SUMMARY OF PLANT OPERATIONS

6-1 to 6-9		Reactor in cold shutdown (OTSG tube leak).
6-10		Reactor in heat-up mode
6-11		Began deboration to criticality.
6-12	0655	Reactor critical.
	0849	Began increasing power to 5%.
	1235	Holding at 5% power.
	1350	Increasing power to 15%.
	1958	Closed OCB's and began increasing power.
	2015	Holding at 40% power.
6-13	1130	Increasing power to 67%.
	1525	Reactor at 67%, holding.
6-15	1337	Began increasing power to 70%.
	1610	Increasing power at 3% per hour.
	1900	Reactor at 75%, holding.
6-16	0015	Began increasing power at 3% per hour.
	0845	Reducing power to 85% due to instability.
	1015	Increasing power to 87%.
	1128	Reactor at 87%.
6-17	1509	Reactor/Turbine trip (Lost Main Feedwater Pump)
	1948	Reactor in hot shut down.
6-18	0140	Began RCS cooldown for replacement of CRD stator.
	2210	RCS in cold shut down.
6-19	0059	Began RCS heatup.
	0511	Terminated RCS heatup due to CRD cooling water leak.

SUMMARY OF PLANT OPERATIONS (Continued)

6-20	2020	Began pulling rods.
	2159	Reactor Critical
	2315	Reactor at 10% power.
6-21	0200	Closed OCB's.
	0210	Reactor trip (lost main feedwater pump).
	0354	Began pulling rods.
	0416	Reactor critical.
	0716	Closed OCB's. Began increasing power to 40%.
	1600	Reactor at 44%, increasing power to 50%.
6-22		Reactor at 50% power.
6-23	1147	Reactor trip (lost main feedwater pump)
	1326	Began pulling rods.
	1358	Reactor critical.
	1656	Closed OCB's and began increasing power.
6-24		Reactor at 50% power.
6-25		Began increasing power to 70% @ 3% per hour.
6-26		Reactor at 70% power.
6-27 to 6-30		Reactor at approximately 75% power.

PERSONNEL CHANGES REQUIRING REPORTING

No personnel changes that require reporting in accordance with Technical Specifications Figure 6.9.2 were made in June, 1981.

SUMMARY OF CHANGES IN ACCORDANCE WITH 10 CFR 50.59(b)

1. Added head rigging with clevises to the reactor vessel head. The modification has been permanently attached to the reactor vessel head to eliminate the critical path outage time presently required for rigging, to lift the vessel head for refueling.

2. Removed the relief valves on the makeup filters (PSV-23007 and PSV-23008) and on the seal return coolers (PSV-24017 and PSV-24018). This was a resasonable step to prevent leakage from systems that contain liquids or gases having large radioactive inventories after a serious transient or accident, as required in NUREG-0578, Section 2.1.6a. Review has indicated that these valves can be removed without endangering the makeup filters, seal return coolers, or associated piping.

3. Removed leaky "B" feedwater cleanup block valve, FWS-022, and sealed the pipe with a cap. The removed valve was not required by any operating or emergency procedures.

4. Completed modifications to allow installation of Movable Incore Detector System (MIDS) for monitoring performance of Axial Blanket Fuel Assemblies.

MAJOR ITEMS OF SAFETY-RELATED MAINTENANCE

1. Repaired loose valve linkage on reactor building personnel hatch outer equalizing valve.

2. Rewound motor of reactor building emergency sump valve.

SECTION 1 - OVERVIEW

Following the fourth refueling of Rancho Seco Unit #1, the startup test program for Cycle 5 was begun with initial criticality established at 0715 hours on May 4, 1981. Zero power physics testing commenced at that time and was successfully completed on May 9, 1981 at 0230 hours. As planned, the Zero power testing program was conducted at the iso-thermal Reactor Coolant temperature of 532°F, and below the power level commensurate with nuclear heat. Power escalation was begun on May 9, 1981 and testing has been completed at 40% and 75% of full power. This final plateau being attained on May 13, 1981.

As of June 30, 1981 the plant has not attained 10 ' of full power due to a return to cold shut down to repair the OTSG tube leak.

The following descriptions of test data and results refer to the Cycle 5 Reload Report, BAW-1667, March 1981 testing commitments and the District's May 7, 1981 response to the Commission's April 10, 1981 request for additional information. Reference is made to that information rather than repeating it here.

