

APRIL 1985

SUMMARY OF PLANT OPERATIONS

The plant has been in cold shutdown for refueling and plant modifications for the entire month of April.

PERSONNEL CHANGES REQUIRING REPORT

None.

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

The following facility changes were completed in April:

1) The allowable leakage from the Decay Heat Removal and Reactor Building Spray Systems was increased from 0.63 gph to 6.0 gph. This change resulted from the 0.63 gph limit not being attainable and is consistent with the Standard Technical Specifications. The dose resulting from a maximum hypothetical accident (MHA) occurring at the 6.0 gph limit was calculated to be 7.21 rem thyroid and <0.01 rem whole body, which is still orders of magnitude less than the 10 CFR 100.11(a)1 dose limits.

This change required revisions to Technical Specification 4.5.3, FSAR Section 14.3.9.3, and surveillance procedure SP 204.07, SP 201.12, and SP 203.09.

2) A change was made to allow the pulling of Class 1 cables without the cable number being painted on the jackets in the associated protection channel color code, as described by USAR paragraph 7.1.4. The change permits the Class 1 cable jackets to be striped in the appropriate color and cable markers provided at the ends of the circuit, which is the general industrial practice. No Technical Specification revisions or unreviewed safety questions resulted from this change.

3) Four (4) out of one hundred and eight (108) lower core barrel bolts were selectively removed for testing purposes. These bolts will not be replaced because of the extreme difficulty and expense involved. An evaluation determined that operation with the remaining 104 lower core barrel bolts is an acceptable configuration in accordance with the ASME Boiler and Pressure Vessel Code. Calculations demonstrated that there was no significant increase in bolt loading from the removal of the four bolts. No Technical Specifications revision or unreviewed safety question resulted from this change.

4) Remote operating capability from the control room was added to valves DHS 001 and DHS 002 and the valves were re-designated HV-20005 and HV-20004, respectively. Operation of these valves is needed to achieve cold shutdown of the plant and, under some accident conditions, high radiation could make these valves inaccessible. The remote operating capability was accomplished by replacing the existing spur gear manual operators on the valves with Class 1E motor operators actuated from the control room. The motor operators fail "as is" on loss of power. Both motor operators will be supplied from separate

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Class 1E power sources and no single failure will cause failure of both valves. This change did not involve a change in the Technical Specifications or an unreviewed safety question.

5) Procedure SP 207.01 (Inservice Inspection) was revised to approve the use of two (2) vendor eddy current inspection procedures at Rancho Seco. The vendor procedures are ISI-410 (Eddy Current Examination of Condenser Tubing, Rev. 5) and ISI-460 (Technical Procedure for the Evaluation of Eddy Current Data of Nuclear Grade Steam Generator Tubing, Rev. 10). These procedures allow the user to take advantage of new techniques and equipment related to Eddy Current Testing.

The procedures were reviewed by a qualified Level III technician from the vendor's staff as well as a qualified Level III member of the District's staff to ensure that applicable codes and standards were met.

6) Radiation monitors (R-15056, R-15058, R-15059, R-15060, R-15061, and R-15062) which were installed on an interim basis to meet the requirements of NUREG-0578 2.1.8.b, were removed from service. The functions of these instruments were to monitor radiation levels at the reactor building vent, auxiliary building vent, main steam line A, and main steam line B following an accident. A new, permanent system has been installed to perform the functions previously performed by the interim monitors.

7) Class 1E power cables were installed between the Class 1E power panels and the multiplexers for the Interim Data Acquisition and Display System (IDADS). The IDADS is an interim system designed to implement the essential features of NUREG 0696 until the Plant Integrated Computer System (PICS) is operable. This change does not introduce any failure modes that have not been analyzed previously.

8) Cycle 6 changes defined by the B&W Cycle 6 Fuel Cycle Design Report and Cycle 6 Reload Report are:

- a) Forty of the assemblies will incorporate axial blankets (Mark BAB) in the upper and lower 6" of the assemblies. This is an extension of the Lead Test Assembly (LTA) irradiations initiated in Cycle 5.
- b) The BPRA's installed in 40 Batch 8B assemblies and 8 of the Batch 8A assemblies will incorporate asymmetric poison stacks (117"). Sixteen of the BPRA's will contain BPRA's of the original design (126").
- c) The Batch 9B assemblies will incorporate the Mark B5 upper end fittings.
- d) The Fuel Cycle will be designed with "gray" APSR's, i.e., the poison section will be replaced with Inconel to reduce the reactivity worth of the APSR's.
- e) The design length of Cycle 6 will be 345 ± 10 EFPD's.

