

SEPTEMBER 1985

SUMMARY OF PLANT OPERATIONS

The plant was in cold shutdown for plant modifications through September 27, 1985. The reactor achieved criticality at 2237 hours on September 28, 1985, and the plant was brought on line at 1855 hours on September 29, 1985. Reactor power was leveled at approximately 40% on September 30, 1985, for physics testing.

PERSONNEL CHANGES REQUIRING REPORT

Steve Redeker, Supervisor of the Shift Technical Advisors (STA) Group, was assigned the position of acting Plant Superintendent. The qualifications of the individual in this position will be reported when the position is officially filled.

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

The documentation for the following facility changes was completed in September. All of the changes have been subjected to the review and approval of the Plant Review Committee (PRC) and the Management Safety Review Committee (MSRC).

1. The Anticipatory Reactor Trip System (ARTS) non-qualified pressure switches were replaced with qualified/safety grade pressure switches to meet the requirements of NUREG 0737, Item II.K.2.10. The switches replaced were: four (4) switches monitoring the main turbine auto stop oil pressure; four (4) switches monitoring the main feedwater pump A governor oil pressure; and four (4) switches monitoring the main feedwater pump B governor oil pressure.
2. A facility change was completed to provide a structure, attached to the reactor closure head service structure, for storing the reactor cavity seal ring. The structure was designed to meet design loads under seismic and pipe-break conditions and to permit movement of the stud tensioners and in-service inspection equipment with the existing chain hoists.
3. A new, qualified solenoid valve was installed in the Makeup and Purification System in parallel to SFV-24013. The purpose of the new valve installation is to provide additional air dump capacity to ensure closure of the reactor coolant pump seal return valve within the required Technical Specification Table 3.6.1 time.
4. The normally open solenoid valves for pressurizer relief tank nitrogen supply isolation valve SFV-92520 were replaced with normally closed solenoid valves. These replacement valves remain de-energized during normal plant operation except when adding nitrogen to the pressurizer relief tank. During an SFAS condition, the new solenoid valves are deenergized closed. Replacement of these valves eliminates the requirement for continuous energization of the solenoid valves coils during normal plant operation and post-DBE, thus preventing overheating of the coils.
5. The letdown orifice manual bypass valve (PLS-036), piping, and valves in the area of the letdown orifice were replaced to restore the letdown system to its original configuration. This equipment had previously been removed under ECN A-2913 because of erosion to the valve body. Restoration of the letdown system to the original configuration eliminates crud traps, thereby reducing radiation levels, and provides increased operational flexibility during maintenance of the letdown flow control valve (HV-22011).

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6. A set of design drawings was developed to provide a generic design basis and installation details for fabricating and installing tubing system and transmitter supports for compliance with NUREG 0737, R.G. 1.97, and IE Bulletin 79.01B. These drawings are included in a generic design package for Class I equipment. Plant changes related to the listed documents will be specified under individual ECN's with reference to the generic design package.
7. Surveillance Procedure SP 211.01C, Control Room/Technical Support Center Emergency Filtering System Refueling Interval Surveillance Test, was revised to provide detailed instructions for performing surveillance to satisfy Technical Specifications 4.10.1 requirements. The procedure change was necessitated by changes to the Control Room Emergency Ventilation System performed under ECN A-3290, Mod 32.
8. A transfer switch, to provide an automatic transfer of 120VAC to the S1J bus from the S1F bus in the event of failure of the S1J inverter, was installed. The ASCo switch was installed inside the S1F enclosure and will improve the reliability of the 120VAC non-vital power distribution system.
9. Auxiliary feedwater flow orifice plates FE-31802 (OTSG A) and FE-31903 (OTSG B) on lines 31821-6"-DB2 and 31823-6"-DB2 respectively, were replaced with QA Class 1 orifice plates. This change provides safety grade indication of AFW flow to each steam generator in accordance with NUREG 0737/II.E.1.2.2.
10. As a result of a review of the applications of Bergen-Paterson EA-3 pipe clamps (IE Information Notice 83-13), the following pipe clamps were replaced:

<u>HANGER</u>	<u>SYSTEM</u>
1RS-50050-13	NSW
4RS-26021-9A	DHS
1SW-21582-2	RCS

The replacements were made under ECN A-5430, Rev.1.

11. The "B" reactor coolant pump seal unit was upgraded to a Eingham-Willamette Company (B-W-C) Model RQV-875B-3 seal. This seal differs from the replaced unit in the basic design of the stationary components, the method of stage-to-stage pressure reduction, and the installation/removal details and represents the B-W-C state-of-the-art balanced stator seal. The new seal is: (1) expected to provide a longer service life; (2) more tolerant of transient conditions, shaft tilt, misalignment, and axial movement; and (3) installed/removed as a single unit, thus minimizing handling time and radiation exposure to service crews.
12. The size of liquid waste return line 70058-3/8"-HD was increased from 3/8" to 5/8" to reduce sampling time and friction losses, and to increase system reliability. The 3/8" line was too small to accommodate a sampling flow of 2 gallons per minute.