SECTION II - PRE-CRITICAL TEST

Control Rod Trip Test

Control rod trip time testing was done prior to establishing initial criticality and while maintaining refueling boron concentration. The conditions were, all four Reactor Coolant pumps running with the Reactor Coolant system established at 532°F and a pressure of 2155 PSIG. All of the safety and regulating control rods, which are assigned to Groups 1 through 7, were fully withdrawn. Group 8 (Axial Power Shaping Rods which do not drop) were established at an intermediate position. Using the manual Reactor trip button to initiate the drop, all 61 droppable control rods were dropped into the core from the fully withdrawn position. Drop time was determined by using the plant computer and measuring the time from "trip" to three-fourths insertion. The fastest rod dropped in 1.163 seconds, and the slowest rod was at 1.2[illegible] seconds. For acceptance, the drop time of Groups 1 through 7 had to be less than 1.66 seconds. The measurement technique includes the control circuit and logic times in addition to the rod travel time. All drop times were well below the acceptance criteria thus meeting the Technical Specifications requirements for full-flow drop time. Confirmation was made that the APSR's (Group 8) did not drop.

SECTION III - ZERO POWER PHYSICS TESTING

.1 All Rods Out Boron Concentration

The All Rods Out (ARO) Boron concentration was measured as described in the Cycle 5 Reload Report.

With control rod Group 8 at 37.5% withdrawn, the results were as follows:

Measured	Vendor Prediction
1201 ppmB	1252 $\pm$ 100 ppmB

The measured data is consistant with the prediction and meets all acceptance criteria.

.2 Boron Concentration at Maximum Controlling Rod Group Insertion Limit

Measured	Vendor Prediction
850 ppmB	868 $\pm$ 100 ppmB

This measurement provides a second just critical Boron concentration measurement corresponding to a predicted value. At the time of this measurement, control rod Groups 5, 6, and 7 were fully inserted and control rod Group 8 positioned at 37.5% withdrawn. The measured data was consistant with predictions and met all acceptance criteria.

.3 Temperature Coefficient of Reactivity at All Rods Out Boron

Measured	Vendor Prediction
-0.40×10^{-4} $\Delta k/k/F^{o}$ at 1217 ppmB	$-0.42 \times 10^{-4} \pm 0.4 \times 10^{-4}$ at 1217 ppmB

The value at this boron concentration met the acceptance criteria of being within the predicted band.

.4 Moderator Coefficient of Reactivity at All Rods Out Boron

The result at 1217 ppmB also met the acceptance criteria for Moderator Coefficient of Reactivity which specifies that, when corrected for fuel doppler effects, the value shall not be more positive than $+0.5 \times 10^{-4}$ $\Delta k/kF^{o}$. The Moderator Coefficient of Reactivity was determined to be -0.20×10^{-4} $\Delta k/k/F^{o}$.

.5 Temperature Coefficient of Reactivity Determined at the Maximum Insertion Boron Concentration

Measured	Vendor Prediction
-1.38×10^{-4} $\Delta k/k/F^{o}$ at 852 ppmB	$-1.132 \times 10^{-4} \pm 0.4 \times 10^{-4}$ $\Delta k/k/F^{o}$ at 852 ppmB

The acceptance criteria for this value is the same as for the ARO temperature coefficient measurement. This measurement met all criteria.

.6 CRA Group Reactivity Worth

	Measured Worth %Δk/k	Vendor Predicted Worth, %Δk/k	Deviation Measured	Deviation Allowed
Group 5	1.117	1.096	-1.9%	± 15%
Group 6	0.989	0.964	-2.51%	± 15%
Group 7	1.389	1.422	-2.36	± 15%
Total	3.495	3.482	-0.37	± 10%

The shutdown margin calculations shown in the Cycle 5 eload Report are substantiated by the above measurements and the excellent agreement between predicted and measured ARO Boron.

.7 Ejected Rod Worth Measurement

Measured Ejected Worth, %Δk/k	Predicted Worth, %Δk/k	Deviation Measured %	Tolerance Allowed
0.58	0.57	-1.72	± 20%

The ejected rod worth is determined for the configuration corresponding to the maximum insertion condition allowed by Technical Specifications, namely, Groups 5, 6, and 7 fully inserted at zero power, with Group 8 at 37.5% WD and all safety rods fully withdrawn. From this configuration, the maximum worth "Ejected Rod," which is a rod in Group 7, was borated to full out and then swapped against Group 5 to return it to the fully inserted position as a second determination of its worth. These two values were then averaged, and are reported as the Measured value. These results are consistent with the prediction and meet the absolute acceptance criteria of Technical Specifications by being less than 1.0 %Δk/k at zero power.