The overall design and safety analyses are provided in the Cycle 6 reload report.

9) Two out of three logic circuits were installed to initiate the nuclear service bus unloading and loading schemes on bus 4A and 4B on overvoltage/undervoltage conditions. This change was made to meet the NRC staff "Position 1: Second Level of Under or Overprotection With a Time Delay" from Enclosure 1 to the NRC's June 3, 1977 letter to the District (Reid to Mattimoe).

10) The auxiliary feed pump turbine casing drain configuration was modified, per vendor recommendation, to ensure proper, efficient operation. The high and low pressure stage drains were headered together to a common steam trap. The low pressure drain should be open to atmosphere or exhaust with no devices creating backpressure located downstream. Proper low pressure casing drainage will prevent excessive condensate buildup and possible turbine wheel tip corrosion. The modification, in effect, ensured that the low pressure drain will be open to atmosphere.

11) The Axial Power Shaping Rods (APSR) were replaced with newly designed APSR's for Fuel Cycle 6. The redesigned APSR's improved resistance to creep and core power distribution characteristics. The previous design was found to be subject to irradiation induced creep mechanisms which could lead to failure or collapse of the APSR cladding. The improved power distribution characteristics were accomplished by replacing the former Ag-Cd-In poison section with Inconel. The change in operating characteristics was incorporated into operator training prior to and during Fuel Cycle 6.

12) An overflow line from the Miscellaneous Water Hold Up Tank connecting to the overflow line from the Borated Water Storage Tank to the Reactor Coolant System Drain Tank was added to prevent unintentional releases of contaminated water to offsite. The overflow line piping is in accordance with piping specification HD and ANSI B31.1 Power Piping.

13) Channelized power was provided to the Non-Nuclear Instrumentation (NNI) signal conversion cabinets so that loss of a single power supply will not cause loss of critical signals to NNI X and Y channels. This change was accomplished by providing AC power for the non-IE portion of the H4SCA Signal Conversion cabinet from a source separate from the source supplying AC power to the non-IE portion of H4SCB. Both AC sources are battery backed, but are not QA Class 1.

14) Plant egress was improved by modifications to the Personnel Access Portal (PAP) Building Area. These changes included the relocation of the existing double door exit and modification of the badge-return window chute. Also, the personnel orientation room was enlarged to allow larger groups to be trained and badged during plant outage preparation. The Security Plan was revised to reflect the changes. The changes will not reduce the effectiveness of the Security Plan.

MAJOR ITEMS OF SAFETY RELATED MAINTENANCE

1) The Diesel Generator Air Start Compressor Discharge Line check valve (EGS-021) was found to be leaking and thereby keeping the compressor loaded. The valve was disassembled and the disc was damaged beyond repair. The disc, nut pin, and gasket were replaced and the valve put back into service.

2) Primary coolant was detected seeping from temperature element TE-21034. Verification that the leakage was not coming from the welded area was made. Replacement of the flexitallic gasket stopped the leakage.

3) The RB Sump Isolation Valve (SFV-66309) was found to be leaking approximately 88.0 to 108.6 sccm past its seat. The valve was disassembled, new Teflon seat and gasket installed, and reassembled and stroked to verify proper operation.

4) The three (3) Ton Bridge Crane motor froze in place when the hoist was overloaded during a surveillance procedure. The rotor ring was turned down and the brushes were replaced to correct the problem. Following this repair, the crane was successfully functionally tested.

