason for outage - The principal reasons for the outage were to inspect the horizontal joint of the high pressure turbine in order to determine the cause of and correct steam leakage across the joint and to install permanent neutron shielding around the eight electrical and piping penetrations in the drywell lower head. Other major work during the outage was 1) the inspection and repair of the protective lining in the emergency condenser shell, 2) inspection and additional repairs to the turbine bypass valve steam baffles inside the main condenser, 3) external inspection of the reactor vessel head to verify that the outer O-ring was holding during operation and 4) relocation of two of the liquid process monitors to areas of lower background radiation.

Outage No. 6

Outage period - 12:25 a.m., to 3:07 a.m., 6-12-63
Duration of outage - 2 hours 42 minutes (0.10 days)
Type of outage - Forced
Reason for outage - A high flux reactor scram occurred as a result of a reactor pressure transient. At the time of the scram the reactor was operating at 165 MWt and the electrical load was 52 MWe. The turbine initial pressure regulator had been taken out of service to make adjustments to its control characteristics. This work was being performed by vendor personnel. The adjustment resulted in much narrower regulation than intended and when the initial pressure regulator was returned to service it had the effect of momentarily closing the turbine control valves. The resulting pressure transient caused the high flux scram. As a result of this scram plant operating procedures have been changed so that the initial pressure regulator is placed in service at reduced power whenever adjustments of this type are being made.

Outage No. 7

Outage period - 8:51 PM, 6/21/63 to 12:15 PM, 7/31/63
Duration of outage 951 hours 24 minutes (39.64 days)
Type of outage - Scheduled
Reason for outage - The principal reason for this outage was to permit vendor personnel to investigate the problem of leakage past the inner O-ring of the reactor vessel head closure. Other major work during the outage was, 1) the removal of 36 of the initial 76 poison curtains, 2) the removal of the special flow instrumented fuel assembly and its replacement with a new fuel assembly, 3) the gamma scanning of approximately half the fuel assemblies in the core, 4) the performance of additional pulse neutron experiments by General Electric Vallecitos Atomic Laboratory personnel, 5) the installation of higher capacity drywell air cooler fans, 6) completion of modifications to the turbine bypass valve steam baffles in the main condenser and 7) the installation of additional neutron and gamma shielding in the area under the drywell and the installation of a shield wall around the air ejector.

Outages Since the Unit Was Placed In
Commercial Operation On 8/1/63

Outage No. 8

Outage period - 9:38 PM, 8/12/63 to 8:03 AM, 8/15/63
Duration of outage - 58 hours 25 minutes (2.43 days)
Type of outage - Scheduled
Reason for outage - During the startup following Outage No. 7 one of the turbine steam bypass valves stuck in an intermediate position during testing of the system. The manual stop valves ahead of the bypass valves were closed during this testing. The startup continued but the stuck valve was left in the isolated condition until this outage at which time repairs were made.

APPENDIX II

RADIATION SURVEY RESULTS

The radiation survey results presented in this appendix were obtained using a variety of instrumentation which included control film badges, stray radiation ionization chambers, "cutie pie" beta gamma dose rate instruments and a double moderator BF_3 neutron dose rate instrument. Table I of this appendix summarizes significant current radiation survey results during normal operation at 165 MWT. It should be noted that many of these measurements were made in areas which are accessible during operation only with the approval of appropriate plant supervision and under the conditions of a Special Work Permit. Areas of this type are designed by an asterisk (*). Table II summerizes significant survey results during shutdown periods following power operation.

TABLE I - RADIATION SURVEY RESULTS DURING NORMAL OPERATION AT 165 MWt

Location	Gramma mr/hr	Neutron mrem/hr
Refueling Building		
Elevation (+) 12' above shield plug	<2	<1
Caisson Access Shaft		
*Elevation (-) 2' cleanup heat exchanger room	65	<1
*Elevation (-) 14' shutdown system room	65	<1
Elevation (-) 24'	5	8
Elevation (-) 34' in front of suppression chamber manhole	10	<1
Elevation (-) 34' in front of manlift	2	<1
Elevation (-) 44'	<1	<1
Elevation (-) 54'	<1	2
Elevation (-) 56' at high radiation area barrier	3	3
*Elevation (-) 66' under the drywell	110	19
Pipe Tunnel		
*Elevation (+) 6' near stairs to pipe gallery	200	17.5
*Elevation (-) 14' at drywell wall	60	29
*Surface of main steam line	1200	-
Air Ejector Area		
*Surface of air ejector after-condenser	10,000	-
Turbine Area		
*Front standard area	32	-
*Surface of high pressure turbine	90	-
*Surface of low pressure turbine	85	-

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TABLE II - RADIATION SURVEY RESULTS DURING SHUTDOWN PERIODS FOLLOWING POWER OPERATION

