NARRATIVE SUMMARY OF PLANT OPERATIONS

6-1 to 6-30 Continuous reactor operation at ~96%. Performed routine surveillance and preventive maintenance items.

MAJOR ITEMS OF SAFETY-RELATED MAINTENANCE

1. Replaced and tested Control Rod Drive undervoltage trip delay.

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

1. Added a key switch & annunciator in Control Room for bypassing control grade reactor trip on loss of feedwater.
2. Provided enclosure around nuclear service transformer X43A.
3. Modification of NNI.

UNIT SHUTDOWNS AND POWER REDUCTIONS

DOCKET NO. 50-312
UNIT NAME Rancho Seco #1
DATE 80-06-30
COMPLETED BY R. W. Colombo
TELEPHONE 916-452-3211

REPORT MONTH June 1980

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
NO OUTAGES OR SIGNIFICANT POWER REDUCTION (GREATER THAN 20% REDUCTION IN AVERAGE DAILY POWER LEVEL FOR THE PRECEDING 24 HOURS) THIS MONTH.									

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)

OPERATING DATA REPORT

DOCKET NO. 50-312
DATE 80-06-30
COMPLETED BY R. W. Colombo
TELEPHONE 916-452-3211

OPERATING STATUS

1. Unit Name: Rancho Seco #1
2. Reporting Period: June 1980
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	4,367	45,624
12. Number Of Hours Reactor Was Critical	720	1,514	27,587.5
13. Reactor Reserve Shutdown Hours	0	0	3,975.1
14. Hours Generator On-Line	720	1,429.6	26,370.9
15. Unit Reserve Shutdown Hours	0	0	1,210.2
16. Gross Thermal Energy Generated (MWH)	1,902,171	3,587,222	66,620,920
17. Gross Electrical Energy Generated (MWH)	638,818	1,196,626	22,481,162
18. Net Electrical Energy Generated (MWH)	607,076	1,123,943	21,234,832
19. Unit Service Factor	100	32.7	58.5
20. Unit Availability Factor	100	32.7	61.2
21. Unit Capacity Factor (Using MDC Net)	96.6	29.5	53.3
22. Unit Capacity Factor (Using DER Net)	91.8	28.0	50.7
23. Unit Forced Outage Rate	0	5.3*	31.0*

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

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

*These figures reflect a correction made to May 1980 report.

AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-312
UNIT Rancho Seco #1
DATE 80-06-30
COMPLETED BY R. W. Colombo
TELEPHONE 916-452-3211

MONTH June 1980

DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)	DAY	AVERAGE DAILY POWER LEVEL (MWe-Net)
1	850	17	867
2	860	18	871
3	863	19	869
4	862	20	872
5	861	21	873
6	872	22	871
7	875	23	871
8	873	24	870
9	870	25	869
10	863	26	873
11	872	27	869
12	872	28	868
13	871	29	870
14	871	30	869
15	868	31	
16	869		

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.

REFUELING INFORMATION REQUEST

1. Name of Facility: Rancho Seco Unit 1
2. Scheduled date for next refueling shutdown: July 1981
3. Scheduled date for restart following refueling: September 1981
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)
   d) Safety Equipment Testing (TS 3.3.3)
5. Scheduled date(s) for submitting proposed licensing action: May 1981
6. Important licensing considerations associated with refueling: None
7. Number of fuel assemblies:
   a) In the core: 177
   b) In the Spent Fuel Pool: 164
8. Present licensed spent fuel capacity: 579
9. Projected date of the last refueling that can be discharged to the Spent Fuel Pool: 1987

Cycle 4 Power Distribution Comparison

In the District's letter to Mr. Robert W. Reid, dated February 27, 1980, we committed to perform Power Distribution Comparisons through Cycle 4 as a result of this being our first reload core utilizing Lumped Burnable Poisons.

