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

November 26, 1979

Mr. Robert W. Reid, Chief
Operating Reactors Branch No. 4
Division of Operating Reactors
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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

Dear Mr. Reid:

The Sacramento Municipal Utility District has reviewed Mr. Harold Denton's letter of October 30, 1979. The District notes the additional requirements contained within some of the clarification items. We will include those added items in our effort.

Attachment 1 contains as much additional detail as is currently available for each NUREG - 0578 item with the exception of auxiliary feedwater. The District's letter of November 19, 1979 contains information related to auxiliary feedwater intention.

The November 19, 1979 letter also conveys our justification for choosing refueling 1980 and refueling 1981 as implementation dates rather than January 1, 1980 and January 1, 1981 respectively. The schedule provided in this document is based on the fact that our next refueling is scheduled for January 19, 1980 and on the assumption that the Commission will equate this date, or in effect the return to power date, to their recommended completion date of January 1, 1980. This assumption allows us to complete most of the January 1, 1980 tasks by the return to power date and to perform many of them in a more efficient, better engineered manner. There will be some items however, that cannot be completed by the end of refueling due to long delivery or other matters beyond our control. It should also be noted that some of the items scheduled for completion by the end of the 1980 refueling could be completed by January 1, 1980, but only by jeopardizing or adversely effecting the total schedule.

Sincerely,

John J. Mattimoe

John J. Mattimoe
Assistant General Manager
and Chief Engineer

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2.1.1 Emergency Power Supply - Pressurizer Heater

The District will power two 126kW sets of pressurizer heaters from separate sets of 480 volt switchgear. The switchgear is safety grade with either an offsite or a diesel generator power source. Heater supply breakers will meet the requirements of Regulatory Guide 1.75 Revision 2 and IEEE Standard 384 of 1974. Upon actuation of the Safety Features Actuation System, pressurizer heater breakers will trip open. Capability will exist to re-energize the heaters from the control room.

This pressurizer heater power supply change will require changes in switchgear required for plant operation. The modifications will be completed during the January 1980 refueling outage.

Prior to startup after the 1980 refueling outage, procedures will be available to provide guidance for re-energizing the heaters.

2.1.1 Emergency Power Supply--PORV Block Valve, PORV and Pressurizer Level

By the end of the 1980 refueling outage the PORV block valve will be powered from a vital bus capable of being supplied by either the diesel generator or offsite power. Switchover in power source can be done automatically or manually in the control room.

The PORV is powered from a battery which may be charged by either a diesel generator or by offsite power. The battery is not safety grade though it is of the same qualification category as the PORV control circuits and the PORV solenoid. Upgrading the power source does not seem likely to improve reliability of the system. In addition, the valve fails closed on loss of power. Therefore, failure of the power source will not cause a failed open PORV. The District intends to retain its current power scheme for the PORV.

The PORV can be opened and closed from the control room. By the end of the refueling outage the operator will be able to open the PORV if RCS pressure is greater than 600 psig and to close it if RCS pressure is less than 2400 psig. The operator will be able to perform these operations from the control room.

The origin of power for pressurizer level is a vital instrument bus with the capability of being supplied from an offsite power source or an emergency power source.

2.1.2 Performance testing for BWR and PWR Relief and Safety Valves

See the District's NUREG 0578 response letter dated October 18, 1979.

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2.1.3.a Direct Indication of Power-Operated Relief Valve and Safety Valve Position Indication for PWR's and BWRs

The District intends to install accoustic PORV and safety valve position monitors during the refueling outage. Each valve will be monitored by two accelerometers. A charge amplifier will amplify each accelerometer output. The signal will then be sent to a signal processing unit mounted in the computer room just outside of the control room. Valve position indication and annunciation for each monitored valve will be available in the control room.

The District currently believes the system will be seismically and environmentally qualified upon initial installation with the exception of the signal processor and the control room mounting location. We hope to have the signal processor qualified by the 1981 refueling outage.

Prior to startup after the 1980 refueling outage, all portions of the system within the reactor building will be installed. The signal conditioner may not be available until May of 1980. It will be installed on receipt to make the system operable.

2.1.3.b Instrumentation for Detection of Inadequate Core Cooling in PWRs and BWRs--Subcooling Meter

The District intends to install redundant subcooling (T_{SAT}) meters during the January 1980 refueling outage. Meters will indicate degrees below saturation temperature for existing pressure conditions. One will be installed for each T_h leg. Each indication system will have two hot leg temperature inputs and one reactor coolant pressure input. The system will be safety grade with the possible exception of the meter mounting panel in the control room. Adequate space for meters on seismically qualified panels may not exist in the control room. Temperature and pressure sources will be appropriately isolated.