Location	Gamma mr/hr
Drywell	
Reactor vessel flange area after head removal	3-10
Control rod drive maintenance area below reactor vessel	3-20
Turbine Plant Equipment In Areas Which Are Not Normally Accessiole During Operation	
Surface of condensate demineralizers	8-20
Other turbine plant equipment (Turbine, condenser and auxiliary equipment which has radioactive steam and water passing through it during operation)	<.1
Fuel Transfer Cask Containing An Irradiated Fuel Assembly	
Side of cask at fuel assembly centerline in area door actuator sleeve	10
Maximum reading other than area described above	2

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APPENDIX III
LEVELS OF RADIOACTIVITY IN PRINCIPAL SYSTEMS

This Appendix presents typical values for the concentration of radioactivity in the principal systems associated with the Unit. The counting technique used to obtain these values consists of counting prepared samples (contained in small glass vials or Petri dishes) with a well type gamma scintillation detector system. A counter efficiency based on a gamma energy of 0.662 Mev and a one gamma photon per disintegration decay scheme is assumed to convert count rate to microcuries. "Crud" results are expressed as µc/ml per turbidity unit and are approximately proportional to µc per unit weight of filtrable suspended material. The sample preparation technique involves measuring sample turbidity in units of parts per million suspended silica. The sample is then filtered and the "crud" sample is counted. The microcuries of "crud" are then divided by the product of the volume of sample filtered and the sample turbidity to obtain "crud" activity.

The typical values for the concentration of radioactivity presented below were obtained during periods of sustained power operation at 165 MWt except where noted.

Nuclear Steam Supply System - The principal (accounting for >98% of the total radioactivity) radionuclides in reactor water filtrate seven days after sampling are Cr-51, Mo-99, and Cu-64. The principal radionuclides in reactor water "crud" seven days after sampling are Cr-51, Cu-64, Fe-59, Mo-99, Co-58, Mn-54, Zn-65, and Co-60.

	Time Since Sampling: 2 hours	Time Since Sampling: 168 hours
Reactor Water		
Filtrate - µc/ml	3.2×10^{-2}	6.8×10^{-5}
Crud - µc/ml per turbidity unit	4.5×10^{-3}	8.2×10^{-5}
Gross Radioiodine - µc/ml	1.6×10^{-4}	-
Reactor Water After Cleanup Demineralizer		
Filtrate - µc/ml	2.7×10^{-4}	6.8×10^{-6}
Crud - µc/ml per turbidity unit	9.1×10^{-5}	1.6×10^{-6}

Condensate and Feedwater System - The principal long lived radionuclides in condensate before and after the condensate demineralizer are the same as in reactor water. These reach the condenser hotwell by two principal mechanisms: reactor steam carryover and the return of reactor water through the cleanup system to the condenser hotwell during startup.

	Time Since Sampling: 2 hours	Time Since Sampling: 168 hours
Condensate Before Condensate Demineralizer		
Filtrate - µc/ml	7.7×10^{-4}	3.6×10^{-6}
Crud - µc/ml per turbidity unit	1.3×10^{-4}	3.2×10^{-7}
Condensate After Condensate Demineralizer		
Filtrate - µc/ml	2.2×10^{-5}	9.1×10^{-6}
Crud - µc/ml per turbidity unit	1.1×10^{-5}	2.7×10^{-7}

Closed Cooling Water System - The principal radionuclides in the potassium chromate treated closed cooling water are K-42 and Cr-51. These result from use of this water in the reactor biological shield cooling coils.

Closed Cooling Water Gross Acitvity - μc/ml $1.2X10^{-5}$

Suppression Pool - The principal radionuclides in the sodium chromate treated suppression pool water are Na-24 and Cr-51.

Suppression Pool Water Gross Activity - μc/ml $1.4X10^{-5}$

Spent Fuel Storage Pool - The results given below for spent fuel storage pool water were obtained during the transfer of the instrumented fuel assembly and the 36 poison curtains from the reactor to the pool. Principal radionuclides are activated corrosion products from the reactor.

Spent Fuel Storage Pool Water

Filtrate - μc/ml	$6.8X10^{-6}$
Crud - μc/ml per turbidity unit	$7.3X10^{-6}$

APPENDIX IV
OFF-SITE SHIPMENTS OF
RADIOACTIVE MATERIALS

Off-site shipments of radioactive materials to date are listed below:

