UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION 12/11/79

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

NRC STAFF RESPONSES TO CALIFORNIA ENERGY COMMISSION'S FIRST SET OF INTERROGATORIES TO THE NUCLEAR REGULATORY COMMISSION

In accordance with 10 CFR § 2.720 and 10 CFR § 2.744, the NRC Staff hereby responds to the California Energy Commission's (CEC) First Set of Interrogatories to the Nuclear Regulatory Commission dated November 15, 1979. Each CEC interrogatory not objected to is restated and a response is provided. Following the responses are affidavits identifying the individuals who prepared the responses and verifying them. To the extent that the NRC Staff objects either in whole or in part to any interrogatory posed by CEC, those objections are raised in the "NRC Staff Request for a Finding Pursuant to 10 CFR § 2.720(h)(2)(ii) Regarding California Energy Commission's First Set of Interrogatories to the Nuclear Regulatory Commission and NRC Staff Request For a Protective Order" filed contemporaneously with these responses.

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Interrogatory 1

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") reactor systems which relate, in whole or in part, to any of the following:

a. Small break loss of coolant accident;
b. Conditions of inadequate core cooling;
c. Sensitivity evaluations of delays in startup of auxiliary feedwater systems;
d. Sensitivity evaluations of steam generator design parameters such as volume and hydraulic characteristics;
e. Sensitivity evaluations of reactor trip setpoints, release in 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.

Identify all persons who have or are continuing to participate in these analyses.

Response

Enclosed are complete listings of all publicly available documents which relate in whole or in part to the Three Mile Island, Unit 2 incident and its effect on Rancho Seco, the other B&W operating plants and the B&W designed 177 Fuel Assembly Nuclear Steam Supply System.

These listings include all principal correspondence between the NRC, the B&W licensees, B&W and many other corporate and private organizations. These listings also include internal NRC Staff memoranda.

Enclosure 1 to this response provides an index to the various listings attached to this response and Enclosure 2 explains the various types of information which can be extracted from these documents. The documents identified in Enclosure 1 are provided with that Enclosure.

All of the documents listed are available in the NRC's public document room in Washington, D.C. In addition most of the documents listed under the Docket Number 50-312 (Rancho Seco) are available in the Local Public Document Room located in the Sacramento City-County Library, 828 I Street, Sacramento, California. Copies of the documents listed in the printouts are available in paper copy or microfiche.

This response was prepared by Robert Capra.

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Interrogatory 2

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 and those persons conducting analyses of the changes. Describe any potential changes which were considered but rejected and the reasons for rejecting them.

Response

The NRC does not routinely review all changes made to operating procedures. However, the following is a list of those procedures the NRC Staff is aware of that were changed or added as a resul+ of the TMI-2 accident:

No.	Title	Originator (all SMUD personnel)	Date of Latest Revision Reviewed by NRC Staff
A.46 Rev 6	Main Turbine System	N/A	N/A
A.51 Rev 5	Auxiliary Feedwater System	M. Carter	5/1/79
A.64 Rev 5	Generator and Exciter System	N/A	N/A
B.2 Rev 11	Plant Heatup and Startup	M. Carter	5/9/79
B.4 Rev 9	Plant Shutdown and Cooldown	M. Carter	N/A
D.1 Rev 5	Load Rejection	W. Ford	N/A
D.2 Rev 4	Turbine Trip	W. Ford	N/A
D.3 Rev 9	Reactor Trip	M. Carter	5/8/79
D.5 Rev 13	Loss of Reactor Coolant/Reactor Coolant System Pressure	M. Carter	N/A
D.10 Rev 6	Loss of Reactor Coolant Flow/ RCP Trip	M. Carter	6/6/79
D.11 Rev 3	Loss of Reactor Coolant Makeup/ Letdown	W. Ford	5/24/79
D.14 Rev	Loss of Steam Generator Feed	M. Carter	N/A
H2YSA Rev 2	Turbine and Secondary Systems - Panel A	M. Carter	5/8/79
SP 210.01A Rev 9	Feed Pump P-318 Surveillance Test	R. Wickert	N/A
SP 210.013 Rev 12	Feed Pump P-313 Surveillance Test	R. Wickert	5/3/79

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No.	Title	Originator (all SMUD personnel)	Date of Latest Revision Reviewed by NRC Staff
SP 214.03 Rev 10	Locked Valve List	N/A	N/A
STP 611, Rev 1	Auxiliary Feedwater Control Valve Failure Mode Test	R. Wickert	5/9/79
STP 612	Auxiliary Feedwater Flow Indicator Functional Test	R. Wickert	5/14/79
STP 627	Test Reactor Trip from Generator/ Turbine Trips and/or Loss of Main Feedwater	M. Young	N/A

N/A - Information not available.

Several copies of these procedures are held by the Staff in Bethesda, Md. However, additional revisions to these procedures were made at the Rancho Seco site as a result of the NRC Restart Team visit from May 31, 1979 to June 2, 1979.

The NRC personnel responsible for reviewing the changes were Bruce Wilson (Operator Licensing Branch), Michael Wilber (IE, Headquarters), and Phil Johnson (IE, Region V). Additional documents may be available in the personal files maintained by these individuals relating to the procedure changes.

