REVISED

SUPPLEMENT NO. 2

TO THE

SAFETY EVALUATION

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

U.S. NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

SACRAMENTO MUNICIPAL UTILITY DISTRICT

RANCHO SECO NUCLEAR GENERATING STATION

UNIT NO. 1

DOCKET NO. 50-312

Date: MAR. 5 1978

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1.0 INTRODUCTION

The operating license for Rancho Seco Unit 1 was issued on August 16, 1974, for operation at 2772 MWt (100% full power). However, the Technical Specifications attached to the operating license temporarily limit core power to levels not exceeding 2568 MWt pending confirmation of anticipated operating performance of the boron feed and bleed system and review by the Advisory Committee on Reactor Safeguards (ACRS).

By letter dated March 12, 1975[1], the licensee, Sacramento Municipal Utility District (SMUD), proposed changes to the Technical Specifications to permit operation at the design power level of 2772 MWt. In support of the proposal, SMUD submitted a Startup Report[2], a plant operating performance report which evaluates the boron feed and bleed operation[3], and reports of operating history[4]. The power escalation program has been completed, and Rancho Seco as of May 31, 1975, had accumulated 1413 hours operation at 2568 MWt. The choice of 2568 MWt as an interim limit for Rancho Seco operation took into consideration the 2568 MWt design power level of the previously licensed Babcock & Wilcox (B&W) prototype plant, Oconee Unit 1. The reactor core for Rancho Seco is substantially the same as that of Oconee Unit 1 (Docket No. 50-269). Both cores contain 177 fuel assemblies with 208 fuel rods per assembly, however, the design heat output of the Rancho Seco core is 2772 MWt which is 8% higher than the design output of the Oconee core (2568 MWt). This 8.0% power increase is accomplished by using a 5.0% larger reactor coolant mass flow rate for Rancho Seco while permitting a 6°F larger reactor coolant temperature rise across the core.

The ACRS issued a report on its review of the Rancho Seco operating license application on September 11, 1973[5], and on November 28, 1973, the staff issued Safety Evaluation Supplement No. 1[6] which addressed areas of concern identified in the ACRS report. This safety evaluation (Supplement No. 2) updates the information contained in the November 28, 1973 Safety Evaluation and presents the staff safety evaluation of the proposed Technical Specification changes permitting 100% full power operation.

Appendix A contains a Bibliography. Appendix B contains a report from the Office of Inspection and Enforcement (OI&E) summarizing inspection results pertinent to the proposed power increase.

2.0 OCONEE UNIT 1 OPERATING EXPERIENCE

We have reviewed the operation of Oconee Unit 1 during its first fuel cycle[7,8,9,10,11&12]. Oconee Unit 1 achieved initial criticality April 19, 1973; reached full power on November 9, 1973; and after 310 equivalent full power days operation on the Cycle 1 core loading shut down for refueling on October 18, 1974. Refueling was completed during December 1974.

Oconee was a first-of-a-kind system for B&W and as such was subjected to an extensive startup test program[12]. In addition to the usual startup tests, load-following transient tasks were performed at 75% and 95% full power (FP) and azimuthal and diagonal (combination of azimuthal and axial) xenon transients were initiated and followed. Oconee performed well during these tests, meeting all acceptance criteria. The deficiencies encountered were of a nature to be expected during the startup of a complex unit and it was concluded that Oconee could be safely operated at full power.

Duke Power Company (the licensee for Oconee) has submitted reports[10&11] on comparisons between measured and calculated assembly power at several times during the initial fuel cycle. Particular attention was given to control rod interchanges. These comparisons show that, for assemblies having powers within 5% of the peak power, the largest difference between calculation and measurement was 4.6% (at BOL) and the average difference for all the measurements was less than 4%. Rod interchanges had only a small effect (<1%) on the comparison.

Review of Oconee operating history for the period July 1973 through December 1974 revealed no abnormal events that would preclude operating Rancho Seco at 2772 MWt. Several modifications to operating procedures and equipment related to reactor systems were instituted during this period. These modifications, which relate to the availability of ECCS equipment, have been reflected in the operating procedures for Rancho Seco.