Specific subcooling meter information requested by Dr. Denton's letter of October 30, 1979 follows:

INFORMATION REQUIRED ON THE SUBCOOLING METER

Display

Information Displayed (T-Tsat, Tsat, Press, etc.)	T-Tsat
Display Type (Analog, Digital, CRT)	Digital
Continuous or on Demand	Continuous
Single or Redundant Display	Redundant
Location of Display	Control Room H2PSB Panel
Alarms (include setpoints)	2 pre-programmed alarm setpoints 30°F 50°F
Overall uncertainty (°F, PSI)	4°F
Range of Display	650°F
Qualifications (seismic, environmental, IEEE323)	IEEE 323 & IEEE 344. Meets Class IE seismic & environmental and safety

Calculator

Type (process computer, dedicated digital or analog calc.)	Dedicated Digital
If process computer is used specify availability. (% of time)	N/A
Single or redundant calculators	Redundant
Selection Logic (highest T., lowest press)	Highest T
Qualifications (seismic, environmental, IEEE323)	IEEE 323 & IEEE 344. Meets Class IE seismic & environmental and safety
Calculational Technique (Steam Tables, Functional Fit, ranges)	Steam, Tables and interpolation routines

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<u>Input</u>

Temperature (RTD's or T/C's)	RTD's
Temperature (number of sensors and locations)	4, 2 in each Hot Leg
Range of temperature sensors	120 - 650^{O}F
Uncertainty* of temperature sensors (^{O}F at 1)	Unknown at this time
Qualifications (seismic, environmental, IEEE323)	Class 1 seismic & environmental
Pressure (specify instrument used)	Foxboro E11GH
Pressure (number of sensors and locations)	2, One in each Hot Leg
Range of Pressure sensors	0 -2500 psi
Uncertainty* of pressure sensors (PSI at 1)	Unknown at this time
Qualifications (seismic, environmental, IEEE323)	Class 1 seismic & environmental

<u>Backup Capability</u>

Availability of Temp & Press	Control room meters and computer
Availability of Steam Tables, etc.	Steam Tables and saturation graph available in Control Room
Training of operators	Operators trained during plant shutdown May, 1979. Verified by I & E examination.
Procedures	Primary procedure is D.5 Loss of Reactor Coolant/ Reactor Coolant System Pressure

*Uncertainties must address conditions of forced flow and natural circulation

1473 005

The District's October 18, 1979 letter presented the program for inadequate core cooling guideline development. The first set of guidelines for the case of low coolant inventory without reactor coolant pumps have been received. The guidelines were incorporated into B & W's existing small break guidelines. These revised guidelines and the supporting analyses are supplied as attachments 2 and 3. Operating procedures incorporating these guidelines and associated training will be completed after NRC approval of the guidelines.

2.1.3.c Instrumentation for Detection of Inadequate Core Cooling--Additional Instrumentation

As pointed out in the District's October 18, 1979 letter concerning NUREG - 0578, inadequate core cooling guidelines are now being developed. Results of the first phase of the effort are included as attachments 2 and 3 to this letter. Final phases of the effort will be received from B & W on December 21, 1979. At that time, B & W, the District, and other members of the B & W Owner's Group will thoroughly review the guidelines and determine if additional instrumentation is needed. Reactor vessel water level indication will be one of the considered instruments. The review will identify necessary additional instrumentation by January 31, 1980. The B & W Owner's Group presented the schedule to the NRC staff on September 13, 1979 as indicated in NRC minutes of the meeting. Work has been proceeding per the schedule. We hope the District would have been informed if the schedule was unsatisfactory.

2.1.4 Containment Isolation

See the District's NUREG - 0578 response letter dated October 18, 1979.

2.1.5.a Dedicated Penetrations for External Recombiner or Post-Accident Purge Systems

See the District's NUREG - 0578 response dated October 18, 1979.

2.1.5.b Inerting BWR Containment

Not applicable to Ranch Seco.

2.1.5.c Capability to Install Hydrogen Recombiner at Each Light Water Nuclear Power Plant

Not applicable to Rancho Seco.

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2.1.6.a Integrity of Systems Outside Containment Likely to Contain Radioactive Material for PWRs and BWRs

As stated in our October 18, 1979 response to NUREG - 0578, the District intends to implement a continuing leak reduction program at Rancho Seco. We are very concerned about the acceptance standards for the program. In response to requests for guidance on acceptability, we have in essence been told that though the NRC staff cannot define what it wants, we must have an acceptable program defined and completed by January 1, 1980.

We are working to define systems which could contain highly radioactive fluids. Those systems will be included in the leak reduction programs.

For each of those systems in the leak reduction program, the District will inspect all components accessible during normal operation for such visible signs of leakage as packing and seal leaks, vent and drain leaks and boric acid crystals.

By January 1, 1980 the District will supply to the NRC staff:

(1) a list of all systems covered by the leak reduction program.

