UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 69 TO FACILITY OPERATING LICENSE NO. DPR-54

SACRAMENTO MUNICIPAL UTILITY DISTRICT

RANCHO SECO NUCLEAR GENERATING STATION

DOCKET NO. 50-312

1.0 INTRODUCTION

By letter dated December 17, 1984, as supplemented by letters dated March 14, 1985, and April 9, 1985, (Ref. 1) the Sacramento Municipal Utility District (SMUD or the licensee) submitted an application to modify the Technical Specifications for the Rancho Seco Nuclear Generating Station to permit operation for Cycle 7. A Cycle 7 reload report was also submitted with the above letters (Ref. 2). The fuel system, the nuclear, the thermal-hydraulic and the accident and transient analyses of this reload are presented in the Reference 2 report and their evaluation follows. An evaluation of the proposed Technical Specification changes is also presented.

1.1 Description of the Cycle 7 Reload

The Rancho Seco reload reactor core consists of 177 fuel assemblies, each of which is a 15 by 15 array containing 208 fuel rods, 16 control rod guide tubes and one in-core instrument guide tube. The fuel pellets are dished on the ends and the cladding is Zircaloy-4. All fuel assemblies, including the 96 axial blanket assemblies and four axial blanket lead test assemblies, maintain a constant nominal fuel loading of 436.6 kg of uranium. There are seven different batches of fuel assemblies present in the Cycle 7 loading. Only 100 assemblies (40 of batch 8B, 56 of batch 9 and 4 lead test assemblies in batch 7) have axial blankets. The maximum initial enrichment of 3.43 w/o in U-235 is in batch 8B. There are 61 Ag-In-Cd full length control rods and 56 burnable poison rod assemblies. Finally, eight axial power shaping rods are provided for additional control of the axial power distribution.

The design of the Cycle 7 core was compared to the design of the Cycle 6 core which was used as the reference cycle.

2.0 FUEL SYSTEM DESIGN EVALUATION

2.1 Fuel Assembly Mechanical Design

All fuel assemblies used in this reload are mechanically interchangeable into any core position. The fresh batch 9 (Mark B5Z) fuel has an axial blanket at the ends and incorporates Zircaloy spacer grids (instead of Inconel). The axial blanket design has been approved on the basis of the performance of lead test assemblies in batch 7 (Ref. 3). Zircaloy results in a better power distribution and better neutron economy compared to Inconel (Ref. 4). In addition, the batch 9 holddown springs are made of Inconel 718 (instead of Inconel 750) providing additional operating margin.

The Cycle 7 reload will include two regenerative neutron sources. For Mark B5 assemblies, the neutron source is built into the burnable poison rod assemblies. The neutron source retainer design has been approved (References 5 and 6).

2.2 Cladding Stress, Strain and Collapse

The batch 6B power history has been shown to be the most limiting for cladding creep collapse for Cycle 7. This case was compared with a generic analysis (which included increased peaking due to very low leakage) based on Reference 7. The analysis predicts a collapse time larger than 33,000 Effective Full Power Hours (EFPH) while the expected residence time is 29,213 EFPH.

The cladding stress parameters for Cycle 7 are enveloped by a conservative analysis performed for Cycle 6, i.e., the reference cycle. The fuel pellet was designed such that the design criteria of 1.0% plastic tensile strain is not to be exceeded for the worst local burnup and heat generation rate. The strain analysis is also conservative based on the upper and lower tolerance values of the pellet and the cladding, respectively.

Based on the above results, the cladding design was found to be acceptable.

2.3 Fuel Thermal Design

The Cycle 7 fuel analysis was performed using the approved TACO2 code (Ref. 8). The design of batch 9 (i.e., the new fuel in Cycle 7) is such as to be equivalent to the thermal design of the fuel batches present in Cycle 7 in the remainder of the core. Densification effects were taken into account and the core protection limits were based on 20.4 kW/ft linear heat generation rate for centerline fuel melt.

The TACO2 results indicate that the end of Cycle 7 maximum burnup will be (in batch 6B fuel) less than 43,000 MWD/MTU. The maximum burnup fuel internal pressure will be less than the nominal reactor coolant pressure of 2,200 psia. The results, therefore, for the fuel thermal design are acceptable.

3.0 NUCLEAR DESIGN EVALUATION

Physics characteristics for Cycle 7 have been calculated and compared to those corresponding to Cycle 6. The comparisons show that the physics parameters of the two cycles are similar. Analysis of the shutdown margin indicates that the minimum value is 2.57% $\Delta k/k$ compared to the required shutdown margin of 1.0% $\Delta k/k$. The new assemblies in Cycle 7 (batch 9) contain an axial blanket, the design of which has been previously approved for Cycles 5 and 6. In addition, batch 9 fuel assemblies contain Zircaloy grids and "short-stack" lumped burnable poisons and gray axial power shaping rods.

The physics analysis was performed using the NOODLE code which is a multidimensional, two group reactor simulator (Ref. 9). The code has been compared to analytical results derived from PDQ07 and measured results and was shown to be of acceptable accuracy. The methodology of the NOODLE code has been recently approved by the NRC staff.

The axial power shaping rods will be withdrawn from the core during the last 30 EFPDs. Analysis has shown that during this period, the axial stability index

will be negative, demonstrating that reactor operation will be stable.

The results of the Cycle 7 reactor core physics analysis were found to be acceptable.