During the review of the affected procedures numerous changes were offered by the licensee with some accepted and some rejected by the Staff. The procedures affected most were D.5, "Loss of Reactor Coolant" and B.4, "Plant Shutdown and Cooldown." After several discussions with the NRC Staff and analysts from B&W, the licensee agreed to revisions which met the B&W Guidelines and were satisfactory to the NRC Staff. All changes proposed by the Staff were adopted by the licensee.

We have not attempted to provide a listing of potential changes that were identified by individual members of the NRC Staff but never became formal Staff positions.

The above portion of this response was prepared by Bruce Wilson.

In order to comply with Commission Order of May 7, 1979 and Bulletins issued by the Office of Inspection and Enforcement, the following facility design or equipment changes have been made:

a. Installation of Clampitron Flowmeters consisting of transducers attached to the AFW piping and the output connected to a flow display computer. Also included is the necessary wiring, instrumentation, etc., to provide flow rate indication in the control room.

b. Installation of control room annunciation for all auto-start conditions of the AFW system.

c. Installation of a hard-wired control-grade reactor trip that would be actuated on loss of main feedwater and/or turbine trip.

d. Change in the set point of the pressurizer power-operated relief valve (PORV) from 2255 psi to 2450 psi.

e. Change in the high pressure reactor trip set point from 2355 psi to 2300 psi

The staff review of these changes is discussed in the enclosure of Mr. Harold R. Denton's letter to Mr. J. J. Mattimoe dated June 27, 1979. A copy of the letter and the enclosure has been made available to the California Energy Commission.

In addition, a number of design and equipment changes are planned in order to satisfy the requirements of NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," and NUREG-0585, "TMI-2 Lessons Learned Task Force Final Report". Copies of these reports have been made available to the parties to this proceeding.

The changes identified above constitute all of the changes to facility design and equipment at Rancho Seco (of which I am aware) required (or presently contemplated) by the NRC as a result of the TMI-2 accident. All changes to facility design and/or equipment proposed by the NRC to the licensee were (or will be) implemented.

The second portion of this response was prepared by Thomas M. Novak.

Interrogatory 3

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 may affect incorporation of the change; and
d. Criteria for allowing the facility to continue operation if delays are encountered in incorporating required changes.

Response

With regard to item (a) of the interrogatory, for each procedure listed in the response to Interrogatory 2 a description of the purpose of the change is given below. The changes were made primarily during the month of May, 1979 with all changes complete to the satisfaction of the NRC Staff Restart Team by June 2, 1979. With regard to items (b), (c) and (d) of the interrogatory, the Commission Order requiring the Rancho Seco facility to be shutdown was not lifted until the changes to the procedures were made and reviewed by the NRC Staff. The NRC responsibility for review of the procedures was limited to those items specified in the Commission Order, namely item (b) dealing with auxiliary feedwater and item (d) dealing with small break operating instructions.

Procedure No.	Reason for Change
A.46	Addition of Reactor Trip upon Turbine-Generator Trip Circuitry
A.51	Addition of Section 7.7, Control of Auxiliary Feedwater Independent of ICS
A.64	Same as A.46
B.2	Same as A.46

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Procedure No.	Reason for Change
B.4	Rewrote Natural Circulation Cooldown Section, added cautions required by trip circuitry changes
D.1 through D.14	To comply with B&W Small Break Guidelines and to include cautions and instructions necessitated by trip circuitry changes
H2YSA	Incorporate new reactor trips
SP 210.U1 A & B	To improve reliability of auxiliary feedwater during functional tests
SP 214.03	To improve reliability of auxiliary feedwater
STP 611	To set criteria for acceptable performance of auxiliary feedwater valves upon loss of control signal
STP 612	New procedure for functional test of auxiliary flow indicators
STP 827	New procedure for testing added trip circuitry

The first portion of this response was prepared by Bruce Wilson.

With regard to item (a) of the interrogatory for each design or equipment change listed in the response to Interrogatory 2, a description of the purpose of the change is given below. The changes were made primarily during the month of May, 1979 with all changes complete to the satisfaction of the NRC Staff Restart Team by June 2, 1979. With regard to items (b), (c), and (d) of the interrogatory, the Commission Order requiring the Rancho Seco facility to be shut down was not lifted until the changes to the equipment were made and reviewed by the NRC Staff. The NRC responsibility for review of the equipment was limited to those items specified in the Commission Order.

Design or Equipment Change	Reason for Change
AFW flow rate indication in control room	To provide operator with direct indication of AFW flow in control room.

Design or Equipment Change	Reason for Change
Annunciator for all auto-start of the AFW system	To provide additional information to the operator to identify startup of the AFW.
Control-grade reactor trip for loss of main feedwater and/or turbine trip	To provide anticipatory trips for secondary side transients thus reducing the potential for opening of the power-operated relief valve (PORV) and/or the safety valves on the pressurizer.
Change in the pressurizer power-operated relief valve (PORV) from 2255 psi to 2450 psi	Reduce the likelihood of PORV and/or safety valve opening in response to anticipated transients.
Change in the high pressure reactor trip set point from 2355 psi to 2300 psi	Reduce the likelihood of PORV and/or safety valve opening in response to anticipated transients.