On the basis of our review, we find the operation of Oconee Unit 1 during its initial fuel cycle to be satisfactory.

3.0 RANCHO SECO OPERATING EXPERIENCE

3.1 Testing and Operation

The operating license for the Rancho Seco Nuclear Station was granted to SMUD on August 16, 19 4. We reviewed the Rancho Seco Startup Report[2] which summarized significant activities from the date of obtaining the operating license to the end of the power escalation testing at 92.6% FP.

The first fuel assembly was inserted into the core on August 19, 1974, and initial fuel loading was completed on August 23, 1974. On September 16, 1974, Rancho Seco successfully achieved criticality.

Zero Power Physics testing which commenced on September 16, 1974, was successfully completed October 3, 1974. This program was conducted primarily at reactor coolant temperatures of 300°F and 532°F.

Power escalation was begun on October 13, 1974, and further power level escalations occurred as required testing was satisfactorily completed. Major power plateaus as defined by the power escalation test program, were initially achieved as follows:

Power Level (%FP)	Date Completed
15	October 13, 1974
40	December 2, 1974
75	January 9, 1975
92.6	January 24, 1975

As of May 31, 1975, Rancho Seco had been operated for more than 2113 hours at power levels greater than 75% FP and 1413 hours at 92% FP.

Testing and demonstration of the Engineered Safety Features for Rancho Seco have revealed normal operation throughout the startup phase of operation, ascension to power, and operation at 2568 MWt. The Engineered Safety Features for Rancho Seco Unit 1 consist of the reactor building and its associated ventilation and isolation systems, the emergency core cooling system, the containment cooling system, the containment spray system, and the emergency feedwater system.

The experience gained by operating personnel and the satisfactory equipment performance during the period of almost a year since the operating license was issued contribute to the safety of the proposed operation at 2772 MWt.

Acceptance criteria were established in advance of testing which allowed for anticipated calculational and measurement uncertainties. The startup test program demonstrated that rod group worths, stuck rod worths, ejected rod worths (after change in rod insertion limits), and reactivity coefficients were well within acceptance criteria and met technical specification limits. Power distribution measurements were performed at 15%, 40%, 75%, and 92.6% FP. Total peaking factors were well within the acceptance criterion. The fuel assembly to average fuel assembly power ratio was outside the criterion for one assembly at the three highest powers. However, further evaluation revealed this condition to be acceptable, since the total peaking factor in this assembly was well within the acceptance criterion and extrapolated DNBR and kW/ft values showed adequate safety margins. It was concluded that continued operation was safe.

Minimum DNBR values and maximum linear heat rate values at each power level were adjusted to account for uncertainties and conservatisms (yielding "worst case" values) and these were extrapolated to the next highest power plateau in order to ensure safe margins at the next level. "Worst case" values at the 92.6% FP level, extrapolated to 105.5% FP, showed a 60.8% margin for DNBR and 12.6% margin for linear heat rate.

3.2 Boron Feed and Bleed System

B&W has provided Rancho Seco with a boron feed and bleed system of greater control capability than for any B&W plant of earlier design. Pursuant to the ACRS recommendation to review Rancho Seco operation, we requested a special performance report for the feed and bleed system.

We have reviewed the performance report[3] submitted by the licensee describing the boron feed and bleed system test program and operating experience

Data were gathered during the operation of the feed and bleed system in order to:

1. permit verification of the ability of the nuclear steam supply system to perform power transients with feed and bleed operations, including the design transient;

2. determine the accuracy of the feed and bleed maneuvers; and

3. permit evaluation of the proficiency of operating personnel to perform feed and bleed operations.

All of the operating shifts successfully demonstrated control of the plant during power ramps. The Integrated Control System was utilized in its various modes (Integrated, Turbine Follcwing, and/or Reactor Following). Both "Feed and Bleed" and "Batch" deboration techniques were employed. A "Pseudo Design Transient" (80% FP to 30% FP and return to 80% FP after 8 hours) was successfully performed*. This transient produced a more severe xenon effect than predicted for the actual design transient (100% FP - 50% FP - 100% FP) since the power change was actually 80-30-80; i.e., greater than 50% of the initial power level. The results showed that extrapolated (to 100% FP) DNBR and linear heat generation rate margins were adequate so it will not be necessary to run the actual design transient.