4.0 THERMAL-HYDRAULIC DESIGN EVALUATION

In section 6.0 of Reference 2, the licensee described the thermal-hydraulic design for Cycle 7. Cycle 6 was used as the reference cycle for the thermal-hydraulic evaluation. Cycle 7 is a transition cycle to Mark BZ fuel assemblies from Mark B. The difference of the two designs is that the Mark BZ assemblies (with Zircaloy grids) have increased fuel assembly pressure drop, thus diverting more flow to the Mark B assemblies. The operation of the Mark BZ assemblies has been reviewed and approved in Section 5 of Reference 4. A transition core penalty was applied in the Cycle 7 radial peaking factors. Analyses indicate that there is sufficient DNB margin to cover the transition penalty. No fuel rod bow penalty was included in the DNB as justified in Reference 10.

The important thermal-hydraulic parameters of Cycle 7 are the same as the corresponding parameters in Cycle 6. The methods used have been reviewed and approved. Based on these results, we find the thermal-hydraulic design evaluation of Cycle 7 acceptable.

5.0 ACCIDENT AND TRANSIENT ANALYSIS EVALUATION

We have reviewed the information presented regarding the transient and accident analysis for Cycle 7. Comparison of the thermal-hydraulic parameters which determine the outcome of the transients with those used in Cycle 6 indicates that the values are similar. In addition, comparison of the kinetic parameters of Cycle 6 and the corresponding predicted values for Cycle 7 indicates that the Cycle 7 values are conservative with respect to their effect on transient analyses. Furthermore, a loss of coolant accident (LOCA) analysis was performed based on the method and criteria reported in Reference 11. Calculations show that

these criteria are conservative compared to those calculated for Cycle 7. On the basis that these parameters were calculated by methods previously used and approved for Rancho Seco, we conclude that the transient and accident analyses have been treated properly and are acceptable.

6.0 EVALUATION OF TECHNICAL SPECIFICATIONS CHANGES

We have reviewed the proposed changes to the following Technical Specifications for Cycle 7 operation:

2.1	Safety Limits, Reactor Core
2.3	Limiting Safety System Settings, Protective Instrumentation
3.2	High Pressure Injection and Chemical Addition Systems
3.5.2.5	Control Rod Positions

2.1 Safety Limits, Reactor Core

These modifications have been introduced because the BWC critical heat flux correlation was used in the calculation of the temperature - pressure limits. The BWC correlation has been approved. Therefore, we find the specified DNB limits acceptable to assure operation of the Rancho Seco Cycle 7 core within the applicable fuel design and performance criteria.

2.3 Limiting Safety System Settings, Protective Instrumentation

The modifications in this specification are a result of the BWC critical heat flux correlation and the DNB limits applied in the accident analysis. The BWC correlation has been approved. Therefore, we find the changes acceptable to assure operation of the Rancho Seco Cycle 7 core within the bounding values for the allowable LOCA peak linear heat rates.

3.2 High Pressure Injection and Chemical Addition Systems

This modification change was necessary to maintain a minimum of 1% $\Delta k/k$ subcritical reactivity for Cycle 7 under a cold shutdown condition with one stuck rod. The requirement is to increase the boric acid storage tank capacity from 9,105 gallons to 9,586 gallons of 7,100 ppm of boron in solution. The capacity of the present boric acid tank is 10,000 gallons. This modification has been derived using approved methods and the predicted Cycle 7 core characteristics are similar to those of previous core loadings.

3.5.2.5 Control Rod Positions

The allowable power imbalance envelope for Cycle 7 has been computed using approved methods in such a manner that specified acceptable fuel design limits will not exceed the acceptance criteria. We find the Technical Specification change acceptable.

7.0 EVALUATION FINDINGS

We have reviewed the fuels, physics, thermal-hydraulic and accident analysis information presented in the Rancho Seco, Cycle 7 reload report as stated above. We find the proposed reload and the associated modified Technical Specifications acceptable.

8.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously

issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

9.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) 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.

Dated: June 4, 1985

Principal Contributor: Lambros Lois

References

1. Letters: R. J. Rodriguez to Director, NRR, dated December 17, 1984, Serial: RJR 84-556, March 14, 1985, Serial No. RJR 85-114 and April 9, 1985, Serial: RJR 85-169.

2. BAW-1850, "Rancho Seco Unit 1 Cycle 7 Reload Report", Volume 2, October 1984.

3. BAW-1664, "Axial Blanket Lead Test Assembly", Licensing report, Babcock and Wilcox, Lynchburg, VA, September 1984.

4. BAW-1781P, "Mark BZ Fuel Assembly Design Report", Rancho Seco Cycle 7 Reload, Volume 1, Babcock and Wilcox, Lynchburg, VA, April 1983.

5. BAW-1496, "BPRA Retainer Design Report", Babcock and Wilcox, Lynchburg, VA, May 1978.

6. Letter: J. F. Stolz to J. H. Taylor (B&W) "BPRA Retainer Reinsertion", July 24, 1984.

7. BAW-10084A, Rev. 2, "Program to Determine In-Reactor Performance of B&W Fuels; Cladding Creep-Collapse", Babcock and Wilcox, Lynchburg, VA, October 1978.

8. BAW-10141PA, "Fuel Pin Performance Analysis", Y. H. Hsii, et.al., Babcock and Wilcox, Lynchburg, VA, June 1983.

9. BAW-10152, "NOODLE: A Multi-Dimensional, Two Group Reactor Simulator", C. W. Mays, et al., Babcock and Wilcox, Lynchburg, VA, Septebmer 1984.

10. BAW-10147P-A, Rev. 1, "Fuel Rod Bowing in B&W Fuel Design", J. C. Moxley, et. al., Babcock and Wilcox, Lynchburg, VA, May 1983.

11. BAW-10103, Rev. 1, "ECCS Analysis of B&W's 177-FA Lowered-Loop NSSS", Babcock and Wilcox, Lynchburg, VA, September 1975.