REFUELING INFORMATION REQUEST

1. Name of Facility Rancho Seco Unit 1
2. Scheduled date for next refueling shutdown: Sept 15, 1986
3. Scheduled date for restart following refueling: Jan 15, 1987
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: April 9, 1986
6. Important licensing considerations associated with refueling: N/A
7. Number of fuel assemblies:
 - a) In the core: 177
 - b) In the Spent Fuel Pool: 316
8. Present licensed spent fuel capacity: 1080
9. Projected date of the last refueling that can be discharged to the Spent Fuel Pool: Dec 3rd, 2001

AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-312

UNIT Rancho Seco Unit 1

DATE 04-30-85

COMPLETED BY R. Colombo

TELEPHONE (916) 452-3211

MONTH April 1985

DAY AVERAGE DAILY POWER LEVEL
(MWe-Net)

1	<u>0</u>
2	<u>0</u>
3	<u>0</u>
4	<u>0</u>
5	<u>0</u>
6	<u>0</u>
7	<u>0</u>
8	<u>0</u>
9	<u>0</u>
10	<u>0</u>
11	<u>0</u>
12	<u>0</u>
13	<u>0</u>
14	<u>0</u>
15	<u>0</u>
16	<u>0</u>

DAY AVERAGE DAILY POWER LEVEL
(MWe-Net)

17	<u>0</u>
18	<u>0</u>
19	<u>0</u>
20	<u>0</u>
21	<u>0</u>
22	<u>0</u>
23	<u>0</u>
24	<u>0</u>
25	<u>0</u>
26	<u>0</u>
27	<u>0</u>
28	<u>0</u>
29	<u>0</u>
30	<u>0</u>
31	<u>0</u>

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.

OPERATING DATA REPORT

DOCKET NO. 50-312

DATE 04/30/85

COMPLETED BY R. Colombo

TELEPHONE (916) 452-3211

OPERATING STATUS

NOTE:

1. Unit Name: Rancho Seco Unit 1
2. Reporting Period: April 1985
3. Licensed Thermal Power (MWt): 2,772
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
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	719	2,879	87,984
12. Number of Hours Reactor Was Critical	0	1,624.5	53,071.9
13. Reactor Reserve Shutdown Hours	0	0*	10,189.9*
14. Hours Generator On-Line	0	1,618.2	49,281.7
15. Unit Reserve Shutdown Hours	0	0	1,210.2
16. Gross Thermal Energy Generated (MWH)	0	4,055,333	125,665,601
17. Gross Electrical Energy Generated (MWH)	0	1,366,846	40,798,809
18. Net Electrical Energy Generated (MWH)	0	1,289,938	37,881,184
19. Unit Service Factor	0.0%	56.21% *	56.01%
20. Unit Availability Factor	0.0%	56.21%	57.39%
21. Unit Capacity Factor (Using MDC Net)	0.0%	51.33%	49.32%
22. Unit Capacity Factor (Using DER Net)	0.0%	48.81%	46.90%
23. Unit Forced Outage Rate	0.0%	8.8%	29.4%
24. Shutdowns Scheduled Over Next 6 Months (Type, Date, and Duration of Each):			
Refueling - March 15, 1985 - June 15, 1985 Three Months			

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 | <u>N/A</u> | <u>N/A</u> |
| INITIAL ELECTRICITY | <u>N/A</u> | <u>N/A</u> |
| COMMERCIAL OPERATION | <u>N/A</u> | <u>N/A</u> |

* Corrected from March 1985 report.

UNIT SHUTDOWNS AND POWER REDUCTIONS

REPORT MONTH APRIL 1985

DOCKET NO. 50-312
 UNIT NAME RANCHO SECO UNIT 1
 DATE 4/30/85
 COMPLETED BY R. Colombo
 TELEPHONE (916)452-3211

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
5	4/1/85	S	719	C	1	N/A	ZZ	ZZZZZZ	Shutdown for Refueling

¹
 F: Forced
 S: Scheduled

(9/77)

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



SMUD

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AN ELECTRIC SYSTEM SERVING THE HEART OF CALIFORNIA

RJR 85-243

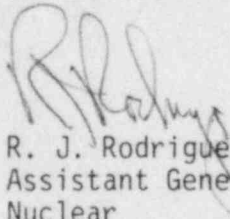
May 8, 1985

DIRECTOR
OFFICE OF INSPECTION AND ENFORCEMENT
U S NUCLEAR REGULATORY COMMISSION
WASHINGTON DC 20555

OPERATING PLANT STATUS REPORT
DOCKET NO. 50-312

Enclosed is the April 1985 Monthly Plant Status Report for Rancho Seco

Unit One.



R. J. Rodriguez
Assistant General Manager,
Nuclear

cc: I&E Washington (9)
Region V
MIPC (2)
INPO

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