Shipment No.	Date of Shipment	Transfer License Nos. From	Transfer License Nos. To	Radioactive Material
1.	10-19-62	DPR-7	General Electric Co., APED, San Jose, Calif. SNM-54	Fuel Assemblies 1,359,390.16 gms U02 containing 35,045.07 gms U-235(3398.55 mc)
2.	12-4-62	DPR-7	General Electric Co., APED, San Jose, Calif. SNM-54	Fuel assemblies 80,160.14 gms U02 containing 2,066.53 gms U-235(160.3 mc)
3.	1-25-63	Byproduct License AEC4-8134-2	Tracerlab Richmond, Calif. AEC4-911-3H-62	Calibration sources Co-60(0.02 mc) Cs-137(35 mc)
4.	5-7-63	DPR-7	General Electric Co., APED, San Jose, Calif. SNM-54	Incore ion chamber assembly 0.12 gms. U-235 ($<$ 0.1 mc)
5.	6-4-63	DPR-7	General Electric Co., VAL, Pleasanton, Calif. SNM-54	Fuel rods from a fuel assembly 3,263.50 gms U02 containing 84.13 gms U-235(7.5 mc)
6.	6-7-63	DPR-7	General Electric Co., VAL, Pleasanton, Calif. SNM-54	Plutonium-beryllium neutron source Pu-239 (1 curie)
7.	7-10-63	DPR-7 and Byproduct License AEC4-8134-2	General Electric Co., VAL, Pleasanton, Calif.-State of California License 0017-60	Vessel head flange O-rings sections Ag-110 (1.35 μc)
8.	7-23-63	DPR-7	General Electric Co., VAL, Pleasanton Calif.-State of California License 0017-60	Pulsed neutron sources H-3 (7.5 curies)

APPENDIX V

PRINCIPAL MAINTENANCE PERFORMED

Mechanical, electrical and instrument maintenance programs were established as soon as operation of the Unit commenced. Most maintenance performed to date falls into the category of minor routine preventive maintenance including a lubrication of equipment, repacking of valves, and routine inspection and servicing of mechanical and electrical equipment and instrumentation systems.

Principal maintenance performed during the report period is as follows:

1. Inspection and adjustment of clean-up system flow control valve. Difficulty with proper stroking and tight shutoff of the remote manual control air operated clean-up flow control valve was experienced during the power operation test program. The valve was disassembled and inspected for the presence of foreign material during Outage No. 1.* The valve was found to be in good condition and after adjustment of valve stroking satisfactory operation resulted.

2. Repair of weld leak on control rod drive flange. A small leak was observed on the flange of control rod drive B-4 during Outage No. 2. The drive was removed and the flange area was inspected by one of the vendor's metallurgists. The cause of leakage was found to be a minor imperfection in the seal weld where one of the hydraulic lines penetrates the vessel control rod nozzle flange. The leakage area was ground out and rewelded.

3. Adjustment of feedwater heater level controls. Feedwater heater level control was erratic during the early part of the power operation test program. Based on performance information obtained during operation, final adjustment and calibration of the control systems was made during Outage No. 3 in preparation for feedwater heater performance testing during the 100-hour rated power warranty run.

4. Turbine maintenance. During the early part of the power test program, difficulty was experienced in obtaining satisfactory performance of the turbine initial pressure regulator. This was traced to a small amount of friction and lost motion in the turbine hydraulic control system. It was found necessary to disassemble, clean and adjust various components of this system. This work occurred over the period of several weeks. Final adjustments were completed on May 6 and operation of the initial pressure regulator has been satisfactory since that time.

Steam leakage in the vicinity of the high pressure turbine flange was observed during full load operation of the unit shortly after the completion of the power operation test program. The high pressure turbine case was lifted for inspection during Outage No. 5. Evidence of leakage in the vicinity of the second and third stage was found although no damage to the joint surface had occurred. This problem was corrected by the replacement of three studs and nuts on each side of the casing with high tensile strength studs and the use of greater stud tensioning in this area of the joint. At the same time other minor sources of steam and lube oil leakage were corrected and the fine mesh start-up screen was removed from the turbine stop valve.

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* Outages are discussed in Appendix I.

5. <u>Reactor nuclear instrumentation</u>. Reactor nuclear instrumentation was subjected to the equivalent of several years of normal operation during plant start-up. A systematic program of inspection and servicing of all components of the nuclear instrumentation systems was undertaken after the completion of the power operation test program and was recently completed. This program included the installation of several minor modifications which were recommended by the vendor to improve the performance of this equipment.

6. <u>Spent fuel pool liner leakage</u>. The stainless steel spent fuel pool liner described in Item 1 of Appendix VI was completed on June 14. The pool was checked for leakage and then placed in service. Subsequent to the transfer of the instrumented fuel assembly and the 36 poison curtains during Outage No. 7, a small amount of leakage was observed in the test sump. This gradually increased to approximately 200 gallons per day by the time preparations were completed to drain the pool and correct the leakage problem. A tank was fabricated and installed in the pool to provide the necessary water shielding for the poison curtains during this work. The instrumentated fuel assembly was stored in the fuel transfer cask. To date, several minor weld leaks have been found and corrected in the corners of the main section of the pool. Work is currently in progress to inspect the fuel shipment cask sump portion of the pool. It is believed that the major source of leakage is defective welds in the liner of this sump.

7. <u>Repair of emergency condenser protective coating</u>. The shell of the emergency condenser was originally lined with a Carboline protective coating. Inspection revealed that the outer two layers of the three-coat application were badly cracked and large sections had peeled off. This was a result of temperature cycling during the emergency condenser tests which were a part of the power operation test program. During Outage No. 7, the outer two layers were stripped off and a second prime coat was applied. The vendor has stated that adequate service life can be expected from a properly applied prime coating.

