8005120411

SMUD

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

May 5, 1980

Mr. Darrell G. Eisenhut, Acting Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Eisenhut:

Your letter of April 3, 1980 discusses concerns expressed in an attached memorandum that the transient of March 20, 1978 reported as LER 78-1 could have led to results as serious as those experienced at Three Mile Island (TMI). The letter further requires that we provide comments on the memorandum.

The District has reviewed the letter and memorandum and prepared the following comments identified by reference to individual portions of the documents:

Eisenhut to Mattimoe letter, April 3, 1980

A. NRC comment, first paragraph - "The memorandum states that after the steam generators boiled dry during your March 1978 transient, the only reason that flow was re-established to the steam generators was that a level indication randomly drifted low enough to cause flow from the Auxiliary Feedwater System to automatically activate."

District response - The statement is not accurate because the operator recognized that the OTSG's were dry and he took action to admit main feedwater. Since the ICS sensed main feed pumps trip because of low discharge pressure (700 psig), it placed the pump controls in a psuedo mode which held pump speed at about 2,200 RPM. The operator defeated that control input for "A" feed pump and raised speed to 3,500 RPM. Water was then admitted to "A" OTSG through the startup feedwater valve. Because the operator's action to admit main feedwater occurred at approximately the same time as the automatic opening of the auxiliary feedwater valve, there is no way to be certain which means provided water first. The important point is that the operator had a means of providing main feedwater to the OTSG's, recognized the need for it, and took action to initiate flow.

B. NRC comment, first paragraph - "It is hypothesized that the steam generator level indication might not have drifted low, and the results would have been boiling in the core without the benefit of high pressure injection flow and without operator knowledge of system indication."

District response - Contrary to the hypothesis, the operators were aware of reactor coolant system (RCS) pressure by observation of the operable SFAS pressure recorder. In addition they monitored valid pressurizer level signals. The operators had foremost concern for maintaining RCS pressure at or near normal (2,000 - 2,200 psig) to prevent boiling and for providing a heat sink by assuring that feedwater was supplied to the OTSG's. The latter was accomplished as detailed in response (A) above. In addition, the operator manually initiated high pressure injection immediately after the trip.

Bernero-Rowsome memorandum, dated March 14, 1980

A. Safety Problem No. 1, Affect on severity of ATWS event; common cause failure to scram.

District response - The District has previously evaluated the grounding techniques utilized in the reactor protection system (RPS). All of the RPS cabinets are bolted and welded together; each channel pair of cabinets is connected to ground by a heavy braided cable. The cabinets are also connected to the vital bus ground. The instruments inside the cabinets are grounded to the frame by copper ground straps in at least 10 locations per cabinet, and the two power supplies are grounded to terminal strips on the frame as well as through their slide mounts. With this design, a loss of ground is not considered credible so the potential safety hazard does not exist.

A special test (STP 609, RPS Instrument Ground Verification) was performed in March 1978 and verified the integrity of the grounding system. As a result of these determinations, the District does not consider a fail-to-trip scenario based on RPS ground fault to be credible.

B. Safety Problem No. 2, Continuance of steam flow and depletion of steam available for turbine driven emergency feedwater pumps.

District response - The Rancho Seco auxiliary feedwater system design employs two pumps, one having a dual drive (turbine plus Class 1E power supplied motor), the other having a motor drive supplied from a separate Class 1E source. Supply of auxiliary feedwater to the OTSG's is, therefore, not dependent solely on a steam source from the OTSG's but can be reliably supplied by electric drives.

C. Safety Problem No. 3, Protection system for main steam line breaks.

District response - At Rancho Seco the protection system for main steam line breaks is called Steam Line Failure Logic. Each main steam line has its own protection system which is actuated when its pressure falls to 435 psig. Upon protection system actuation, main feedwater to the affected OTSG is isolated, however, the unaffected OTSG may still receive main feedwater and the ability to supply auxiliary feedwater to both OTSG's remains unaffected. Steam pressure to operate the turbine-driven auxiliary feed pump would, therefore, be available, backed-up by two Class 1E motors (one for each pump). Operator action to provide auxiliary feedwater for the steam line break scenario is not required. The District concludes that the Rancho Seco steam line failure logic is useful but not "counter-productive," as was stated in the memorandum.

As a general comment, the District notes that Bernero and Rowsome expressed confidence in the comparative safety of Rancho Seco with respect to the scenario in question. Rancho Seco has operated safely and the design changes and training since the March 20, 1978 incident have served to enhance the margin of safety.

Respectfully,

J. J. Mattimoe

J. J. Mattimoe
Assistant General Manager
and Chief Engineer

JJM:JVM:cmb

cs: Director, Region V I&E (3)
MIPC (3)