SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT □ 6201 S Street, Box 15830, Sacramento, California 95813; (916) 452-3211

August 11, 1980

Director of Nuclear Reactor Regulation
Attention: Mr. Robert W. Reid, Chief
Operating Reactors, Branch No. 4
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Docket No. 50-312
Proposed Amendment No. 70
Rancho Seco Nuclear Generating
Station, Unit No. 1

Dear Mr. Reid:

In accordance with 10 CFR 50.59, the Sacramento Municipal Utility District proposes to amend its operating license, DPR-54, for Rancho Seco Nuclear Generating Station No. 1, by submitting Proposed Amendment No. 70 on August 18, 1980. Today, we are submitting forty (40) copies of Proposed Amendment No. 70 which shows the changes we are proposing. Under separate cover, we will be providing payment for this submittal as required per 10 CFR 170.

We request your expeditious review and issue of this Proposed Amendment to our license to assist in resolving our concern that we not inadvertently challenge the Reactor Protection System due to spurious Reactor Coolant System flow oscillations. This revision will change the safety analysis assumed "flux to flow" ratio from 1.05 to 1.08. Such a change is justified as a result of reanalysis performed by our fuel supplier which takes credit for the decrease in reactor core bypass flow due to the installation of 52 lumped burnable poison assemblies within the core. The resulting increase in effective core coolant flow has been incorporated into the safety analysis and demonstrates that Rancho Seco can operate at the design power level and meet all transient and safety considerations successfully with the new power to flow ratio. In support of this request, we are submitting revised text for Sections 6 and 7 in the Cycle 4 Reload Report, BAW-1560.

Sincerely,

J. J. Mattimoe

J. J. Mattimoe
Assistant General Manager
and Chief Engineer

JJM: RWC:jr

Sworn to and subscribed before me
this 21st day of August, 1980.

Patricia K. Geisler

Notary Public

OFFICIAL SEAL
PATRICIA K. GEISLER
NOTARY PUBLIC CALIFORNIA
PRINCIPAL OFFICE IN
SACRAMENTO COUNTY
My Commission Expires November 22, 1983

8009050 240

6. THERMAL-HYDRAULIC DESIGN

The incoming batch 6 fuel is hydraulically and geometrically similar to the fuel remaining in the core from previous cycles. The thermal-hydraulic design evaluation supporting cycle 4 operation utilized the methods and models described in references 2, 3, and 7 except for the core bypass flow and the inclusion of retainers to provide positive holddown of burnable poison rod assemblies (BPRAs) and neutron sources.

The maximum core bypass flow due to the removal of all orifice rod assemblies (ORAs) in cycle 3 increased to 10.4%. For cycle 4 operation, 52 BPRAs will be inserted, leaving 56 vacant fuel assemblies and resulting in a decrease in calculated maximum core bypass flow to 8.3%. The BPRA retainers introduce a small DNBR penalty, as discussed in reference 1. Reactor core safety limits have been re-evaluated based on the insertion of these BPRAs with retainers and increased core flow. The cycle 3 and 4 maximum design conditions and significant parameters are shown in Table 6-1. The increase in core flow more than compensates for the decrease in DNBR due to the BPRA retainers so that the cycle 3 analysis is conservative and applicable to cycle 4.

A flux/flow trip setpoint of 1.08 has been established for cycle 4 operation. This setpoint value maintains a DNBR margin which is greater than 10%, relative to the design minimum DNBR of 1.30 (BAW-2). The previous setpoint value of 1.05 had initially been established for first core operation and has been verified as conservative and, therefore, applicable for succeeding cycles.

In response to reference 8, B&W has committed to prepare a topical report addressing the potential for and effects of fuel rod bow. In addition, B&W has submitted an interim rod bow penalty evaluation procedure [9,10] for use until the topical report is completed and reviewed.

The rod bow penalty applicable to cycle 4 was calculated using the interim rod bow penalty evaluation procedure. As in the previous cycle the burnup is based on the maximum fuel assembly burnup of the batch that contains the fuel assembly with the maximum radial x local peak. For cycle 4 this burnup is 14,537 MWd/mtU, a batch 6 fuel assembly. The calculated penalty using this procedure is less than 0.8%. Utilizing the 1% DNB credit for the flow area reduction factor, the actual penalty applied to the DNB calculations is zero.

7. ACCIDENT AND TRANSIENT ANALYSIS

7.1. General Safety Analysis

Each FSAR[2] accident analysis has been examined with respect to changes in cycle 4 parameters to determine the effect of the cycle 4 reload and to ensure that thermal performance during hypothetical transients is not degraded. The effects of fuel densification on the FSAR accident results have been evaluated and are reported in BAW-1393.[7] Since the cycle 4 parameters are conservative with respect to the reference 7 report, the conclusions in that reference are still valid.

Improved fuel utilization and the inherent increase in core average burnup experienced in cycle 4 have resulted in a higher plutonium-to-uranium fission ratio than that used in the FASR. A study of the major FSAR Chapter 14 accidents using the cycle 4 iodine and noble gas inventories concluded that the thyroid and whole body doses were well below the 10 CFR 100 limits.

7.2. Accident Evaluation

The key parameters that have the greatest effect on determining the outcome of a transient can typically be classified in three major areas: core thermal, thermal-hydraulic, and kinetics parameters, including the reactivity feedback coefficients and control rod worths.

Core thermal properties used in the FSAR accident analysis were design operating values based on calculational values plus uncertainties. First-core values[2] of core thermal parameters and subsequent fuel batches are compared to those used in cycle 4 analyses in Table 4-2. The cycle 4 thermal-hydraulic maximum design conditions are compared to cycle 3 values to Table 6-1. These parameters are common to all the accidents considered in this report. A comparison of the key kinetics parameters from the FSAR and cycle 4 is provided in Table 7-1. Cycle 4 parameters include the effects of removing the orifice rod assemblies. Additionally, all accident analyses, and their related evaluations, continue to be valid with respect to the new flux/flow setpoint of 1.08.