8. <u>Investigation of leakage past reactor vessel head closure inner O-ring</u>. During the initial nuclear heating phase of the power operation test program, the reactor head leak detection system showed the inner O-ring seal was leaking at a rate approximately an order of magnitude greater than the design leakage rate of 1.0 pound, per hour. The leakage rate test system was valved out at that time to prevent unnecessary cutting of the sealing surface. During Outage No. 2, the drywell was opened and the reactor head bolting was tensioned an additional amount at the recommendation of the manufacturer. Subsequent operation showed that the leakage rate past the inner O-ring was still greater than design. Visual inspection during Outage No. 2 and later during Outage No. 5 verified that no leakage past the outer O-ring was occurring during operation. Radiochemical tests on samples of drywell air and water were also made on several occasions during operation to verify that external leakage was not occurring.

During Outage No. 7, the reactor vessel head was removed and the problem of head leakage was thoroughly investigated by plant and vendor personnel. Visual inspection of the vessel flange sealing area showed that leakage was occurring across the contact area between the inner O-ring and the vessel flange surface in five locations in the vicinity of studs Nos. 22 through 25. No evidence of leakage was found across the contact area between either O-ring and the head flange surface. Detailed metallurgical examination of the five leakage areas on the vessel flange revealed that each of these was associated with a factory weld repair of the Inconel weld overlay flange surface. These repairs were made prior to final machining of the vessel flange surface. It was noted that very fine radial cracks were present in the parent metal adjacent to these patches and that the patches were somewhat undercut. Ultrasonic examination of the cracks showed that they were 5 to 10 mils in depth. The vendor is of the opinion that the defects were present since the time of manufacture but were so fine that they were not evident until iron oxide stains and cutting from steam leakage had occurred. The flange was stoned in the area of leakage to remove the undercutting and cracks. The last crack disappeared when approximately 10 mils of metal had been removed. The flange surface was finished by blending the area of metal removal over about 90° of the flange.

The next phase of the investigation was to determine the amount of flange rotation which occurs as the head closure studs are tensioned. Prussian blue and red lead were applied to the flange surfaces and pieces of 1 mil thick mylar tape were laid at varying radial distances between the inside of the flange and inner O-ring bearing surface. The head was then put in place and dial indicators were positioned to measure the movement of the outside of the flange. The head closure studs were then tensioned. Dial indicator readings and the fact that all mylar tapes could be pulled indicated that flange rotation was occurring. The vendor indicated that although flange rotation was greater than predicted, the closure should seal if O-ring resiliency was proper. The next step was to remove the head, install new O-rings, re-install the head and prepare the vessel for hydrostatic test. Additional flange rotation measurements made with dial indicators showed that no appreciable additional rotation occurred as the vessel was pressurized to 1200 psig. Leakage measurements past the inner O-ring during this hydrostatic test showed that a small amount of leakage was probably occurring.

The recommendation of the vendor was that the reactor be returned to service to ascertain if the repairs which had been made would correct the leakage problem. Leakage of about 3 pounds per hour past the inner O-ring was observed shortly after the reactor reached rated pressure. The inner O-ring leak detection system was secured to prevent cutting the vessel flange. Operation of the reactor will continue as long as there is no evidence of leakage past the outer O-ring. The vendor is continuing to study this problem and is in the process of fabricating the necessary tools to perform machine work on the flanges. It is expected that this will consist of machining the vessel flange to return it to design flatness and machining a taper in either the vessel or head flange to compensate for the head rotation which occurs when the closure studs are tensioned. This work will take place during a future outage.

9. Turbine steam bypass valve repair. One of the turbine steam bypass valves was disassembled and inspected during Outage No. 8 to determine the reason for it jamming in an intermediate position. The valve stem and guide bushing were found to be badly galled. This condition was apparently caused by foreign material getting between the stem and bushing. The running surfaces were cleaned and polished and the valve was reassembled. Test operation showed that the valve operated smoothly.

APPENDIX VI

PRINCIPAL CHANGES MADE IN FACILITY

The changes described below were completed between April 23 and August 16, 1963. These changes were reviewed by the General Office Review Committee or by the On-Site Review Committee as appropriate and as required by Section IX-C of the Technical Specifications. None of the changes were found to involve a change in the Technical Specifications or an "unreviewed safety question".

1. Installation of a spent fuel pool liner - The spent fuel storage pool was originally provided with a carboline lining. In order to minimize maintenance and eliminate the potential for leakage from the pool a stainless steel liner was installed over the existing carboline coating. The walls of the liner are 10 gauge stainless steel sheet and the pool floor consists of 1/4 inch stainless steel plate. The walls of the stainless steel liner are off-set from the walls of the pool and are anchored at the top of the fuel pool. A sump has been provided at the bottom of the shipping cask pit. Any water which collects between the carboline coating and the stainless steel liner drains to this sump and is pumped to radwaste. This provides a method of monitoring for any pool leakage.

