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

June 17, 1980

Mr. R. H. Engelken, Director
Region V Office of Inspection and Enforcement
U. S. Nuclear Regulatory Commission
1990 North California Boulevard
Walnut Creek Plaza, Suite 202
Walnut Creek, California 94936

Re: Operating License DPR-54
Docket No. 50-312
Reportable Occurrence 80-31

Dear Mr. Engelken:

In accordance with Technical Specifications for Rancho Seco Nuclear Generating Station, Section 6.9.4.2b and Regulatory Guide 1.16, Revision 4, Section C.2.b(2), the Sacramento Municipal Utility District is hereby submitting a thirty-day report of Reportable Occurrence 80-31.

During a review of all work performed in relation to I&E Bulletin 79-14, it was discovered that a particular stress analysis of an "as found" piping configuration had been overlooked.

To accomplish the requirements of I&E Bulletin 79-14, the District contracted the Architect Engineer, Bechtel Norwalk, to assist in the review and analysis. During the inspection it was noted that the isometric drawings had not shown a 6-inch branch connection at a 10-inch elbow in line No. 26120 - 10" - GD. This 10-inch line is the discharge piping between the DHR pump P-261A and the DHR cooler E-260A. The 6-inch branch line is the DHR cooler bypass line. The support, No. 4U-26120-6, and associated stress problem, was flagged for reanalysis. However, the reanalysis was overlooked and not included in the final submittal of the I&E Bulletin 79-14 response.

On July 3, 1980, the District was notified by Bechtel that as a result of a review of 79-14 work the error was discovered. A reanalysis was performed on the line and indicated that the stresses calculated exceed the allowable limits.

Upon receipt of this information, the "A" DHR system was declared inoperable. Work was started on modification of the support to alleviate the overstressed condition by limiting N-S pipe movement. This was completed within

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The 48 hours allowable per Technical Specifications Section 3.3.1 for continued reactor operation. Additionally, a complete review of all 79-14 stress problems has been made to assure all problems subject to reanalysis have been identified and the reanalysis completed.

There were no transients nor power reductions associated with this event.

Respectfully submitted,

J. J. Mattimoe

J. J. Mattimoe
Assistant General Manager
and Chief Engineer

JJM:HH:jr

cs: Director, I&E (30)
Director, MIPC (3)