(2) a list of components for which visual inspection has shown that leakage reduction will be undertaken. The components will be worked during the refueling outage scheduled to start in January 1980.

By the end of the refueling outage scheduled to commence in January 1980 the District will conduct a helium leak test of the waste gas system. Results of that test will be used to determine leaks which will be reduced. In doing the helium leak testing, we will be mindful of the direction given in the D. G. Eisenhut to operating reactors letter of October 17, 1979.

One month after startup after the refueling outage scheduled for January 1980, the District will provide a list of components which exhibit discernible leakage in systems covered by the leak reduction program.

This leakage reduction program will be repeated each refueling cycle.

2.1.6.b Design Review of Plant Shielding and Environmental Qualification of Equipment for Spaces/Systems Which May be Used in Post Accident Situations.

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In the letter dated October 18, 1979, the District stated that it expects to complete a review of systems and surrounding spaces by February 15, 1980. For a full review even this date may be optimistic.

However, as the review progresses we are hoping to be able to eliminate some of the analyses we are now planning to do.

Isotopic half lives coupled with system flow rates may eliminate evaluation of some systems.

By January 1, 1980 the District will provide a summary of those analyses completed. On that date we hope to determine dose rates for high source level operation of the decay heat system, reactor building spray system and the reactor coolant sampling systems. We will continue to use February 15, 1980 as a realistic date for completion of the full dose rate review.

2.1.7.a Auto Initiation of the Auxiliary Feedwater System (AFWS)

See the District's letter to you dated November 19, 1979.

2.1.7.b Auxiliary Feedwater Flow Indication to Steam Generator

See the District's letter to you dated November 19, 1979.

2.1.8.a Improved Post-Accident Sampling Capability

The District is actively working on development of high source level sampling capability. We will consider the additional requirements outlined in Dr. Denton's October 30, 1979 letter.

We are examining many possible methods to obtain and analyze samples. These methods include:

1. Batch dilution of pressurized samples.
2. Batch dilution of unpressurized samples.
3. Batch dilution of dissolved gas samples.
4. Remote instruments for containment air monitoring.
5. Remote instruments for reactor coolant letdown analysis.

Preliminary results show that most of the sampling and analysis can be achieved in the long term with significant plant design changes. In the short term we will not be able to sample at the high levels required by NUREG - 0578 and still remain within specified exposure limits. We will continue to work toward that goal.

2.1.8.b Increased Range of Radiation Monitors

The District notes the additional requirement for interim effluent particulate monitoring capability by January 1, 1980. We are investigating the possibility of providing this interim capability prior to startup after the 1980 refueling outage.

See the Districts NUREG - 0578 response letter dated October 18, 1979 for previous commitment. See the Districts letter of November 19, 1979 for justification of refueling outage 1981 in lieu of January 1, 1981 for final completion of items.

2.1.8.c Improved In-Plant Iodine Instrumentation

The District will have improved in-plant airborne iodine instrumentation and associated procedures and training prior to startup after refueling 1980.

2.1.9 Analysis of Design and Off-Normal Transients and Accidents

As described in the October 18, 1979 letter, the District is participating in the Abnormal Transient Operation Guideline (ATOG) program. The B & W Owner's Group explained the program to the NRC staff at the September 13, 1979 meeting with the staff.

Guideline development will demand many man months of operator review. According to the original schedule, the District was the third of the B & W Owner's Group utilities to complete the effort. However, this schedule would have required much operator effort during the refueling outage. Therefore, in order to minimize operator demand during the refueling outage, Rancho Seco has been shifted to the sixth utility position with completion scheduled for June 6, 1980.

2.2.1.a Shift Supervisor Responsibilities

The District's NUREG - 0578 response dated October 18, 1979 considered the information unofficially distributed at the September 26, 1979 implementation meeting in Las Vegas, Nevada. We assume the clarification to 2.2.1.a in Dr. Denton's October 30, 1979 letter officially transmitted the information from that meeting. Therefore, our response remains the same.

2.2.1.b Shift Technical Advisor

DISTRICT'S INTENT TO COMPLY WITH STA'S PRIOR TO STARTUP AFTER THE 1980 REFUELING OUTAGE

The District intends to comply with the onsite Shift Technical Advisor requirement utilizing present staff engineers by startup after the 1980 refueling outage. The qualifications of these engineers will meet the present ANSI-3.1 draft for Shift Technical Advisors, "Qualifications and Training of Nuclear Power Plant Personnel," which is "one year of nuclear power plant experience, 6 months of which must be associated with the nuclear power plant for which the individual is serving in the capacity of Shift Technical Advisor." These staff engineers will meet the above requirements.