2. Modifications to improve drywell cooling - The capacity of both drywell air coolers was initially inadequate to consistently maintain maximum drywell air temperatures below the 175°F license limit. To prevent exceeding this limit reactor operation had to be suspended several times during the power test program to allow the drywell to cool. A turning vane was installed in the suction duct work ahead of the cooling fans and one of the fan suction boxes was modified during Outage No. 2 to give better cooling air flow characteristics and greater fan capacity. Although this modification improved performance considerably, fan capacity was still less than design. The two fans were therefore replaced by new larger capacity fans during Outage No. 7. At the same time the suction duct work ahead of the cooling fans was extended to the area above the reactor vessel head to provide better circulation through the drywell. Tests subsequent to this outage showed that performance was satisfactory if both fans and coolers were in service but that a single fan and cooler was inadequate to maintain temperature. Investigation of this problem is continuing.

3. Additional neutron and gamma shielding for the refueling building caisson access shaft and El.-66 ft. areas - The suppression pool manhole openings and the electrical penetrations and piping penetration nozzles in the bottom of the drywell vessel provided a leakage path for neutrons. Metal cans filled with polyethylene beads were fabricated to fill the manhole openings and to cover the drywell penetrations. Blocks prefabricated from an epoxy resin and polyethylene beads were fitted into the space between the lower section of the drywell vessel (just above the lower head) and the concrete caisson. A 2 inch layer of borated epoxy resin was then installed below these blocks. Additional neutron shielding for the drywell lower head was provided by 2 inch thick sheets of borated epoxy resin which were fitted to the head. The coolant cleanup supply line at el.-66 ft. was provided with lead shielding to reduce the gamma radiation field in the area below the drywell.

4. Installation of a reactor vessel head flange leak detection system for detecting leakage past the outer O-ring seal - This leak detection system was installed during Outage No. 7. It supplements the existing system for detecting leakage past the inner O-ring and consists of asbestos packing tamped into the gap between the reactor head and vessel flanges and a metal band clamped around the outside circumference of the flange to seal off the gap. Piping connects this gap to a level switch alarm system located outside the drywell. A vent is provided inside of the drywell to prevent pressurizing the system. This system required a new drywell penetration which was made in accordance with Section III.B.7 of the Technical Specifications.

5. Modifications to the turbine steam bypass system - Inspection of the steam bypass section of the main condenser following the use of the turbine bypass valve system during the power test program revealed that a number of welds in the baffling had failed. The baffles were strengthened and additional baffles were added to protect the condenser tubes. This work took place during Outages No. 2, 5 and 7.

6. Air ejector area shield wall - A removable shield wall constructed of high density concrete blocks has been provided on the north and east sides of the main condenser air ejector and gland seal exhauster. This permits access to the main condenser area without being exposed to the high radiation field that exists around the air ejector during power operation.

7. Modifications to the liquid process radiation monitoring system - Two of the detectors were relocated because the radiation background during power operation at their original locations was too high. These were the condensate to demineralizer detector which was moved from the section of condensate line in the main air ejector room to a section of the same line in the condensate demineralizer room and the closed cooling water return to storage tank detector which was moved from a run of cooling water return line adjacent to the main condenser to a thimble inside of the closed cooling water return tank. A less sensitive scintillation detector was installed in the reactor water to cleanup demineralizer channel.

8. Off-gas monitoring systems - Both off-gas monitoring channels proved to be more sensitive than design. Reduction of sensitivity was accomplished by providing new samplers for these systems. The principal features of the new samplers which give the desired decrease in sensitivity are smaller sample volume, additional lead shielding to reduce the background at the detector and increased shielding of the detectors so that they are exposed to only a collimated radiation beam from the sample.

9. Modification of instrumented fuel assembly - The instrumented fuel assembly used for measurement of core flow during the power operation test program was removed from the core and replaced with a new fuel assembly. The instrumented assembly was placed in the spent fuel pool and modified by removing the flow meter and its signal cable. A new handle was also installed on the fuel assembly because the original handle was damaged during this work. When this work was complete, the fuel assembly was given a standard inspection and was found to meet all mechanical specifications. No unusual or significant deposition of corrosion products were observed.

APPENDIX VII

DISCUSSION OF SIGNIFICANT TESTS

This appendix discusses significant tests which were performed and present an analysis of test results which have been completed during the report period.

