December 4, 1979

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	
SACRAMENTO MUNICIPAL UTILITY DISTRICT	)	Docket No. 50-312
	)	
(Rancno Seco Nuclear Generating Station)	)	

LICENSEE'S ANSWERS (SET NO. 1) TO THE FIRST SET OF INTERROGATORIES OF THE CALIFORNIA ENERGY COMMISSION DATED NOVEMBER 15, 1979

1. INTERROGATORY: Identify and provide summaries and conclusions of any documents prepared since March 28, 1979, with respect to the facility or with respect to Babcock and Wilcox ("B&W") reactory systems which relate, in whole or in part, to any of the following:

   a. Small break loss-of-coolant accidents;
   b. Conditions of inadequate core cooling;
   c. Sensitivity evaluations of delays in start-up of the auxiliary feedwater system;
   d. Sensitivity evaluations of steam generator design parameters such as volume and hydraulic characteristics;
   e. Sensitivity evaluations of reactor trip setpoints, relief and safety valve setpoints, and ECCS setpoints;
   f. Sensitivity evaluations of operating reactor power level;
   g. Sensitivity evaluations of pressurizer size and hydraulic characteristics; and
   h. Sensitivity evaluations of reactor drain tank size and design pressure.

   ANSWER: Documents identified in response to this interrogatory will be provided pursuant to Licensee's Response to California Energy Commission's First Request for Production of Documents. Because the documents themselves will be produced, summaries and conclusions are not provided.

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2. INTERROGATORY: Describe the changes, if any, in facility design, equipment and/or operating procedures that have been proposed and/or instituted or are being studied or contemplated for the facility as a result of the Three Mile Island ("TMI") incident. Identify all documents related to any such changes.

ANSWER: The following changes in design, equipment or operating procedures have been instituted at Rancho Seco as a result of the Three Mile Island incident:

In a letter of April 16, 1979, SMUD informed the Nuclear Regulatory Commission of several changes to Rancho Seco operating procedures relating to the operation of the high pressure injection system, reactor coolant pumps, and operator utilization of pressure/temperature relationships.

In a letter of April 22, 1979, SMUD informed the Nuclear Regulatory Commission of a change which lowered the high pressure reactor trip setpoint from 2355 psig to 2300 psig, and raised the pilot operated relief valve setpoint from 2255 psig to 2450 psig.

In a letter of May 2, 1979, SMUD informed the Nuclear Regulatory Commission of procedure changes: for establishing and maintaining natural circulation; for consideration of reactor vessel integrity in the determination of high pressure injection system termination; to require prompt manual trip of the reactor on loss of feedwater to the steam generators, during a turbine trip, on loss of offsite power with loss of reactor coolant flow, on low steam generator level, or

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on low pressurizer level; and for reporting procedures for NRC notification any time the Rancho Seco reactor is not in a controlled or expected condition of operation.

In a letter of May 14, 1979, SMUD informed the Nuclear Regulatory Commission of several operating procedure changes relating to the operation of the auxiliary feedwater system. The NRC was also informed, in that letter, of a change in the control room annunciation for automatic start conditions of the auxiliary feedwater system, the addition of auxiliary feedwater flow indication in the control room, and the addition of the hard wired control grade reactor trip on loss of main feedwater or turbine trip. The control grade trip of the reactor on turbine trip or loss of main feedwater was further described to the NRC by a telecopy transmittal on May 30, 1979.

In a letter of August 27, 1979, SMUD informed the Nuclear Regulatory Commission of an operating procedure change requiring a trip of all operating reactor coolant pumps upon a reactor trip and initiation of high pressure injection caused by low reactor coolant system pressure.

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The following changes in design, equipment or operating procedures have been proposed for Rancho Seco as a result of the Three Mile Island incident:

In a letter of September 17, 1979, SMUD informed the Nuclear Regulatory Commission of several design changes it is considering which would improve the auxiliary feedwater system reliability.

In a letter of May 21, 1979, SMUD informed the Nuclear Regulatory Commission of a proposed change for a safety grade automatic anticipatory reactor SCRAM on loss of feedwater or turbine trip. On October 5, 1979, SMUD informed the Nuclear Regulatory Commission of further details for the safety grade anticipatory reactor trip.