Power Distribution Analyses at the beginning of Cycle 4, and 25 EFPD, were accomplished per the techniques and criteria specified in the Power Escalation Test program. The results of that program are included elsewhere in this month's report. Additionally, B&W has performed an RMS analysis on the 25 EFPD power distribution. The value determined was 0.0248 which compares favorably with the requirement that it be less than 0.0731.

As of the end of June, approximately 43 EFPD had been accumulated, hence the data for RMS comparison specified for 50 EFPD has not been done.

SECTION I - OVERVIEW

Following the third refueling of Rancho Seco Unit #1, the startup test program for Cycle 4 was begun with initial criticality established at 0510 hours on May 9, 1980. Zero power physics testing commenced at that time and was successfully completed on May 10, 1980 at 1600 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 10, 1980 and testing was done at three major power plateaus of 40%, 75% and 96% of full power. This final plateau being attained on May 18, 1980.

As of June 30, 1980 the plant has not attained 100% of full power due to a self-imposed restriction to insure inadvertant Power/Flow trips would not occur. See sections on Reactor Coolant Flow and Flow Coastdown testing. Tests intended at full power were completed at 96% full power on June 16, 1980.

The following descriptions of test data and results refer to the Cycle 4 Reload Report, BAW-1560, August 1979 testing commitments and the District's February 19, 1980 and February 27, 1980 responses to the Commission's February 11, 1980 request for additional information and commitment. Reference is made to that information rather than repeating it here.

SECTION II - PRE-CRITICAL TEST

.1 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 droppable 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 1.176 seconds, and the slowest rod was at 1.238 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.

.2 Reactor Coolant Flow

The steady state four pump flow was determined for the hot zero power condition as being 404,820 GPM. This can be compared to the Technical Specification minimum acceptable value of 387,600 GPM. The maximum flow is established based on core lift criteria. This upper limit for

.2 Reactor Coolant Flow (Continued)

Cycle 4 is 419,600 GPM. These measurements met the Cycle 4 performance requirements. This test was performed at BOC-4 to verify performance following the installation of 52 Lumped Burnable Poison Assemblies in fuel which was unrodded during Cycle 3 testing. Correcting BOC-3 data to BOC-4 conditions shows an apparent reduction of 6356 GPM, or about 1.6%. Analysis had expected the effect on core bypass flow to be a decrease from 10.4% to 8.3% of total flow. These measurements show the effect has been properly anticipated and the results to be acceptable.

.3 Reactor Coolant Flow Coastdown

From the four pump configuration described above, the reactor coolant pump determined to be the highest flow pump was tripped, and the total flow through the reactor core determined as a function of time. The acceptance criteria was applied to the before trip conservative error-reduced value. It was determined that the actual coastdown transient flow exceeded the minimum acceptable flow for the period of interest by a margin in excess of 4000 GPM. This test met the requirements for operation of Cycle 4.

A second feature of this test was to verify that the time delay assumed in the flow coastdown safety analysis was not exceeded. This time delay is due to the use of hydraulic snubbers in the sense lines to the flow signal ΔP transmitters. A conservative one-second delay had been assumed. The test compared the rate of flow coastdown between one channel, free of snubbers, and the three remaining Reactor Protection Flow channels, whose

.3 Reactor Coolant Flow Coastdown (Continued)

snubbers had been set to provide a slightly less damped signal than previously. The comparison showed the delay to range between 0.65 and 0.73 seconds, well within the assumed interval.

As a result of this snubber position, the noise seen on the flow signals has caused intermittent drops in the conservatively set Power/Flow RPS Trip signal to the point that the trip signal could be generated at as low as 100.8% full power, down from its Technical Specification upper limit of 105.0% full power. For this reason, power has been limited to a nominal 96% full power while analysis in support of a licensing action is undertaken. Measured flow is approximately 109.5% of design flow, hence such an analysis is in order.

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 4 Reload Report.