With respect to NUREG-0578 and 0585, the information sought in the interrogatory is set forth in those documents, copies of which have been previously supplied to the parties to this proceeding.

The second portion of this response was prepared by Thomas M. Novak.

Interrogatory 4

Describe the potential for improving the safe operation of the facility based upon results from a failure and affects analysis of the integrated control system.

Response

A failure mode and effects analysis (FMEA) is a systematic procedure for identifying the modes of failure of a system and for evaluating their consequences. A FMEA is considered (as stated in IEEE 352-1975, "IEEE Guide for General Principles of Reliability Analysis of Nuclear Power Generating Station Protective Systems") to be the first general step of a reliability analysis. It can potentially provide some early useful information and provide a basis for later studies and/or analyses.

Typically a FMEA has been utilized as a tool to help systematically evaluate plant safety systems (such as the reactor protection and engineered safety features actuation system) to determine if a single failure can prevent the system safety function. It is a requirement that for plant safety systems no single failure shall prevent the system safety function.

Plant control systems such as the integrated control system (ICS) have typically not been required to meet this single failure criterion. However, for any system, including a control system, a FMEA can be used to identify failure modes which could lead to undesirable consequences.

B&W has performed an FMEA on the Integrated Control System (ICS) as part of its reliability analysis of the ICS. The other part of the reliability analysis is a review of the system's "Operating Experience." The FMEA and Operating Experience are documented in B&W report BAW 1564, "Integrated Control System Reliability Analysis," which has been endorsed by SMUD as applicable to Rancho Seco. (A copy of this report is Enclosure 1 to this response.)

Based on the overall reliability analysis, the report makes recommendations to be evaluated on a plant-specific basis for potential improvements to the operation of the facility. Based on these recommendations, the NRC requested (by letter of November 7, 1979, a copy of which has been previously sent to the parties to this proceeding) that all B&W licensees (including Rancho Seco) evaluate the report's recommendations and include followup action plans. We are presently awaiting the responses. In addition, Oak Ridge National Laboratory (ORNL) is reviewing the B&W report for the NRC. It is expected that the need for further studies or analyses may be identified by ORNL and in that case we will determine any further action to be required.

This response was prepared by Dale Thatcher.

Interrogatory 5

Describe any analysis which has been performed to document the acceptability of natural circulation for core cooling, including limiting conditions of low primary system water inventory. Describe all testing or operating data that have been used to verify the adequacy of the analyses performed.

Response

The concept of natural circulation has long been recognized as a means of heat removal from a nuclear reactor. This method is described by C. F. Bonilla in his textbook entitled "Nuclear Engineering". Pages 448 through 455 of that text are provided as Enclosure 1 to this response. Natural circulation is described as a simple and dependable method, and is obtained by designing the steam generators such that their thermal center is located at an elevation higher than the thermal center of the reactor core. The resulting difference in density heads between the thermal centers provides partial loop flow if the pumps fail or are tripped.

Tests to verify natural circulation have been performed as part of the normal startup program for two B&W designed reactors. One of the tests was performed at Oconee Unit No. 1 which is similar to Rancho Seco. Natural circulation has also been demonstrated in several instances when the reactor coolant pumps were inadvertently tripped. These events provide experimental confirmation of the acceptability of natural circulation, and are described in a document entitled "Appendix 1 - Natural Circulation in B&W Operating Plants," a copy of which is provided as Enclosure 2 to this response.

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A description of the analytical methods used by B&W for natural circulation calculations is provided in the May 18, 1979 letter to the NRC Staff from B&W and attached report which are provided as Enclosure 3 to this response. This reference also provides a comparison of the analysis methods to test data.

The above information relates to conditions when the primary system is essentially single phase. Natural circulation can also occur with two phase fluid (low system inventory) in the primary system; the concept of differences in the density heads between the core and steam generator thermal centers still applies. However, natural circulation under these conditions has not yet been demonstrated experimentally. The NRC Staff expects to recommend that such verification be provided by January 1981. It should be noted that natural circulation did not occur during the course of the incident at TMI-2 when the primary system contained significant voids (low water inventory). A qualitative explanation of this event is provided in Appendix A to NUREG-0623 entitled "Generic Assessment of Delayed Reactor Coolant Pump Trip During Small Break of Loss-of-Coolant Accidents in Pressurized Water Reactors." A copy of the Appendix is provided as Enclosure 4 to this response. This discussion indicates that natural circulation in lowered-loop plants can be maintained provided sufficient inventory exists to raise the liquid level in the steam generators above that of the bottom of the pump discharge nozzle (required to clear the loop seal).

The reactor core at TMI-2 was cooled by the flow provided by a single reactor coolant pump for a period of approximately one month following the accident. In early May 1979, the pump was tripped, and natural circulation was quickly established. The system has been operating under natural circulation (single phase) since that time. The NRC Staff evaluation of potential methods to establish natural circulation at TMI-2 is provided in Chapters 3 and 4 of NUREG-0557 entitled "Evaluation of Long-Term Post-Accident Core Cooling of Three Mile Island Unit 2" which is attached to this response as Enclosure 5. An analysis method for the evaluation of natural circulation with a heavily damaged core region is also provided in the report attached to the June 7, 1979 Memorandum from S. Fabic. Copies of these documents are attached as Enclosure 6 to this response. The analysis methods described also apply to undamaged cores.