In addition to the above described changes, SMUD is considering various other modifications as a result of the NRC Staff Report, "TMI-2 Lessons Learned Task Force Status Report and Short-term Recommendation", NUREG-0578. These modifications are described in SMUD's letters to the NRC dated October 18, 1979, November 19, 1979, and November 26, 1979.

The letters identified above, along with other documents related to the changes, describe the changes and will be provided pursuant to Licensee's Response to California

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Energy Commission's First Request for Production of Documents.

3. INTERROGATORY: For each change described in response to Interrogatory 2, provide the following additional data:

   a. A description of the purpose of the change;
   b. A schedule detailing when the change was, shall or may be instituted;
   c. A description of any constraints, including legal, regulatory, technological, or economic, which affect incorporation of the change; and
   d. An estimate of the cost of the change.

ANSWER a, b: The purpose(s) of each change and the schedule for implementation are included in the documents identified above in the answer to Interrogatory No. 2.

ANSWER c: All proposed design and procedure changes must receive sufficient development, review and evaluation to determine their adequacy and acceptability prior to implementation. Depending on the complexity of the items, this process may be protracted. NRC approval may also be required, in which case further delay may be incurred and additional proposals may be required.

Following necessary reviews and approvals, hardware procurement may take several weeks to several years. Quality assurance requirements must be met to assure that components meet the safety requirements and quality standards incorporated in the original safety analysis, and may cause delays in procurement or installation.

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Many changes, particularly those relating to the engineered safety features and reactor protection systems, necessitate a reactor shutdown to implement the change. The costs of a shutdown are large, even for a minor change. Therefore, it is economically desirable to make multiple changes at one time, usually at an annual refueling and maintenance outage when the plant is shut down for several weeks.

ANSWER d: The dominant portion of the costs for the changes identified is for professional engineering time which is not accounted for in each specific change. It is estimated, however, that total SMUD expenditures to date for these changes have been approximately $800,000.

4. INTERROGATORY: With respect to the potential small break analyses performed by SMUD in response to the NRC's Order of May 7, 1979, what operating instructions have been implemented by SMUD to define proper operator actions? Describe any additional changes in operating instructions which have been implemented since March 28, 1979, together with their rationale and dates of implementation. Identify all documents relating to the small break analyses and/or operating instruction changes.

ANSWER: The small break analysis information has been submitted to the Nuclear Regulatory Commission on May 14, July 2, August 27, and September 19, 1979. The letters transmitting these analyses describe the operating procedure changes which were made as a result of the guidelines developed from the analyses. These letters will be provided pursuant to Licensee's Response to California Energy Commission's First Request for the Production of Documents. Additional changes to operating procedures not related to

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these small break analyses have been made as described in the answers above to Interrogatories 2 and 3. See the answer above to Interrogatory 1 for additional documents related to the small break analyses.

7. <u>INTERROGATORY:</u> Describe all documents or safety analyses prepared by SMUD for combinations of failures in the safety and relief valves of the primary system.

<u>ANSWER:</u> The failure of a reactor coolant system safety and/or relief valve in effect is a small break loss of coolant accident. The small break analysis discussed above in the answer to Interrogatory No. 4 covers a spectrum of breaks which includes that represented by a stuck-open relief valve and any combination of stuck-open relief and/or safety valves. See the answers to Interrogatories 1 and 4 for the documents related to small break analyses.

8. <u>INTERROGATORY:</u> Describe the design bases of each component, including the integrated control system, which affects the reliability of the auxiliary feedwater system. Describe the degree of seismic design and protection from other potential hazards such as on-site explosions, fires, or seismic failures of nearby non-safety related equipment and structures (including the natural draft cooling towers). The availability of backup water supplies for the auxiliary feedwater system should be included in response to this interrogatory.

<u>ANSWER:</u> On September 17, 1979, SMUD provided the Nuclear Regulatory Commission with a draft version of an auxiliary feedwater system reliability analysis. This analysis will be finalized in December, 1979, at which time it will be made available for inspection and copying. In the meantime, the basic conclusions of the draft report are valid and provide the information requested by this interrogatory.