SECTION IV - POWER ESCALATION

.1 Core Power Distribution

Core power distributions have been taken and analyzed at the nominal Reactor power test plateaus of 40% and 75% during Cycle 5 power escalation. The purpose of these measurements was to verify that the minimum DNBR, maximum linear hear rate, quadrant power tilt, power imbalance, and related power peaking factors would not exceed allowable limits. In each case the measured variables were extrapolated to the over-power trip setpoint for the next test plateau so as to assess the margin of conservatism prior to escalation. A summary of the test results follows:

POWER DISTRIBUTION TEST RESULTS

	Measured/Desired	
Date of Data	5/10/81	5/14/81
Power level, %FP	41.0/40	75.0/75
Core Burnup, EFPD	0.35/2.0	2.70/3.0
Group 1-5, %WD	100/100	100/100
Group 6, %WD	100/100	100/100
Group 7, %WD	89.8/87.0	91.0/87.0
Group 8, %WD	34.9/35.3	27.0/28.2
Boron Concentration, ppmB	888/860	715/760
Axial Imbalance, %FP	-0.68/1.01	1.34/0.47
Max Core Quadrant Power Tilt, %FP	1.45/<3.40	1.35/<3.40
Minimum DNBR	9.87/≥1.30	2.80/≥1.30
Worst Case LHR, Kw/ft	4.46/≤20.4	9.03/≤20.4
Max Radial Power Peak	1.264/1.282	1.279/1.267
Max Total Power Peak	1.465/1.520	1.520/1.472
Max Peak at Core Grid	K-10/K-12	K-12/K-12
Max Peak in Fuel Batch Number	7/7	7/7
Equilibrium Xenon	Yes/2D	Yes/2D
Acceptable for Power Escalation	Yes	Yes
Extrapolations done to, %FP	92.1	112.0

Acceptance criteria which applies to the radial and total peaking factors is ± 5% when compared to the predictions for the peak assembly at the 75% power plateau. All acceptance criteria were met, and escalation based upon these results proved to be conservative. The measured DNBR and linear heat rates verified that the Reactor Protective system setpoints provide protection for the core against exceeding transient DNBR and/or maximum linear heat rates assumed in the Safety Analysis and are sufficient to protect against exceeding the limiting Technical Specification LOCA heat rates.

.2 Core Symmetry Test

The core symmetry test was used during this cycle as a method of verifying the symmetry of the core. Previously the symmetric ejected rod method was used to provide this information, The core symmetry test uses the incore instrumentation to evaluate the quadrant tilt from 15% to 40% power. The acceptance criteria is that the tilt be less than 3.4%. The maximum measured tilt in this power interval was -1.73, well below the acceptance criteria maximum.

.3 Power Imbalance Detector Correlation Test

This test is performed to establish the relationship between the out-of-core nuclear instrumentation and the full set of incore self-powered neutron detectors. Both systems provide axial power imbalance data, with the incore system being the standard.

Due to the effect of refueling on the neutron flux exiting the reactor, the out-of-core indication of imbalance is expected to change. Since the nature and magnitude of this change is not easily predicted, this test is performed at a low power level to establish that the relationship between the two systems is conservative. Should it be desired to alter the out-of-core/incore relationship, regaining the out-of-core NI difference amplifier is required.

During this power escalation, the initial results showed the out-of-core Nuclear Instrumentation differencing amplifier had to be regained. Anytime regaining is done, a retest is required. This regaining and retest was accomplished at 40%FP and all applicable acceptance criteria met.

Cycle 5 safety analysis assumes that the correlation slope is greater than or equal to 1.15. This correlation criteria was satisfied on all protective channels, and the relationship between the incore and out-of-core instrumentation is shown to be conservative. At the same time that this data was obtained, the relationship between the full set of incore instrumentation and those on the backup recorders was also determined to meet its acceptance criteria.

SECTION V - SUMMARY

The startup test program was initiated on May 2, 1981. All tests at zero power were completed on May 9, 1981 and the power escalation to 100% was initiated. Testing at 40% and 75% was completed on May 15, 1981. Escalation to 100% was delayed due to the OTSG tube repair.

The results of early Cycle 5 testing provided in this report demonstrates that Rancho Seco Unit 1, Cycle 5, has been properly designed; and that the unit can be operated in a manner that will not endager the health and safety of the public.

OPERATING DATA REPORT

DOCKET NO. 50-312
DATE 81-06-30
COMPLETED BY R. W. Colombo
TELEPHONE (916) 452-3211

OPERATING STATUS

1. Unit Name: Rancho Seco Unit 1
2. Reporting Period: June, 1981
3. Licensed Thermal Power (MWt): 2772
4. Nameplate Rating (Gross MWe): 963
5. Design Electrical Rating (Net MWe): 918
6. Maximum Dependable Capacity (Gross MWe): 917
7. Maximum Dependable Capacity (Net MWe): 873
8. If Changes Occur in Capacity Ratings (Items Number 3 Through 7) Since Last Report, Give Reasons: N/A