An additional document, Technical Memorandum NRC-102-1, Natural Circulation in the Primary Loop at TMI During Long Term Cooling, is provided as Enclosure 7.

This response was prepared by Paul Norian.

Interrogatory 6

Describe any feedwater transients occurring in any licensed United States nuclear power reactor since the TMI incident. In your answer, describe any responses made by NRC or the licensee, any safety precautions which have subsequently been instituted to assure safe responses to such transients, and any documents relating to such transients.

Response

In response to this interrogatory, it must be noted that all feedwater transients occurring in licensed nuclear power plants are not in and of themselves events which must be reported to the NRC. Regulatory Guide 1.16 ("Reporting of Operating Information - Appendix A Technical Specifications") provides guidance to licensees on which types of events must be reported to the NRC as a "reportable occurrence". A reportable occurrence is defined as any unscheduled or unanticipated operational event which may be significant from the standpoint of public health or safety. If a reportable occurrence takes place at any licensed facility, it must be reported to the NRC through a licensee event report (LER). Since the vast majority of feedwater transients do not get reported as LERs, details of such events are not avilable to send you. However, all unit shutdowns and significant reductions in power, whether scheduled or unscheduled, are required to be reported by each licensee in its "monthly operating report". These are tabulated and published monthly in a document entitled "Operating Units Status Report - Licensed Operating Reactors," (NUREG-0020, commonly referred to as "the gray book"). Therefore, if a feedwater transient caused a reactor trip, it would be reported to the NRC, along with all other outages, in the monthly operating report and subsequently tabulated in the gray book. The gray book is a publicly available document which is on file at the Public Document Room in Washington, D.C. and at the Local Public Document Room in Sacramento, California. (The Rancho Seco Local Public Document Room only keeps the latest edition of the gray book).

Enclosed with this response is a computer printout (Enclosure 1) of all forced outages during the months of April through September 1979 which has been extracted from the gray books. The transients which either directly or indirectly appear to be feedwater induced reactor trips have been underlined.

A more detailed summary of the feedwater transients on the B&W operating plants can be found in the "Summary of Meeting Held on August 23, 1979, with the Babcock & Wilcox (B&W) Operating Plant Licensees to Discuss Recent (Post TMI-2) Feedwater Transients," dated September 13, 1979 (Enclosure 2). Three additional feedwater transients which have occurred at B&W operating plants are summarized in Enclosures 3 through 5, describing transients at Rancho Seco (4/22/79), Oconee Unit 3 (11/10/79), and Crystal River 3 (9/15/79), respectively.

The NRC and all operating plant licensees have taken a great number of actions, subsequent to the TMI-2 incident, to mitigate the consequences of feedwater transients. However, no additional steps or safety precautions, other than those already completed or scheduled to be completed, have been issued as a direct result of any of the feedwater transients listed in the enclosures.

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This response was prepared by Robert Capra.

Interrogatory 7

Describe the methods which are available to purge the accumulation of gases in the primary system and describe NRC's assessment of the acceptability of each such method discussed. Identify any documents related to these methods.

Response

This response presents the two methods which are currently available at Ranch Seco for removing gases in the primary system. We are providing documents which discuss these methods. Other documents may exist which discuss the utilization of these methods at TMI-2. Those documents are not included in this response.

A discussion of the potential sources and effects of non-condensible gases for the Rancho Seco reactor is provided in Enclosure 1, a letter dated November 5, 1979 from the licensee to the NRC Staff, with attachment. This reference states that accumulation of significant quantities of noncondensible gases within the primary system is not expected during a small break loss of coolant accident providing the core remains covered. The Staff agrees with this assessment. For larger breaks (greater than approximately a two-inch diameter pipe), heat transfer to the secondary system is not required since the core decay energy can be removed through the break. Consequently, the introduction of significant quantities of noncondensible gas (e.g., from the core flooding tanks) will have no significant effect on the performance of the emergency core cooling systems (ECCS) because natural circulation cooling is not required.

Two methods that are currently available to purge the accumulation of gases in the primary system are also discussed in Enclosure 1 (page 6). These include: (1) venting the pressurizer steam - gas space to the Quench Tank, and (2) utilizing the Letdown System to degas via the Makeup Tank. The use of the Quench Tank directly removes the gas from the pressurizer vapor region where it tends to collect. The use of the Letdown System enables the high pressure primary system fluid, containing dissolved noncondensible gases, to be transferred to the low pressure Makeup Tank. The solubility of gas in the primary system fluid is lower at the reduced pressure, and the dissolved gas is stripped in the tank. The degassed fluid is then returned to the primary system. Both of these methods are standard techniques for removing gases and are acceptable to the Staff.