1. Reactor critical testing incident to the removal of poison curtains. During Outage No. 7, 36 of the initial 76 poison curtains were removed. The curtain removal procedure used during this work complied in all respects with the requirements related to core alterations which increase reactivity as described in the Technical Specifications. Curtains were first removed from the southeast quadrant of the core. Each step (a step consisted of removal of curtains from one cell) was followed by a cold shutdown margin check. When curtain removal was completed in this quadrant, a series of critical tests using adjacent pairs of control rods were performed. These showed that the cold shutdown margin requirement would be easily met after the removal of the remaining curtains. The curtain removal procedure beginning at this point consisted of removing curtains from all cells in a quadrant before a cold shutdown margin check was made. A subcriticality check, however, was always made prior to and after the removal of curtains from each cell. When all 36 curtains were removed another series of critical tests using adjacent pairs of control rods were performed. It was found that the strongest rod and strongest adjacent rod consisted of the four symmetrically equivalent pairs of rods of the C-1, D-1 type. With one rod of this type fully withdrawn and the adjacent rod withdrawn to notch 9, the reactor is approximately 0.1% ΔK subcritical. The cold shutdown margin requirement is that the reactor must be subcritical with the adjacent rod at notch 7. The results of this critical testing showed that core excess reactivity had been increased approximately 2% ΔK which agrees with calculations performed prior to curtain removal. When this testing was complete, a modified normal start-up rod pattern was checked to demonstrate satisfactory start-up channel response.

2. Gamma scanning of fuel assemblies for the purpose of determining core power distribution. The reactor was operated with essentially a constant control rod pattern during approximately 20 full power days prior to Outage No. 7. This rod pattern was symmetrical around the centerline of the core. During Outage No. 7, all unique fuel assemblies in one-half of the core were placed in a special fixture which was attached to the core chimney and were gamma scanned at two-inch axial intervals using a miniature gamma sensitive ionization chamber. A selected fuel assembly was probed once each 24 hours to provide decay correction data. Selected fuel assemblies in the other half of the core were also probed to check the validity of the assumption of core symmetry.

The data was reduced to give axial and radial peak to average values for each fuel assembly in the core. The "hot channel" was found and the gross peaking factor was calculated to be 2.08. This result, when compared with the gross peaking factor of 1.97 obtained from wire irradiation data, shows good agreement between these two experimental methods.

3. Subcritical core pulsed neutron tests. The series of core pulsed neutron tests which were made during the initial loading of the reactor were continued during Outage No. 7. The main purpose of this work was to investigate the effect of the high gamma field in the vicinity of the irradiated core on the pulse neutron source and the special neutron detectors employed in the test work. The results of this testing which is being performed by General Electric Vallecitos Atomic Laboratory personnel under contract to the AEC will be reported by General Electric.

4. Routine operational testing of the Unit. All routine operational tests required by the Technical Specification (Table IX-1) and plant operating procedures were performed prior to initial operation of the Unit. These have been performed periodically since that time in accordance with established procedures. This testing has revealed three problems to date:

a. Control rod selector valves are routinely checked for the need for maintenance by operating them at reduced air pressure. On several occasions, selector valves have shown a tendency to be sticky during this testing and have been removed and serviced.

b. The emergency engine generator failed to start during its routine monthly operational test in March. An electrical fault had occurred in the engine starting circuit which caused the battery to discharge. This fault was promptly corrected and satisfactory operation has been observed during subsequent testing.

c. During the performance of quarterly control rod tests during Outage No. 5, difficulty was experienced with control rod E-5 in obtaining proper flow control valve adjustment to meet withdrawal and insertion time limits. Satisfactory adjustment was finally obtained prior to reactor start-up. Difficulty was experienced with excessively slow withdrawal time several days later. (Insertion time was normal.) During an attempt to adjust withdrawal time, the adjustment screw was broken off resulting in a hydraulic water leak. After a review of the situation by the On-site Review Committee a "soft patch" was installed to stop the leak and the drive was isolated to prevent further withdrawal. During Outage No. 7, the flow control valve was removed and inspected. No apparent problems were found but a new valve was installed. The drive timing was then adjusted. Operation has been satisfactory since that time.

APPENDIX VIII

METEOROLOGICAL DATA

This appendix presents a tabulation and discussion of the important meteorological data collected to date and a brief description of the data reduction procedure.

Table 1 presents data in a comparable form to Table 18, Appendix I of the Final Hazards Summary Report (FHSR) which was the basis of diffusion calculations. A comparison shows that nearly all the percentages of stable cases derived from tower data are lower than reported in the FHSR, while wind speeds and frequencies of northerly winds are somewhat higher.

Tables 2 and 3 show the effect of season on the stabilities given in Table 1.

Tables 4 and 5 present data comparable to the data in Tables 1 and 2 of the FHSR.

Tables 6 and 7 present the diurnal variation of stability and wind direction. The prevailing wind direction in the dry season is northerly at all hours. In the wet season it is northerly from mid-morning until evening, then south-southeasterly through the night and early morning. South-southeast as a favored direction in the wet season is not inconsistent with Figure 1 in the FHSR, or Table 4 of this appendix. Observer bias in favor of the primary directions obscures the true situation in Figure 1 of the FHSR. Table 4 is not representative of the entire wet season since the anemometer failed in high winds during October and was out of service through December.