The accuracy of the bleed and feed operations (as measured by the comparison between the targeted final boron concentration and the actual concentration) was within the accuracy of the boron concentration measurement.

Based on our review, we have concluded that satisfactory operation of the feed and bleed system has been sufficiently demonstrated to permit plant opeation at 2772 MWt.

3.3 Operational Occurrences

We have reviewed abnormal occurrences, unusual events, and test results for the Rancho Seco startup and operation through May 1975[13]. No events have occurred that would preclude operation at 2772 MWt. However, certain findings of interest that were encountered during the startup test program are discussed below.

3.3.1 Ejected Rod Worth

The ejected rod worth at zero power was measured to be 1.24% $\Delta k/k$ compared to the predicted value of 0.9% $\Delta k/k$ and the Technical Specification limit of 1.0% $\Delta k/k$ (for operation except for low power physics testing). Following the measurement, additional calculations were performed (by B&W) and a new correlation was developed between ejected rod worth and inserted rod worth using all available calculations and measurements. The new correlation was used to obtain a value for the maximum permissible rod insertion at zero power and insertion limits were altered accordingly after our review and approval.

*The design transient could not be performed since 100% FP operation was prohibited by Technical Specifications.

3.3.2 Reactor Vessel Noise

An unexpected sound was recorded on the loose parts monitor during non-nuclear single pump operation. After analysis, it was concluded that the sounds came from motion of the core support structure against the core support lugs and that motion of the core support structure within its design envelope could produce the sounds. The sounds occurred only during operation of the "A" and "C" pumps singly in the operating temperature range. It should be noted that single pump operation is not a mode of operation permitted by Technical Specifications when the reactor is critical. The licensee and his consultant have initiated a program using excore detectors in addition to the loose parts monitoring system to gain additional information over an extended period of time.

4.0 REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The ACRS has issued a report on its review of the Rancho Seco operating license application[5] and the staff has considered the comments and recommendations contained in that report. The steps which the staff has taken or will take relative to these comments and recommendations are described in the following paragraphs.

The Committee recommended that three conditions be satisfied before this plant be allowed to operate at full power.

a. The operation of Unit 1 of the Oconee Nuclear Station should be reviewed and found satisfactory by the Regulatory staff.

b. Following an appropriate period of operation at power levels up to 2568 MWt, the operating experience of Rancho Seco Unit 1 should be reviewed by the Regulatory staff and the ACRS.

c. Prior to the review in b above, the Regulatory staff should perform and report on an independent confirmation of the licensee's linear heat generation rates, operating limits and ECCS efficacy.

The staff has completed its review of the Oconee power operation and the Rancho Seco operation at 2568 MWt. Oconee Unit 1 operation is discussed in Section 2.0 of this report and Rancho Seco operation is discussed in Section 3.0 of this report. With respect to c above and ECCS efficacy, the licensee has presented a new LOCA analysis in response to 10 CFR 50.46.

The licensee's evaluation of the ECCS cooling performance submitted in August 1974 was based on the model developed by B&W. The staff concluded that the evaluation model required certain modifications to conform to 10 CFR Part 50, Appendix K. As a result, the Commission issued an order on December 27, 1974, limiting the linear heat generation rate based on the requirements of 10 CFR 50.46 (FAC). This order further states that the plant should be operated in accordance with limits based on the IAC and FAC until a LOCA analysis with an approved model is submitted. On July 8, 1975, the licensee submitted this analysis which is presently being reviewed.

With respect to independent confirmation of linear heat generation rates and operating limits, the NRC staff through its consultant, BNL, has performed an independent calculation of BOL power distribution for the Rancho Seco reactor. Calculations were done with the PDQ-7, Version 2 code on the CDC-7600 computer. This version of PDQ-7 incorporates thermal-hydraulic feedback effects and has an automatic criticality search on boron concentrations. The standard BNL cross-section data set was processed by the HAMMER code (for cell homogenization) and TWOTRAN code (for assembly homogenization) to produce four group cross-section sets for the PDQ-7 calculations.