The NRC has also required that each PWR provide remotely operable high point vents to remove gas in the primary system. See Enclosure 2 to this response, letter of September 13, 1979 to all nuclear power plant licensees from D. G. Eisenhut with enclosures, especially its Enclosure 2. The required implementation date for the venting system is January 1, 1981. The installation of the high point vents will provide an additional method to purge gases from the primary system.

This response was prepared by Paul Norian.

Interrogatory 8

Describe the emergency procedures and/or training which have been or are being instituted or are being studied, contemplated or proposed to improve operator performance during small break loss-of-coolant accidents. Identify all documents related to such procedures and training.

Response

Rancho Seco emergency procedure D.5, "Loss of Reactor Coolant/Reactor Coolant System Pressure" was revised to incorporate the B&W Small Break Guidelines. The significant revisions to this procedure were intended to assure that a TMI-2 type incident does not happen at Rancho Seco by recognizing that a small break may be due to a stuck open power operated relief valve and criteria for turning off or throttling the high pressure injection (HPI) system. The operators are not allowed to terminate HPI unless both low pressure injection pumps are in operation and producing a flow rate in excess of 1000 gpm each and the situation has been stable for 20 minutes or the primary coolant system is 50° subcooled. The revisions to procedure D.5 were reviewed on-site by the NRC Restart Team, principally Bruce Wilson. These revisions were judged to conform with the approved B&W Small Break Guidelines.

The training that the licensed personnel have received is as follows:

1. TMI-2 sequence training on the B&W Simulator;
2. Lectures administered by the Rancho Seco staff ;
3. Lectures by General Physics, Corp.;

4. One week requalification training on the B&W simulator (18 people);

5. Incorporation of the TMI-2 incident and lessons learned in the Rancho Seco hot license and requalification programs.

A legal size folder is in the possession of Mr. Wilson that contains various typed and hand-written documents related to procedure revisions and training of Rancho Seco personnel. Telephone conversations regarding the training of Rancho Seco personnel were made between Mr. Wilson and Mr. Norman Elliott, Manager of B&W training department, on October 30, 1979 and between Mr. Wilson and Mr. Jack Mau, Rancho Seco training coordinator on about November 1, 1979. Notes of these conversations are contained in the folder.

The Licensee's up-to-date procedures would, of course, contain all of the revisions which have been instituted. The Staff does not have a complete up-to-date set of these procedures, but they can be obtained from the Licensee.

This response was prepared by Bruce Wilson.

Interrogatory 9

What systems outside of the containment released radioactivity in the atmosphere during the TMI incident?

Response

From the information available today, the systems and components outside of the containment which released radioactivity in the atmosphere during the TMI incident include the waste gas vent header and compressors in the waste gas system, the reactor coolant bleed holdup tank relief valve in the letdown system, the fuel handling and auxiliary building sump tanks, the radwaste system pumps which took suction from the reactor coolant bleed tanks, and the valves and instruments in the reactor coolant makeup and purification system. The source of this information is the NRC report NUREG-0600, "Investigation into the March 28, 1979, Three Mile Island Accident by the Office of Inspection and Enforcement," specifically sections 3.1 and 3.2 beginning at p. II-3-1. A copy of that document has been previously sent to you.

This response was prepared by James Wing.

Interrogatory 10

What plant areas outside the containment were contaminated by radioactivity during the TMI accident sufficient to restrict or control personnel access to those areas?

Response

During the early days of the incident, access to all the areas beyond the security building was periodically restricted because it was thought that these areas were contaminated by airborne radioactivity. It was determined later that only the following plant areas outside the containment were contaminated by radioactivity during the TMI incident sufficient to restrict or control personnel access: the auxiliary buildings of Units 1 and 2, the fuel handling buildings of Units 1 and 2, the basement of the diesel generator building of Unit 2, small areas in the basement of the control and service building of Unit 2, the area between the control and service building and the containment building of Unit 2, the area between the auxiliary building and the containment of Unit 2, and the area between the fuel handling building and the containment of Unit 2.

This response was prepared by James Wing.

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Interrogatory 11

Describe all evaluations or studies of controlled filtered venting which have been performed by NRC or are in NRC's possession.

Response

The term "controlled filtered venting" implies a process in which the containment atmosphere is deliberately released to the environment through filters. In nuclear power plants, there are various systems that chould be considered to be "controlled filtered venting" systems - these systems are reviewed and evaluated during the licensing process. Analyses of these systems are performed for each plant and can be found in the various safety analysis reports (including the Rancho Seco FSAR) for these facilities.

There are fewer studies evaluating controlled filtered venting of containments in a core-melt scenario. For the core-melt accident, there are two studies of which I am aware which have been performed for the NRC: 1) "Conceptual Design FFTF Containment Margins," and 2) "Program Plan for the Investigation of Vent-Filtered Containment Conceptual Design for Light Water Reactors." Copies of these studies are Enclosures 1 and 2 to this response.

Enclosure 1 is a study performed for the Fast Flux Test Facility (FFTF), a sodium-cooled, fast spectrum experimental reactor owned by the Federal Government. The report was prepared for the NRC Staff to follow up on a recommendation made by the NRC Staff that containment margins be augmented by a controlled containment vent and by provisions for hydrogen detection and control. This recommendation had been supported by the ACRS. Enclosure 1 gives the criteria used as a basis for the design of the Containment Margins Systems and describes the design concepts that have been selected in detail.