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

Measured	Vendor Prediction
1361.85 ppmB	1368 $\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
1012 ppmB	1004 $\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 Goup 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
-0.231×10^{-4} $\Delta k/k/F°$ at 1359 ppmB	$-0.29\times10^{-4}\pm0.4\times10^{-4}$ at 1359 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 1359 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 $+0.5\times10^{-4}\ \Delta k/k/F°$. The Moderator Coefficient of Reactivity was determined to be $-0.03\times10^{-4}\ \Delta k/k/F°$.

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

Measured	Vendor Prediction
$-0.918\times10^{-4}\ \Delta k/k/F°$ at 1012 ppmB	$-0.95\times10^{-4} \pm 0.4\times10^{-4}\ \Delta k/k/F°$ at 1012 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	0.866	0.98	-13.16	±15%
Group 6	0.825	0.87	-5.45	±15%
Group 7	1.389	1.46	-5.11	±15%
Total	3.080	3.31	-7.47	[illegible]

As the measured total group worth was within ±10% of the predicted value, further actions committed to in the District's February 19, 1980 letter were not required. The shutdown margin calculations shown in the Cycle 4 Reload Report are substantiated by the above measurements and the excellent agreement between predicted and measured ARO Boron.

.7 Ejected Rod Worth Measurement

Error Adjusted Ejected Rod Worth, %Δk/k	Measured Ejected Worth, %Δk/k	Predicted Worth, %Δk/k	Tolerance Allowed
0.[illegible]	0.7805	0.76	±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. Furthermore, the worth of the three Group 7 rods symmetric with the measured ejected rod were determined by swapping them against Group 5 and using the calibrated worth of Group 5 over its interval to estimate the ejected rod worth. The non-error adjusted worths ranged from a high of 0.794%Δk/k to the minimum measured at 0.758%Δk/k. These results are certainly within the margin of tolerance for the measurement technique and provide an early confirmation of power distribution symmetry and lack of power tilt for Cycle 4. Subsequent observations during power escalation confirm the tilt free nature of this core.

SECTION IV - POWER ESCALATION

.1 Core Power Distribution

Core power distributions were taken and analyzed at the nominal Reactor power test plateaus of 40%, 75%, and 96%FP during Cycle 4 power escalation. The purpose of these measurements was to verify that the minimum DNBR, maximum linear heat 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 plateuu 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	5/13/80	5/16/80	5/19/80	6/10/80
Power level, %FP	41.0/40	74.7/75	95.5/100	95.5/100
Core Burnup, EFPD	1.0/2.0	1.92/3.0	5.0/4.0	23.6/25.0
Group 1-5, %WD	100/100	100/100	100/100	100/100
Group 6, %WD	100/100	100/100	100/100	100/100
Group 7, %WD	87.1/87.0	87.0/87.0	98/87.0	98.9/87.0
Group 8, %WD	25.9/25.0	23.0/22.0	23.0/19.0	22.0/19.0
Boron Concentration, ppmB	1021/1000	933/880	866/862	789/811
Axial Imbalance, %FP	-2.13/-0.29	-1.76/-0.07	-0.67/-2.95	-3.67/-3.07
Max Incore Quadrant Power Tilt, %FP	0.61/<3.64	0.58/<3.64	0.57/<3.64	0.48/<3.64
Minimum DNBR	8.54/≥1.30	4.26/≥1.30	3.30/≥1.30	3.30/≥1.30
Worse Case LHR, Kw/ft	4.85/≤20.4	8.73/≤20.4	11.09/≤20.4	11.09/≤20.4
Max Radial Power Peak	1.281/1.306	1.276/1.291	1.264/1.285	1.285/1.289
Max Total Power Peak	1.515/1.529	1.510/1.525	1.456/1.528	1.493/1.527
Max Peak at Core Grid	L-13/H-11	L-13/H-11	L-13/H-11	L-13/H-13
Max Peak in Fuel Batch Number	6/6	6/6	6/6	6/6
Equilibrium Xenon	Yes, 2D	Yes, 2D	Yes, 3D	Yes, 3D
Acceptable for Power Escalation	Yes	Yes	Yes	Yes
Extrapolations done to, %FP	91.5	112.0	112.0	112.0