The District will assign one staff engineer to the duty of Shift Technical Advisor for a period of 24 hours. This individual will remain on the Rancho Seco site during this 24-hour period of time and will be available within 10 minutes of being summoned by the shift supervisor. It is not the District's intent to man the position of Shift Technical Advisor during periods when the plant is off the line for maintenance or refueling outages. Since he will be in this duty status for periods of time longer than one shift, he will, therefore, sleep during some portion of the 24-hour duty cycle. This individual will be present during all shift supervisor reliefs and will be abreast of the plant status and any future changes to the plant status during the duty.

The on-duty Shift Technical Advisor will have ready access to an on-call management individual. This management individual will be a senior licensed operator and will have gone through the training related to the TMI accident. He will be up-to-date on the latest small break guidelines and operating procedures.

It is the District's contention that the on-duty staff engineer who is familiar within his area of engineering expertise with the Rancho Seco Station combined with the availability of readily obtainable assistance from an on-call management individual, who is a senior licensed operator, does enhance the accident assessment function in the short term.

The approach to operating experience and assessment capabilities has not yet been resolved.

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DISTRICT'S LONG-TERM INTENT TO COMPLY WITH STA'S BY 1-1-81

The District is presently recruiting in order to hire 6 graduate engineers. It is the District's intent that no more than two engineers from any one discipline will be hired in order to diversify the experience within this group. By this it is meant no more than two nuclear engineers, two electrical engineers, etc. These engineers will be placed into a formal training program which will more than meet the requirements of NUREG - 0578 and the philosophy of their position. Once they have completed their training, they will be designated as Shift Technical Advisors.

The present thinking within the District is to have one individual perform the duty of STA for a period of 48 hours. Naturally, due to the period of time, the individual will be asleep during some period of his 48-hour duty. The 10-minute response availability will be assured. This same group of individuals, due to their diversified engineering background, will also perform the operating experience assessment function. It is not the District's intent to man the position of Shift Technical Advisor during periods when the plant is off the line for maintenance or refueling outages.

The District feels that by January 1, 1981 fully-trained Shift Technical Advisors will be on duty. This will enhance the accident and operating experience assessment function at the Rancho Seco Plant.

2.2.1.c Shift and Relief Turnover Procedures

See the District's NUREG - 0578 response dated October 18, 1979

2.2.2.a Control Room Access

See the District's NUREG - 0578 response dated October 18, 1979

2.2.2.b Onsite Technical Support Center

As stated in our October 18, 1979 NUREG - 0578 response letter, the District will have an interim onsite technical support center by January 1, 1980.

The District is currently working toward defining the requirements of the final technical support center. Much input to that definition effort remains to be obtained. For example, communications requirements for the offsite emergency operations center for federal, state and local personnel remains to be specified. Final resolution of those requirements may not be forthcoming until July 1, 1980. An another example, inadequate core cooling instrumentation will not be defined until late January, 1980. If such instrumentation is determined to be necessary, it should probably be included in the final technical support center.

Many more unresolved situations related to the final technical support center currently exist. The District believes that a January 1, 1980 response on technical support center configuration would serve as little more than a paperwork exercise. By March 1, 1980 we 'ieve that we can supply a response with significantly greater substance.

NUREG - 0578 Implementing Letter of September 13, 1979 Encl. 3, Item 3 (1) — Containment Pressure Monitor

See the District's NUREG - 0578 response of October 18, 1979. For justification of installation date see the District letter to Dr. Denton dated November 19, 1979.

NUREG - 0578 Implementing Letter of September 13, 1979 Encl. 3, Item 3 (2) — Containment Hydrogen Concentration Monitor

See the District's NUREG - 0578 response of October 18, 1979. For justification of installation date see the District letter to Dr. Denton dated November 19, 1979.

NUREG - 0578 Implementing Letter of September 13, 1979 Encl. 3, Item 3 (3) — Containment Water Level

See the District's NUREG - 0578 response of October 18, 1979. For justification of installation date see the District letter to Dr. Denton dated November 19, 1979.

NUREG - 0578 Implementing Letter of September 13, 1979 Encl. 4 — Reactor Coolant System Vents

As stated in the NUREG - 0578 commitment letter of October 18, 1979, the District will install reactor coolant system vents during the refueling outage scheduled for 1981. As justification for choosing this date see the District's letter to Dr. Denton dated November 19, 1979.

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Babcock and Wilcox is currently doing preliminary design and analysis work on reactor coolant system vents. Their recommendations will be transmitted to the District on December 21, 1979. The District will then carefully review the preliminary design. If the preliminary design is accepted we will start analyzing combustable gas control in the containment, power supply design, and piping layout. A failure modes and effects analysis per IEEE - 279 will be done. We will procure hardware as design progress allows. This effort can probably be complete by May 1, 1980.

Procedures for venting will be available prior to startup after the 1981 refueling outage.

1473 013