The NRC Staff evaluation, advice, and recommendations regarding the adequacy of containment margins for postulated core melt down events for the FFTF are provided in sections 1.9, 15.3.6, 15.3.7, 19.3 and Appendices C, D, and F of the Safety Evaluation Report for the Fast Flux Test Facility (NUREG-0358). Addition information regarding the containment adequacy for core melt events for the FFTF are provided in sections 1.8(2), 15.3.6, 15.3.7, 19.3B and Appendices C, D, E, F, G and H of Supplement 1 to the Safety Evaluation Report for the Fast Flux Test Facility (NUREG-0358). Copies of these documents are provided as Enclosures 3 and 4 to this response.

Enclosure 2 is the first of a series of reports that will be prepared for the NRC by Sandia Laboratories. In April, 1978, the NRC submitted to Congress a plan outlining seven key areas of research to be conducted over a 3-year period. Of the various research projects proposed, a program for the development and analysis of vent-filtered containment conceptual design was accorded particularly high priority. Sandia Laboratories was awarded the contract to perform this study. This report addresses the major issues involved in accident mitigation systems and discusses the engineering, technical and economic questions that will have to be studied before judgments can be made regarding feasibility and effectiveness of such a system.

The NRC also has a Technical Library in which there are reports and studies on controlled filtered venting of containments which have been published by other organizations and are available to the public. They include, but are not limited to, such publications as: Nuclear Power Issues and Choices sponsored by the Ford Foundation and administered by the MITRE Corporation; Evaluation of the Feasibility, Economic Impact, and Effectiveness of Underground Nuclear Power Plants, prepared for the California Energy Commission by the Aerospace Corporation; and Conceptual Contained Nuclear Power Plants, a study for California Energy Commission by Sargent and Lundy Engineers.

This response was prepared by Thomas A. Greene.

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Interrogatory 12

Describe any special features and/or improved instrumentation which can be added to the Rancho Seco control room to improve operator comprehension and performance during off-normal conditions including those to diagnose conditions caused by inappropriate operator actions or unexpected system status.

Response

The NRC has established certain requirements for installation of instrumentation to improve operator comprehension and performance during off-normal conditions. The NRC is also requiring licensees to undertake certain studies to determine what improvements in other instrumentation can be made. These requirements are set forth in NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations", and NUREG-0585, "TMI-2 Lessons Learned Task Force Final Report". Copies of both reports have been previously provided to the parties to this proceeding.

In NUREG-0578, the relevant requirements are:

2.1.3A Direct Indication of Power-Operated Relief Valve and Safety Valve Position;
2.1.3b Instrumentation for Detection of Inadequate Core Cooling in PWRs and BWRs;
2.1.7b Auxiliary Feedwater Flow Indication to Steam Generator for PWRs.

In NUREG-0585, the relevant requirements are:

5. Verification of Correct Performance of Operating Activities;
7. Man-Machine Interface.

Other possible improvements in instrumentation may have been identified by organizations other than the NRC and could be found in publications of those organizations.

This response has been prepared by Dale Thatcher.

Interrogatory 13

Describe the studies, if any, which have been prepared or funded by NRC relating to the facility or to B&W reactor systems concerning control room design, human engineering factors, or improved operator training methods.

Response

NRC has prepared or funded the following studies concerning control room design, human engineering factors or improved operator training methods.

NRC funded a study (Enclosure 1) by the Aerospace Corporation to review the characteristics of reactor operators, the tasks requested of them, the aids in performing those tasks and the features of control rooms. The report recommended areas in which NRC might provide additional guidance intended to improve the human factors aspects of nuclear reactor design and operation.

NRC funded a study (Enclosure 2) by Sandia Laboratories to review the human factors aspects of the Zion Nuclear Power Plant. The report recommended ways in which the likelihood of inappropriate operator action might be reduced and suggested possible follow-up studies.

NRC funded a study ("Criteria for Safety-Related Nuclear Plant Operator Actions: A Preliminary Assessment of Available Data," NUREG/CR-0901, July 1979, we are in the process of locating a copy of this document to make available to CEC) by Oak Ridge National Laboratory to review a standard on criteria for safety-related operator actions during postulated accidents.

The report recommended developing interim criteria based on best available information and outlined a procedure for obtaining the data necessary to establish more definitve criteria.

NRC is currently preparing or funding the following studies.

NRC is funding a study by Sandia Laboratories and other national laboratories to review and evaluate nuclear and nonnuclear operating experience and to develop quantitative probabilistic models of human performance for use in risk analyses. The findings, together with a description of factors shaping human performance, will be published in a handbook.

NRC is funding a study (Enclosure 3) by the Essex Corporation to review the design of the control room at Three Mile Island-2 and determine to what extent its design contributed to the course of events during the accident in March 1979. The findings will be reported in the report by the NRC/TMI Special Inquiry. Essex is also developing guidelines for the review of control rooms of all other operating reactors. The Staff recommended (Recommendation 7.1 in NUREG-0585) that all licensees be required to complete a one-year review of their control rooms and procedures by January 1, 1981.