Present meteorological instrumentation at the site provides for data collection by means of strip chart recorders and an automatic data logging system. Strip chart records of wind speed, direction and temperature have been obtained since April, 1962. Temperature differences between the top of the tower and the 7 foot elevation and ten minute averages of wind speed and direction at the top were read at hourly intervals. These data were sorted and tabulated in various ways to develop the seven tables in this Appendix. The automatic data logging system is expected to be in routine operation in the near future. This equipment will greatly facilitate data reduction.

TABLE 1

JOINT WIND AND STABILITY,
COMBINED DRY AND WET SEASON,
1962-1963

Distribution of stability for each wind direction in percent or total frequency of winds from each direction.

		STABLE		UNSTABLE (c)	
Wind Direction	Frequency(a) In Percent	Frequency In Percent	Average (b) Wind Velocity	Frequency In Percent	Average (b) Wind Velocity
N	23.3	10	7.9	90	12.8
NNE	8.0	19	8.1	81	8.1
NE	4.2	30	5.2	70	5.8
ENE	3.1	40	4.4	60	4.8
E	3.5	44	5.1	56	5.5
ESE	2.6	48	5.1	52	8.7
SE	4.2	46	5.6	54	14.6
SSE	8.1	33	4.3	67	10.6
S	5.9	38	6.6	62	10.6
SSW	5.8	34	7.6	66	10.8
SW	5.6	25	6.6	75	9.2
WSW	3.1	23	4.4	77	7.0
W	3.5	15	3.9	85	6.4
WNW	2.3	33	4.9	67	5.4
NW	5.6	20	5.7	80	7.3
NNW	11.1	13	8.6	87	10.1

(a), 4521 joint observations of stability and direction.

(b), 3169 joint observations of wind velocity, stability, and direction.

(c), Temperature at Top-Temperature at 7 feet less than or equal to zero.

TABLE 2

JOINT WIND AND STABILITY,
DRY SEASON, 1962

Distribution of stability for each wind direction in percent of total frequency of winds from each direction.

		STABLE		UNSTABLE(c)	
Wind Direction	Frequency(a) In Percent	Frequency In Percent	Average (b) Wind Velocity	Frequency In Percent	Average (b) Wind Velocity
N	32.0	10	8.4	90	13.0
NNE	8.8	20	9.6	80	9.1
NE	2.8	25	4.8	75	5.9
ENE	1.7	34	4.2	66	4.3
E	1.3	44	4.8	56	3.8
ESE	1.1	36	5.4	64	5.5
SE	1.2	40	4.0	60	4.4
SSE	1.2	30	3.8	70	7.0
S	4.3	19	6.9	81	9.2
SSW	4.8	38	7.0	62	7.7
SW	5.4	23	5.5	77	8.7
WSW	3.8	16	4.6	84	7.0
W	4.4	11	4.1	89	5.9
WNW	3.2	35	5.0	65	5.1
NW	7.8	24	5.7	76	7.0
NNW	16.1	16	8.7	84	9.8

(a), 2448 joint observations of stability and direction.

(b), 2358 joint observations of wind velocity, stability, and direction.

(c), Temperature at top - Temperature at 7 feet less than or equal to zero.

TABLE 3

JOINT WIND AND STABILITY,
WET SEASON 1962-63

Distribution of stability for each wind direction in percent of total frequency of winds from each direction.

		STABLE		UNSTABLE	
Wind Direction	Frequency(a) In Percent	Frequency In Percent	Average (b) Wind Velocity	Frequency In Percent	Average (b) Wind Velocity
N	12.9	12	4.5	88	11.0
NNE	7.0	18	4.1	82	4.6
NE	5.9	33	5.6	67	5.7
ENE	4.7	42	4.6	58	5.5
E	6.2	44	5.3	56	6.3
ESE	4.3	52	4.9	48	11.5
SE	7.8	46	6.6	53	18.8
SSE	16.3	33	4.9	66	12.8
S	7.7	51	5.9	49	13.6
SSW	7.0	30	10.4	70	15.7
SW	5.7	27	18.0	73	11.0
WSW	2.4	35	3.3	65	7.5
W	2.4	22	2.0	78	8.1
WNW	1.4	29	3.0	71	6.9
NW	3.0	10	(d)	90	8.7
NNW	5.3	4	4.0	96	12.2

(a), 2703 joint observations of stability and direction.

(b), 711 joint observations of wind velocity, stability, and direction.

(c), Temperature at Top - Temperature at 7 feet less than or equal to zero.

(d), No cases

TABLE 4

JOINT WIND FREQUENCIES BY DIRECTION AND SPEED CLASSES, WET SEASON 1962-1963

Percent of cases in given wind direction and speed group.