The analysis identifies contributors to the Rancho Seco auxiliary feedwater system reliability, and employs a fault-tree analysis technique, using operating reliability data for contributing components. It should be noted that the integrated control system does not affect the reliability of the auxiliary feedwater system since Rancho Seco operators are trained to control the auxiliary feedwater system independent from the integrated control system. In addition, SMUD has committed to install a control system separate from the integrated control system.

The auxiliary feedwater system is classified as a Seismic Class I system and is protected from on-site hazards such as explosions, fires, or seismic failures by location or barriers.

SMUD's letter of September 17, 1979, providing the draft reliability analysis and the commitment for a separate control system, and the Final Safety Analysis Report, will be provided pursuant to Licensee's Response to the California Energy Commission's First Request for Production of Documents.

The backup water supplies for the auxiliary feedwater system consist of both the Folsom-South Canal and the 2850 acre feet Rancho Seco Lake, which provide an essentially unlimited amount of water.

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9. INTERROGATORY: Describe the design and/or safety analyses for containment building hydrogen control, if any, which have been prepared by SMUD or which SMUD has obtained from other sources. Describe the containment building instrumentation and any anticipated changes.

ANSWER: The design and analysis of the containment building hydrogen control system is described in Appendix 14C to the Rancho Seco Final Safety Analysis Report. The containment building sampling system which would be used to measure hydrogen concentration is also described in this appendix to the FSAR. SMUD does not anticipate any changes to the instrumentation system for measuring hydrogen concentration as a result of the Three Mile Island accident.

10. INTERROGATORY: Describe each evaluation or study, if any, of controlled filtered venting which has been performed by SMUD or which SMUD has obtained from other sources.

ANSWER: The Rancho Seco design incorporates a filtered ventilation system for the reactor building. This system is described in Section 9.7 of the Final Safety Analysis Report. This system is used to filter and clean the reactor building air during normal plant operation and is not intended for use following an accident. The Nuclear Regulatory Commission's "TMI-2 Lessons Learned Task Force Final Report", NUREG-0585, recommends further study of design features for core-damage and core-melt accidents. SMUD has not performed any such evaluations or studies and is not aware of any such studies at this time.

30. INTERROGATORY: Identify each person who SMUD expects to call to testify at the hearing in this proceeding. For each person identified, provide: the subject(s) upon which the person may testify; a description of the substance of

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the testimony; and a description of the person's educational background and professional qualifications.

<u>ANSWER</u>: Robert A. Dieterich, Senior Nuclear Engineer in SMUD's Generation Engineering Department, is expected to provide testimony on contentions, identified below, relating to the design of Rancho Seco. Mr. Dieterich is a graduate of the University of Kansas with a Bachelor of Science degree in Engineering Physics, with a major area of study in nuclear engineering. He has also taken graduate level courses in nuclear engineering from the University of Washington while employed by the General Electric Company in that state.

Mr. Dieterich worked for three years as a Process Physicist for the General Electric Company at the Hanford operations in Richland, Washington. In this position he performed all routine physics calculations for an assigned plutonium production reactor, providing fuel loading patterns and operating techniques consistent with established safety criteria. Following this period, Mr. Dieterich accepted a position as a Nuclear Engineer with the General Electric Company in their Nuclear Energy Division in San Jose, California. In this position he performed analyses of design basis reactivity accidents for safety analysis reports. He also participated in the licensing efforts for the Oyster Creek and Nine Mile Point nuclear power plants, and had total responsibility for the preparation of the Nine Mile Point and Monticello technical specifications.

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Following his employment with the General Electric Company, Mr. Dieterich accepted a position with the Sacramento Municipal Utility District, where he has been employed for the last nine years. Mr. Dieterich is presently a Senior Nuclear Engineer in the Generation Engineering Department and has had responsibilities in the design, erection, startup and licensing of Rancho Seco .

Mr. Dieterich is a member of Sigma Pi Sigma (a physics honorary society) and the American Nuclear Society. Mr. Dieterich is currently registered as a Professional Nuclear Engineer in the state of California, registration number N103.

The specific contentions to which Mr. Dieterich will testify are listed below with a description of the substance of his testimony.

Issue CEC 1-6

Will the modifications of subparagraphs a-e still leave the Rancho Seco emergency feedwater system in a condition of low reliability?