NRC is funding or is likely to fund additional research studies related to enhancing the capability of reactor operators. A brief description of each of these studies follows.

Studies are performed to develop and analyze systematically the requirements for plant monitoring. The starting point is the definition and description of accident sequences having a high probability of leading to core damage. These efforts supplement activities by the staff to develop and implement positions related to status monitoring (e.g., Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident;" Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems;" definition of plant safety status vector; and capabilities at on- and off-site technical support centers).

Studies are performed to identify and evaluate the validity of pertinent methodologies used in computerized diagnostic systems. The findings help the staff determine the need for and nature of requirements for such systems. Other operational aids are examined.

Studies are performed to improve the ways that data are collected, stored and presented to the operator. Risk insights and human engineering principles are being used to generate improved video formats and other displays of safety related information. Recommendations regarding better utilization of computer technology, graphic and audio display are being generated.

Studies are being performed to indicate means by which reactor simulators might be used more effectively to train operators. Important accident sequences are reviewed to identify those combinations of equipment failures and operator errors which should be reproducible by simulators. Advanced codes are used to calculate the physical response of plant systems during these conditions to assure that the simulators properly represent these responses.

A study will be performed (a copy of the "Statement of Work" is Enclosure 4 to this response) in which current requirements for operator licensing are reviewed and recommendations for improvements in those requirements are generated.

This portion of the response to Interrogatory 13 was prepared by Raymond DiSalvo.

To the best of my knowledge, the status of any studies prepared or funded by the NRC relating to improved operator training methods (additional to those identified by Dr. DiSalvo) is as follows:

In response to a request by the Commission, the Operator Licensing Branch conducted a detailed review of the Operator Licensing Program. A report, SECY 79-330E, dated July 30, 1979, was prepared and sent to the Commission summarizing the results of the review and the recommendations. A copy of this report is Enclosure 5 to this response.

The NRC Staff also made recommendations in the area of improved operator training in the Lessons Learned reports, NUREG-0578 and 0585, which are more fully identified in the response to Interrogatory 12.

In NUREG-0578, a copy of which has been previously provided to the parties to this proceeding, recommendations for improvements in what might broadly be called "operator training methods" were made in the following areas:

- 2.2.1.a Management directions and plant procedures that define and assign responsibility for safe operation of the plant to the shift supervisor.
- 2.2.1.b An on-shift Technical Advisor with a Bachelor of Science degree, or equivalent, and specific training in response and analysis of the plant for transients and accidents.
- 2.2.1.c Shift and relief turnover procedures.
- 2.2.2.a Procedures that limit access to the control room and that establish a clear line of authority in the control room in the event of an emergency.

The Long-Term Recommendations of the Lessons Learned Task Force (NUREG-0585), a copy of whic'. has been previously provided to the parties to this proceeding, concerning or relating to improved operator training methods are contained in section 1 to Appendix A of NUREG-0585. The specific subsections of Section 1 recommend improvement in the following areas:

1. Involvement of corporate management in the selection, training, and qualifications of operations personnel.
2. Training programs for all operations personnel.
3. Conduct of in-plant drills.
4. Operator Licensing Program.
5. NRC Staff Coordination.
6. Licensed Operator Qualifications.
7. Licensee Technical and Management Support.
8. Licensing of Additional Operating Personnel.

Revisions are presently being made by the American Nuclear Society, ANS-3, Subcommittee to the "Standard for Qualification and Training of Personnel for Nuclear Power Plants." Discussions concerning these revisions have been held between the NRC and the ANS-3 Subcommittee in November and early December, 1979. Therefore, the final version of the Standard is unknown at this time. However, most of the recommendations of Lessons Learned and SECY-79-330E are expected to be incorporated in the Standard. ANS-3.5, "Nuclear Power Plant Simulator for use in Operator Training," is presently being revised by a Working Group of the ANS-2 Subcommittee. Concurrently, revised NRC Standards are being developed on the matters being addressed by the two Subcommittees.

This portion of the response to Interrogotory 13 was prepared by Bruce Wilson.

Interrogatory 15

Describe all studies, evaluations, and/or tests which have been prepared or funded by NRC subsequent to the TMI incident which analyze operator training or operators' ability to respond safely to feedwater transients either at the facility or at other B&W reactor facilities.

Response

See response to Interrogatory 13 for studies and evaluations which analyze operator training.

Routine operator licensing examinations have been given at only two operating B&W facilities since the TMI-2 accident. The NRC administered written and oral examinations at the Oconee Nuclear Station in August, 1979. These were routine Operator License examinations given to 9 applicants. One Senior Reactor Operator Examination was administered at Rancho Seco during the week of November 26, 1979.

Additionally, of course, the Staff conducted oral audits of licensed personnel at each of the B&W operating plants before approving resumption of operation of that facility. The nature of the audits administered at Rancho Seco and the Staff's conclusions are set forth in the "Evaluation of Licensee's Compliance with the NRC Order Dated May 7, 1979", dated June 27, 1979, a copy of which was previously sent to all parties to this proceeding.

This response was prepared by Bruce Wilson.