13. Four (4) Class 1E differential pressure transmitters (2 per channel) were added for hot leg level indication in the main control room on SPDS and IDADS. This instrumentation was required to comply with NUREG 0737/II.F.2 and NRC letter dated December 10, 1982, Docket 50-312, regarding Inadequate Core Cooling Instrumentation System.
14. Segments of steam generator tubes were removed from rows near the open lane for subsequent examination, analysis and testing of defects found during eddy current testing. Removal of these tubes from service has no measurable effect on thermal/hydraulic performance of the steam generators and no measurable reduction of reactor coolant system flow will result. This work satisfies a District commitment made in a November 20, 1984, NRC meeting.
15. The Control Room space arrangement of the conference room area was changed by removing two existing walls and a door, and adding a new wall. Figure 7.4-1 of the USAR was revised to reflect the change. The change was designed to provide more space for Control Room personnel and to reduce traffic in the operating area of the Control Room.
16. Additional printed circuit boards were installed in the IDADS multiplexer to provide additional input capability. The boards were purchased under SMUD Specification M19.113 to ensure compatibility with existing equipment.
17. Surveillance Procedure SP203.07 was revised to include measurement of the pressure drop across the NSCW heat exchangers. The pressure drop will be monitored to determine if plugged tubes exist and when corrective action is required.
18. An analysis was performed to address the potential impact on safety resulting from disabling supply breakers of certain plant valves in the open position. The valves addressed were associated with the Decay Heat Removal System and the breakers were being postulated disabled to prevent potential fire-induced spurious operations (10 CFR 50, Appendix R).
19. The Once Through Steam Generators (OTSG A & B) outlet pressure transmitters PT-20519A, PT-20519B, PT-20520A, and PT-20520B were replaced with environmentally qualified pressure transmitters. This replacement was made in order to comply with the requirements of 10 CFR 50.49(b)(2), i.e., non-safety related electric equipment whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions.
20. The following transmitters and pressure switches were replaced with qualified transmitters and pressure switches:

PSH-53606	(RPS, Containment Building)
PSH-53617	" "
PSH-53619	" "
PSH-53620	" "
PT-53606B	(SFAS, Containment Building)
PT-53607	" "
PT-53608	" "

PT-21042	(SFAS, Reactor Coolant System)
PT-21043	" "
PT-21092	" "
PT-21099	(Decay Heat Valve Interlock, Reactor Coolant System)
FT-26048	(Indication, Decay Heat System)
FT-26049	" "

These replacements were made to comply with IE Bulletin 79-01 B.

MAJOR ITEMS OF SAFETY RELATED MAINTENANCE

1. Auxiliary feedwater pump P-318 was disassembled because it would not pass the flow capacity test of surveillance procedure SP 210.01A. New rotating elements were installed, all internal components were refitted, and the pump reassembled. Thereafter, the pump successfully completed SP 210.01A and special test procedure STP-177. STP-177 contained a 48 hour endurance run and assured compliance with NRC IE Bulletin 83-05, ASME Nuclear Code Pumps and Spare Pumps manufactured by the Hayward Tyler Pump Company.
2. The RCS "B" loop SFAS transmitter (PT-21099) output voltage was spiking, resulting in the automatic closure of decay heat removal system block valve HV-20002 (reported in LER 85-16). PT-21099 primary circuit board and calibration board were replaced, along with the input current buffer card in the corresponding SFAS cabinet. The channel was then calibrated in accordance with I.203 (Decay Heat Valve Interlock Channel B Calibration).

REFUELING INFORMATION REQUEST

1. Name of Facility Rancho Seco Unit 1
2. Scheduled date for next refueling shutdown: Jan 1, 1987
3. Scheduled date for restart following refueling: April 1, 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: Aug 1, 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: December 3, 2001

AVERAGE DAILY UNIT POWER LEVEL

DOCKET NO. 50-312

UNIT Rancho Seco Unit 1

DATE 09-30-85

COMPLETED BY R. Colombo

TELEPHONE (916) 452-3211

MONTH SEPTEMBER 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>31</u>
30	<u>328</u>
31	<u></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 09/30/85
 COMPLETED BY R. Colombo
 TELEPHONE (916) 452-3211

OPERATING STATUS

NOTE:

1. Unit Name: Rancho Seco Unit 1
2. Reporting Period: September 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	720	6,551	91,656
12. Number of Hours Reactor Was Critical	49.4	1,767.9	53,215.3
13. Reactor Reserve Shutdown Hours	0	110.5	10,300.4
14. Hours Generator On-Line	29.1	1,647.3	49,310.8
15. Unit Reserve Shutdown Hours	0	0	1,210.2
16. Gross Thermal Energy Generated (MWH)	31,888	4,098,861	125,709,129
17. Gross Electrical Energy Generated (MWH)	8,615	1,375,462	40,807,425
18. Net Electrical Energy Generated (MWH)	0	1,289,988	37,881,184
19. Unit Service Factor	4.04%	25.15%	53.8%
20. Unit Availability Factor	4.04%	25.15%	55.12%
21. Unit Capacity Factor (Using MDC Net)	0.0%	22.56%	47.34%
22. Unit Capacity Factor (Using DER Net)	0.0%	21.45%	45.02%
23. Unit Forced Outage Rate	0.0%	8.7%	29.38%
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	<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>

UNIT SHUTDOWNS AND POWER REDUCTIONS

REPORT MONTH SEPTEMBER 1985

DOCKET NO. 50-312
 UNIT NAME Rancho Seco Unit 1
 DATE 9/30/85
 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	9/29/85	S	690.9	C	1	N/A	ZZ	ZZZZZZ	Refueling Outage

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



SMUD

SACRAMENTO MUNICIPAL UTILITY DISTRICT ☐ 6201 S Street, P.O. Box 15830, Sacramento, CA 95813; (916) 452-3211
AN ELECTRIC SYSTEM SERVING THE HEART OF CALIFORNIA

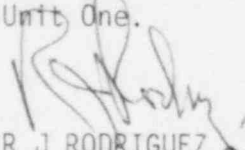
RJR 85-508

October 18, 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 September 1985 Monthly Plant Status Report for Rancho Seco
Unit One.


R J RODRIGUEZ
ASSISTANT GENERAL MANAGER,
NUCLEAR

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