Power Distribution Test Results (Continued)

Acceptance criteria which applies to the radial and total peaking factors is +5% and +7.5% respectively when compared to the predictions for the peak assembly at the 75% and 100% power plateaus. All acceptance criteria was 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 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 escalatio , the initial results showed the out-of-core Nuclear Instrumentation tc e very conservative. Anytime regaining is

.2 Power Imbalance Detector Correlation Test (Continued)

done, a retest is required. This regaining and retest was accomplished at 90%FP and all applicable acceptance criteria met. The results corresponding to the maximum and minimum imbalance conditions are shown here:

NI Channel	40%FP			~90%FP (Retest)		
	Difference Amplifier Gain	Target Correlation Slope	Measured Correlation Slope	Difference Amplifier Gain	Target Correlation Slope	Measured Correlation Slope
NI-5	4.13	≥1.15	1.55	3.54	≥1.15	1.266
NI-6	4.13	≥1.15	1.52	3.54	≥1.15	1.217
NI-7	4.13	≥1.15	1.55	3.54	≥1.15	1.240
NI-8	4.13	≥1.15	1.54	3.54	≥1.15	1.232
Acceptance Criteria	--	≥1.15	Met	--	≥1.15	Acceptable

Cycle 4 safety analysis assumes that the correlation slope is greater than or equal to 1.150. As the above data shows, this correlation criteria is 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.

.2 Power Imbalance Detector Correlation Test (Co.:tinued)

A gain factor of 4.13 set into the Nuclear Instru.:entation differential amplifier circui' · :'r this cycle was determined to be exessively conservative. Thus the gains were subsequently reset to 3.54 and an operational transient used to induce the imbalance changes necessary to demonstrate acceptable correlation. The results are shown above, thus Cycle 4 will operate with these values.

.3 Power Doppler Coefficient of Reactivity

From equilibrium conditions at near 96%FP, the power doppler coefficient was determined. The value obtained was -1.68×10^{-4} $\Delta k/k/\%FP$. The acceptance criteria for this parameter was that the value shall always be more negative than -0.55×10^{-4} $\Delta k/k/\%FP$. This criteria is therefore satisfied.

.4 Moderator Temperature Coefficient of Reactivity at Power

The "at power" moderator temperature coefficient was measured as described in the Cycle 4 Reload Report, while operating the Reactor at equilibrium conditions and near 96%FP. Measurements determined the coefficient to be -0.99×10^{-4} $\Delta k/k/F°$ compared to a vendor predicted value of -1.44×10^{-4} $\Delta k/k/F°$. The acceptance criteria for this parameter is that it shall not be "positive" for Reactor operations above 95%FP. This condition for operation is satisfied for Cycle 4.

SECTION V - SUMMARY

The District's letter of February 27, 1980 committed to a follow-on program of core power distribution review, analysis, and reporting due to the unique fuel management scheme (LBP in Reload Fuel) being utilized. This program involves B&W analysis each 50 EFPD to determine their ability to predict LBP behavior in a reload. Requisite reports will be included in the monthly plant performance report to the NRC.

Since this startup program was completed at 96%FP, the increment to 100%FP and the associated 100%FP Power Distribution Analysis will be reported in a monthly report.

The final test in this program was the Reactivity Coefficients at Power measurements reported above. Those tests were completed on June 16, 1980, hence this report is due submission within 45 days, or by July 31, 1980.

The results of early Cycle 4 testing provided in this report demonstrate that Rancho Seco Unit 1, Cycle 4, has been properly designed; and that the unit can be operated in a manner that will not endanger the health and safety of the public.