Notes

9. Power Level To Which Restricted, If Any (Net MWe): N/A
10. Reasons For Restrictions, If Any: N/A

	This Month	Yr.-to-Date	Cumulative
11. Hours In Reporting Period	720	4,343	54,384
12. Number Of Hours Reactor Was Critical	365.9	1,320.1	32,816.3
13. Reactor Reserve Shutdown Hours	0	0	4,469.6
14. Hours Generator On-Line	342.8	1,232.2	31,477.8
15. Unit Reserve Shutdown Hours	0	0	1,210.2
16. Gross Thermal Energy Generated (MWH)	657,516	2,928,217	79,900,472
17. Gross Electrical Energy Generated (MWH)	176,808	931,166	26,878,880
18. Net Electrical Energy Generated (MWH)	155,552	865,235	25,391,360
19. Unit Service Factor	47.6	28.4	57.9
20. Unit Availability Factor	47.6	28.4	60.1
21. Unit Capacity Factor (Using MDC Net)	24.7	22.8	53.5
22. Unit Capacity Factor (Using DER Net)	23.5	21.7	50.9
23. Unit Forced Outage Rate	52.4	37.4	29.4

24. Shutdowns Scheduled Over Next 6 Months (Type, Date, and Duration of Each): N/A

25. If Shut Down At End Of Report Period, Estimated Date of Startup: N/A

26. Units In Test Status (Prior to Commercial Operation):

	Forecast	Achieved
INITIAL CRITICALITY	N/A	N/A
INITIAL ELECTRICITY	N/A	N/A
COMMERCIAL OPERATION	N/A	N/A

AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-312
UNIT Rancho Seco Unit 1
DATE 81-06-30
COMPLETED BY R. W. Colombo
TELEPHONE (916) 452-3211

MONTH June, 1981

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1	0	17	354
2	0	18	0
3	0	19	0
4	0	20	0
5	0	21	144
6	0	22	306
7	0	23	245
8	0	24	328
9	0	25	426
10	0	26	538
11	0	27	565
12	31	28	580
13	395	29	588
14	510	30	606
15	530	31	
16	623		

INSTRUCTIONS

On this format, list the average daily unit power level in MWe-Net for each day in the reporting month. Compute to the nearest whole megawatt.

(9/77)

UNIT SHUTDOWNS AND POWER REDUCTIONS

DOCKET NO. 50-312
UNIT NAME Rancho Seco Unit 1
DATE 81-06-30
COMPLETED BY R. W. Colombo
TELEPHONE (916) 452-3211

REPORT MONTH June, 1981

No.	Date	Type[1]	Duration (Hours)	Reason[2]	Method of Shutting Down Reactor[3]	Licensee Event Report #	System Code[4]	Component Code[5]	Cause & Corrective Action to Prevent Recurrence
2	81-06-01	F	284	A	1	81-026 01 T	CC	HTEXCH	Continued shutdown due to OTSG tube leak.
3	81-06-17	F	82.9	A	3	N/A	CF	PUMPXX	RC Pressure Trip (Loss of Main Feedwater Pump
4	81-06-21	F	5.1	A	3	N/A	CF	PUMPXX	RC Pressure Trip (Loss of Main Feedwater Pump)
5	81-06-23	F	5.2	A	3	N/A	CF	PUMPXX	RC Pressure Trip (Loss of Main Feedwater Pump)

1
F: Forced
S: Scheduled

2
Reason:
A-Equipment Failure (Explain)
B-Maintenance or Test
C-Refueling
D-Regulatory Restriction
E-Operator Training & License Examination
F-Administrative
G-Operational Error (Explain)
H-Other (Explain)

3
Method:
1-Manual
2-Manual Scram.
3-Automatic Scram.
4-Other (Explain)

4
Exhibit G - Instructions for Preparation of Data Entry Sheets for Licensee Event Report (LER) File (NUREG-0161)

5
Exhibit I - Same Source

(9/77)

REFUELING INFORMATION REQUEST

1. Name of Facility: Rancho Seco Unit 1
2. Scheduled date for next refueling shutdown: April, 1982
3. Scheduled date for restart following refueling: October, 1982
4. Technical Specification change or other license amendment required:
   a) Change to Rod Index vs. Power Level Curve (TS 3.5.2)
   b) Change to Core Imbalance vs. Power Level Curve (TS 3.5.2)
   c) Tilt Limits (TS 3.5.2)
5. Scheduled date(s) for submitting proposed licensing action: February, 1982
6. Important licensing considerations associated with refueling: None
7. Number of fuel assemblies:
   a) In the core: 177
   b) In the Spent Fuel Pool: 196
8. Present licensed spent fuel capacity: 579
9. Projected date of the last refueling that can be discharged to the Spent Fuel Pool: 1987