Hursh-Castro Contention 7

Rancho Seco, being a Babcock and Wilcox designed reactor, has insufficient timeliness and reliability of the emergency feedwater system, and therefore is unsafe and endangers the health and safety of Petitioners, constituents of Petitioners and the public.

Description of Substance of Testimony

This testimony will demonstrate that the Rancho Seco auxiliary feedwater system is a system of high reliability and sufficient timeliness, will discuss a reliability study

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performed for that system, and will show that the modifications described in subparagraph a to e have improved that reliability. [See also Licensee's Answers (Set No. 2).]

Issue CEC 5-1

Whether those systems identified as contributing to releases of radioactivity during the TMI accident, which are outside containment, should be changed to vent into the containment building?

Description of Substance of Testimony

This testimony will show that the containment isolation system at Rancho Seco is different from that at Three Mile Island, and that the possibility of radioactivity releases from outside the containment are reduced at Rancho Seco. Therefore, radioactivity containing systems outside the containment building at Rancho Seco need not be changed to vent into the containment building.

Issue CEC 5-2

Whether the containment building should be modified to provide overpressurization protection with a controlled filtered venting system to mitigate unavoidable releases of radionuclides?

Description of Substance of Testimony

This testimony will show that the reactor containment building is designed to withstand accident pressures.

Hursh-Castro Contention 3

Rancho Seco, being a Babcock and Wilcox designed reactor, has a lack of direct initiation of reactor trip upon the occurrence of off-normal conditions in the feedwater

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system, and therefore is unsafe and endangers the health and safety of the Petitioners, constituents of Petitioners and the public.

Hursh-Castro Contention 9

Rancho Seco, being a Babcock and Wilcox designed reactor, has not installed adequate hard-wire control grade reactor trip on loss of main feedwater and/or on turbine trip, and therefore is unsafe and endangers the health and safety of Petitioners, constituents of Petitioners and the public.

Description of Substance of Testimony

This testimony will indicate that a direct initiation of reactor trip upon turbine trip or loss of feedwater has been installed at Rancho Seco.

Hursh-Castro Contention 5

Rancho Seco, being a Babcock and Wilcox designed reactor, has an actuation before reactor trip of a pilot operated relief valve on the primary system pressurizer which, if the valve sticks open, can aggravate an accident, and therefore is unsafe and endangers the health and safety of Petitioners, constituents of Petitioners and the public.

Description of Substance of Testimony

This testimony will show that the setpoints for the reactor trip on high primary system pressure and the pilot operated relief valve have been changed so that the reactor trip occurs before the pilot operated relief valve is actuated.

Hursh-Castro Contention 20

Rancho Seco, being a Babcock and Wilcox designed reactor, does not have a hydrogen recombiner which may be necessary in the event of an accident caused by a loss of feedwater transient, and therefore is unsafe and endangers the health and safety of Petitioners, constituents of Petitioners and the public.

Description of Substance of Testimony

This testimony will describe arrangements SMUD has made for the provision of a hydrogen recombiner on short notice at Rancho Seco.

FOE Contention III(c)

The NRC orders in issue do not reasonably assure adequate safety because there is no reasonable time for implementation of the long-term modifications established in the Commission orders.

Description of Substance of Testimony

This testimony will provide the schedules for the implementation of the long-term modifications established in the Commission Order. It will be shown that Rancho Seco is being operated safely at the present time and that the long-term modifications are sufficient to provide continued reasonable assurance that the facility will respond safely to feedwater transients.

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AFFIDAVIT OF ROBERT A. DIETERICH

County of Sacramento)
: SS
State of California)

Robert A. Dieterich, being duly sworn according to law, deposes and says that he is a Senior Nuclear Engineer in the Generation Engineering Department of the Sacramento Municipal Utility District; and that the answers contained in "Licensee's Answers (Set No. 1) to the First Set of Interrogatories of the California Energy Commission dated November 15, 1979" are true and correct to the best of his knowledge and belief.

Robert A. Dieterich

Robert A. Dieterich

Sworn to and subscribed before
me this 4th day of December, 1979.

OFFICIAL SEAL
PATRICIA J. CROOKS
NOTARY PUBLIC - CALIFORNIA
SACRAMENTO COUNTY
My Commission Expires June 28, 1983

____
Notary Public

My Commission expires ______________.

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