The results[14] were compared to those obtained by B&W as reported in BAW-1393[15] and to additional data obtained from B&W. Steady state power distributions were obtained for BOL full power conditions and the design power transient (100% FP - 50% FP - 100% FP) at BOL was calculated. The BNL values of maximum total peaking factors agreed closely (<3.5%) with those calculated by B&W and were generally lower. Radial peaking factors ($F^N_{\Delta H}$) agreed within 4.5% of values calculated by B&W.

On the basis of these audit calculations, we find the heat generation rates calculated for Rancho Seco to be acceptable.

We have reviewed the methods used by B&W to determine operating limits. Topical Reports[15&16] and informal discussions with B&W were used in the review. The major calculational tool is the PDQ-7 code with thermal-hydraulic feedback. A large number of calculations are performed with various regulating rod configurations and a complete range of axial power shaping rod positions. Xenon transients produced by the design power transient were calculated at BOL, near EOL, and at one or more additional times during the cycle. The calculated power peaks are used, after correction for uncertainties and conservatisms, to establish operating limits. Operating limits for a particular parameter (e.g., axial imbalance) are established under the assumption that all other parameters have their most adverse permissible values.

On the basis of our review, we find the methods used to obtain operating limits to be acceptable.

Other concerns expressed by the ACRS[5] have been addressed in our safety evaluation of November 8, 1973[6]. The following information updates the November 8, 1973 report.

Fuel Loading Procedures

Detailed fuel loading procedures were developed by the licensee which provided for obtaining a permanent record of the installed location of every fuel assembly. These procedures and records

were reviewed by the staff. The performance of independent fuel assembly identification with respect to core loading location was required by the procedures and was confirmed by OI&E personnel monitoring the initial fuel loading.

Common Mode Failure and Anticipated Transients Without Scram

The staff technical report, "Anticipated Transients Without Scram for Water Cooled Power Reactors", and a request that the licensee identify the course of action to resolve ATWS was sent to the licensee on October 19, 1973[17]. The licensee responded to the request by letters dated September 30, 1974, and December 30, 1974[18], referencing B&W topical reports BAW-10016 and BAW-10099. The staff review of these reports and application of them to individual plants is in progress.

5.0 TECHNICAL SPECIFICATION CHANGES

We have reviewed the changes in Technical Specifications proposed by the licensee[1&19] to permit operation of Rancho Seco at the rated power of 2772 MWt. These include the rod withdrawal limits proposed by the licensee which represent extensions of the present limits to full power operation, and the core imbalance curves which have been modified to permit 100% FP operation and take into account the fact that core exposure is now greater than 100 equivalent full power days.

We have also added a Technical Specification change proposed by the licensee in his letter of April 7, 1974[19], requiring regulating rod positioning prior to deboration. This change adds a restriction to the Technical Specifications which had previously been imposed administratively by the licensee as a result of the ejected rod worth measurements described in Section 3.3.1 of this report.

In addition we have added a Technical Specification change requested by the licensee by letter dated May 30, 1975, which replaces Figure 2.3-2 to provide more restrictive flux /imbalance/ flow limiting safety system settings than exist in the present Technical Specifications. This change is required to correct an error which was discovered in the calculations on which the present settings are based.

The Technical Specification changes proposed[1&19], as discussed above, address the changes needed for full power operation of Rancho Seco at the current core exposure, and also impose additional restrictions on rod positioning and flux /imbalance/ flow limiting safety system settings.

6.0 CONCLUSIONS

We have reviewed the performance of Oconee Unit 1 B&W prototype plant operating at 2568 MWt and have found its operation to be satisfactory. We have reviewed the operation of Rancho Seco Unit 1, including the startup tests, the boron feed and bleed system performance, and initial operating performance up to and including 1413 hours operation at 2568 MWt. We have found the operation of Rancho Seco to be satisfactory. In our review we have found no reason to preclude operation at the proposed power level of 2772 MWt. The Advisory Committee on Reactor Safeguards reviewed the proposed operation of Rancho Seco at 2772 MWt on July 10, 1975, and by letter dated July 16, 1975, concluded that there is reasonable assurance that Rancho Seco can be operated at 2772 MWt without undue risk to the health and safety of the public.