Wind Direction	Speed Groups, MPH(a) Calm(b)	4-15	16-31	32-47	47+	All Speeds
N		10.3	5.4	0.1		15.8
NNE		4.1	0.1	(c)		4.2
NE		5.1	(c)	(c)		5.1
ENE		4.1	(c)	(c)		4.1
E		4.1	(c)	(c)		4.1
ESE		3.5	0.9	(c)		4.4
SE		4.6	4.7	0.3		9.6
SSE		4.0	2.1	(c)		6.1
S		3.3	1.2	0.7		4.2
SSW		3.5	1.9	0.3		5.7
SW		2.4	1.1	(c)		3.5
WSW		1.9	0.2	(c)		2.1
W		2.7	(c)	(c)		2.7
WNW		1.2	(c)	(c)		1.2
NW		2.7	(c)	(c)		2.7
NNW		4.1	2.4	(c)		6.5
All Directions	16.6	61.6	20.0	1.3	(c)	

(a), 1028 joint observations of direction and speed, Jan. 63 - Mar. 63

(b), 0 - 3 MPH

(c), No cases

TABLE 5

JOINT WIND FREQUENCIES BY
DIRECTION AND SPEED CLASSES,
DRY SEASON, 1962

Percent of cases in given wind direction and speed group.

Wind Direction	Speed Groups, MPH(a) Calm(b)	4-15	16-31	32-47	47+	All Speeds
N		21.6	9.9			31.5
NNE		7.7	0.7			8.4
NE		2.5	(c)			2.5
ENE		1.0	(c)			1.0
E		0.9	(c)			0.9
ESE		1.1	(c)			1.1
SE		0.7	(c)			0.7
SSE		0.9	(c)			0.9
S		2.7	0.6			3.3
SSW		4.1	(d)			4.1
SW		3.8	0.7			4.5
WSW		2.6	0.1			2.7
W		3.1	0.1			3.2
WNW		2.3	(c)			2.3
NW		6.4	0.1			6.5
NNW		12.2	1.8			14.0
All Directions	12.4	73.6	14.0	(c)	(c)	

(a), 2687 joint observations of direction and speed, May - August, 1962

(b), 0 - 3 MPH

(c), No cases

(d), Less than 0.1%

TABLE 6

AVERAGE STABILITY AND PREVAILING DIRECTIONS BY HOUR OF THE DAY, WET SEASON, 1962-63

Hour of Day	Mean Temperature Difference (°F)(a)(b)	Prevailing Directions (a)		
		Most Frequent	2nd	3rd
1	+1.1	SSE	SE	S
2	+1.3	SSE	S	E,NNE(c)
3	+1.4	SSE	SE	S
4	+1.3	SSE	E,ENE(c)	S
5	+1.3	SSE	NE	SE
6	+1.6	SSE	SE	N
7	+1.0	SSE	ENE	E,ESE,S(c)
8	+0.9	SSE	SE	S
9	+0.2	SSE	E	N
10	-0.4	SSE	N,NNE,E(c)	SSW
11	-0.8	SSE,N(c)	SW	NNW,SE(c)
12	-1.2	N	NNW	SSE
13	-1.3	N	NNW	SSE
14	-1.1	N	NNW	SW
15	-1.1	N	NNW	SW
16	-0.9	N	SSE	NNW,SW(c)
17	-0.7	N	SSE	NNW
18	0.0	N	SSE	NNE
19	+0.3	N	NNE,NE,S(c)	SSE
20	+0.7	N	SSE	SE
21	+0.8	SSE	SE	N
22	+1.0	SSE	SSW	S
23	+0.9	SSE	SE	N
24	+1.2	SSE	S	SE,ENE(c)

(a), 2073 joint observations of stability and direction

(b), Temperature at Top - Less Temperature at 7 feet

(c), Tied

TABLE 7

AVERAGE STABILITY AND PREVAILING DIRECTIONS BY HOUR OF THE DAY, DRY SEASON, 1962

Hour of Day	Mean Temperature Difference (°F)(a)(b)	Prevailing Directions (a)		
		Most Frequent	2nd	3rd
1	+0.1	N	NNW	SSE
2	+0.1	N	SSW	SW
3	-0.0	N	SSW	S
4	+0.0	N	S	NNE
5	+0.1	N	S	NNE
6	-0.0	N	NNW	NNE
7	-0.2	N	NNW	NNE
8	-0.3	N	NNW	NW
9	-0.2	N	NNW	NW
10	-0.0	N	NNW	NW
11	+0.1	NNW	N	NW
12	+0.1	N	NNW	NW
13	-0.2	N	NNW	NW
14	-0.2	N	NNW	NW
15	-0.4	N	NNW	NW
16	-0.3	N	NNW	NNE
17	-0.3	N	NNW	NNE
18	-0.3	N	NNW	NNE
19	-0.2	N	NNE	NNW
20	-0.3	N	NNE	NNW
21	-0.2	N	NNE	NE
22	-0.1	N	NNE	NNW
23	+0.0	N	NNE	NNW
24	+0.1	N	SSW	NNE

(a), 2448 joint observations of stability and direction

(b), Temperature at Top - Less Temperature at 7 feet