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level not previously accepted and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental statement, negative declaration, or environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

We have concluded, based on the considerations discussed above, that: (1) because the change does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the change does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safey of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

DATE:

APPENDIX A

BIBLIOGRAPHY

1. Letter SMUD to NRC re Proposed Technical Specification Change No. 1 dated March 12, 1975.

2. Sacramento Municipal Utility District, Rancho Seco Nuclear Generating Station, Unit 1 Startup Report, March 1975.

3. Sacramento Municipal Utility District, Rancho Seco Nuclear Generating Station, Unit 1 Performance Report, March 1975.

4. Letter SMUD to NRC re January-March 1975 Operating Reports dated May 8, 1975.

5. Letter H. G. Mangelsdorf to D. L. Ray re "Report on Rancho Seco Nuclear Generating Station, Unit 1" dated September 11, 1973.

6. Supplement No. 1 to the Safety Evaluation by the Directorate of Licensing, USAEC, Rancho Seco Nuclear Generating Station Unit No. 1, November 28, 1973.

7. Duke Power Company, Oconee Nuclear Station, Semiannual Report, Period Ending December 30, 1973.

8. Duke Power Company, Oconee Nuclear Station, Semiannual Report, Period Ending June 30, 1974.

9. Duke Power Company, Oconee Nuclear Station, Semiannual Report, Period Ending December 31, 1974.

10. Letter Thies to Giambusso re Power Distribution Comparison Status Report dated December 21, 1973.

11. Letter Thies to Giambusso re Power Distribution Comparison Status Report dated July 19, 1974.

12. Duke Power Company, Oconee Nuclear Station Unit 1, Startup Report, November 16, 1973.

13. Sacramento Municipal Utility District, Rancho Seco Nuclear Generating Station, Unit 1 Annual Report, March 1975.

14. D. Diamond, Audit of B&W Power Peaking Calculations for Rancho Seco Unit 1, BNL memo (to be published).

15. Rancho Seco Unit 1 Fuel Densification Report, BAW-1393, June 1973.

16. Operational Parameters for B&W Rodded Plants, BAW-10078, September 1973.

17. Letter AEC to SMUD re Request for ATWS Information dated October 19, 1973.

18. Letters SMUD to NRC re ATWS Analysis for Rancho Seco dated September 30, 1974, and December 30, 1974.

19. Letter SMUD to NRC re Addendum 1 to Proposed Amendment No. 29 dated April 7, 1975.

APPENDIX B

OFFICE OF INSPECTION AND ENFORCEMENT
INSPECTION SUMMARY

SACRAMENTO MUNICIPAL UTILITY DISTRICT
RANCHO SECO
DOCKET NO. 50-312

The operation of the Rancho Seco Nuclear Generating Station by the Sacramento Municipal Utility District (SMUD) has been examined by the Office of Inspection and Enforcement since the operating license was issued on August 16, 1974. Since issuance of the operating license ten inspections have been performed, seven of which were unannounced. The total amount of time spent at the plant site during these inspections was approximately seventy-five man days.

The results of the NRC inspection program to date show that the Rancho Seco Nuclear Generating Station has been operated safely since initial startup in September 1974. The performance characteristics of the reactor and engineered safeguards systems have been determined during the startup test program through 92.6% of full power (2568 MWt). The startup test results have been found to meet the test acceptance criteria determined from the plant design bases described in the Final Safety Analysis Report.

The significant results of the inspection program that are pertinent to the proposed power increase from 2568 MWt to 2772 MWt are discussed below.

1. Startup Test Program

The startup test program commenced with initial fuel loading on August 19, 1974 and was completed through 92.6% of full power (2568 MWt) on March 22, 1975. Testing was performed at zero power, 15%, 40%, 75% and 92% of full power. The test data obtained during the performance of the following startup tests have been reviewed and evaluated by our inspectors.

a. Zero Power Physics Test
b. Nuclear Instrumentation Calibration at Power
c. Core Power Distribution
d. Reactor/Turbine Trip
e. Dropped Control Rod Assembly
f. Power Imbalance Detector Calibration
g. Nuclear Steam Supply System Heat Balance
h. Reactivity Coefficient at Power
i. Psuedo Control Rod Assembly Ejection

The evaluation of the Psuedo Control Rod Assembly Ejection test, conducted during zero power testing, revealed that the measured value of the rod worth differed significantly from the predicted value. The licensee informed our inspector that the predicted value was in error and a reevaluation of the physics calculation indicated that the measured worth should have been anticipated. The resolution of the psuedo ejected rod worth value and corrective action taken by the licensee are described in the licensee's letter to Licensing (RJR 74-401 dated October 23, 1974) and in Proposed Amendment No. 29 to the FSAR formally submitted on December 6, 1974. The data for all other tests evaluated by the inspector were found to be consistent with the predicted values of the parameters measured.

In addition to the above tests, the initial fuel loading, initial criticality, unit loss of electrical load test, and the loss of offsite power test were directly observed by our inspectors. During the witnessing of these tests, compliance with approved procedures was verified and the acceptability of test results was independently verified by our inspector with no anomalies observed.

A special test procedure, "Bleed and Feed Demonstration", was used by the licensee to satisfy the requirements of technical specification 3.12 which required that significant load changes be performed by operating personnel to demonstrate satisfactory system operability. Our inspector reviewed the raw test data obtained during the performance of the special test and found it consistent with the information contained in the licensee's performance report dated March 1975.

2. Plant Operations

Through observations of plant operation, examination of facility records, and discussion with licensee representatives, the operation of the facility has been found consistent with the requirements of the technical specifications. Tours of the facility have been made on each inspection. During these tours, observation of mechanical equipment and piping systems have shown no excessive leaks or vibrations. Instrumentation system including nuclear instrumentation, reactor protection and safety features actuation systems have been found to be operating normally with the required tests and calibrations being performed as scheduled. The unidentified reactor coolant leakage has averaged less than 0.4 gallon per minute throughout plant operation to date as compared to the technical specification limit of 1.0 gallon per minute.

There have been no major outages caused by the failure of safety related equipment or components. Two outages of significant duration have occurred since initial operation, a 26-day outage in October/November 1974 for the repair of condenser tube leakage and modification of the turbine stop valves and a 22-day outage in March/April 1975 for the inspection of turbine bearings. Maintenance of safety

related equipment has been recorded by the licensee in Monthly Operating Reports. Examination of maintenance records verified that the maintenance had been performed consistent with the licensee's management control system.

The plant availability factor since commercial operation at 92.6% power has been 100%. The capacity factor for the facility has been approximately 95% since commercial operation.

3. Unusual Occurrences

The licensee submitted fourteen (14) abnormal occurrence reports in 1974 and to date has submitted nine (9) abnormal occurrence reports in 1975. The circumstances and corrective action described by the licensee in each abnormal occurrence report have been verified during the inspection program.

4. Radiological Protection

Our inspectors have verified that the licensee's radiological protection program has been implemented consistent with regulatory requirements. Results of radiological surveys performed during plant startup tests indicate that the radiation zones, based on current data projected to 100% of full power, will be as described in the FSAR, Section 11.2.1.1.

The results of the radiological environmental monitoring program for the last two quarters of 1974 did not identify any significant adverse environmental effects resulting from the operation of the facility. The records of routine surveys performed by the licensee have shown that radiation and contamination levels in uncontrolled areas have been insignificant.

5. Quality Assurance Program for Operations

The implementation of a Quality Assurance Program for Operations was verified prior to the receipt of the Operating License. The quality assurance program has been subject to a continued examination by our inspectors during the initial operation phase. Items of noncompliance were identified related to inspection planning, receipt inspection performance, independence of inspection personnel and implementation of corrective action for deficiencies identified by the licensee's internal audit program. The licensee has promptly responded with corrective action commitments which were subsequently verified by our inspectors to satisfactorily resolve the enforcement items. All other operational activities have been found to be consistent with the requirements of the quality assurance program for operations.