Interrogatory 16

Identify those Staff persons who prepared the conditions of the May 7, 1979 confirmatory order. Briefly describe what meetings which [sic] took place with SMUD or others to arrive at the conditions specified.

Response

The conditions set forth in the May 7, 1979 Commission Order were arrived at through interactions of various members of the NRC Staff with the Commission, representatives of B&W, and representatives of SMUD. Enclosure 1 to this response identifies NRC personnel involved. The chronology of the development of the conditions of the Order is as follows:

1. A meeting was held in Bethesda on April 18, 1979 that included the NRC Staff and representatives from B&W. A list of the NRC personnel in attendance is given in the meeting summary provided as Enclosure 2 to this response. The subjects of this meeting were natural circulation capabilities of B&W reactor plants and other related concerns raised by C. Michelson (an engineer with the Tennessee Valley Authority and consultant to the ACRS), the NRC Staff, and the ACRS. As a result of this meeting, the Staff accepted the B&W recommendations that the high pressure reactor trip setpoint should be reduced and the PORV actuation setpoint should be raised on all operating B&W plants.

2. A Commission briefing was held in Washington on April 23, 1979 to discuss the status of operating plants, particularly those of B&W design. The members of the Staff attending the briefing are shown in the briefing

transcript provided as Enclosure 3 to this response. The Staff identified to the Commission at this time the design aspects of the B&W plants that provide a greater sensitivity than plants of other design to off-normal conditions in the feedwater system. Conditions set forth in the recently issued IE Bulletins (79-05A and 79-05B) were discussed, as well as a transient that had occurred at the Rancho Seco facility the day before. The recommendation was presented to the Commission that the B&W operating plants should be shutdown pending completion of certain modifications. At the Staff's request action on this recommendation was deferred, however, until the Staff had the opportunity to determine what action the B&W licensees would take on their own initiative.

3. A meeting was held in Bethesda on April 24, 1979 that included members of the NRC Staff and representatives from various B&W reactor licensees, including SMUD. A list of attendees is given in the meeting summary provided as Enclosure 4 to this response. The topics of the meeting included the types and frequency of challenging transients in the feedwater system, the role of the ICS, the thermal-hydraulic behavior of the primary and secondary systems, the mitigation of challenging transients, and remedial measures. No commitments were received from the licensees at this meeting.

4. A Commission briefing was held in Washington on April 25, 1979 to discuss the contents of the document "NRR Status Report on Feedwater Transients in B&W Plants" dated April 25, 1979 provided as Enclosure 5 to this response. The principal authors of this report were D. Ross, R. Tedesco, and S. Hanauer.

Attendees at the briefing are listed in the briefing transcript provided as Enclosure 6 to this response. At this session, the Staff presented to the Commission the types of short-term actions that could be taken with respect to operating B&W reactor plants. This briefing was a continuation of the one that was held on April 23, 1979. At the conclusion of the briefing, it was decided that the Commission and the Staff would further consider the views that had been presented, and that another briefing would be held in which a decision would be made concerning the necessity to shut down the B&W operating plants.

5. On April 26 and 27, 1979, meetings were held in the Bethesda office of H. Denton with representatives of B&W reactor licensees, including SMUD. As a result of this meeting, SMUD (and the other B&W licensees, as more specifically delineated in each of their letters) agreed to shut down the Rancho Seco facility until certain modifications in equipment, procedures, and training were completed. This letter, a copy of which is attached as Enclosure 8, was signed on April 27, 1979 and provided the basis for the required actions set forth in the May 7, 1979 Commission Order.

6. A Commission briefing was held in Washington on April 27, 1979 to discuss the commitment letters received from the various utilities. The substance of these letters was clarified to the Commission, and the Commission concluded the briefing by directing the Staff to prepare confirmatory orders to formalize the agreements reached with the utilities. A list of NRC Staff members present at the briefing is provided in the briefing transcript provided as Enclosure 7 to this response.

7. Immediately following the Commission briefing on April 27, 1979, the Staff began drafting the confirmatory Order to SMUD. This effort continued through the weekend with a final draft being forwarded to the Commission the following week, a copy of the final Order is Enclosure 9 to this response. The principal NRC personnel involved in the preparation of the draft were: R. Reid and D. Garner from ONRR; G. Cunningham, J. Murray, J. Scinto, and T. Engelhardt from OELD; and L. Slaggie from OGC.

This response was prepared by Daniel Garner and Robert Capra.

1591 296

Interrogatory 17

Identify each person who NRC 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 the testimony; and a description of the person's educational background and professional qualifications.

Response

Listed below is each contention admitted in the proceeding. The person(s) presently expected to testify for the NRC Staff on each contention is identified, however, the designation of witnesses could change. Attached are statements of professional qualifications for these witnesses.

Contentions or Issues	Witness(es)
Licensing Board Questions	
CEC 1-2	Paul Norian
CEC 1-4	Norian
CEC 1-6	Philip Matthews
CEC 1-7	Bruce Wilson
CEC 1-10	Norian
CEC 5-3a	Wilson
California Energy Commission	Witness(es)
1-1	Mark Rubin
1-12	Rubin
3-1	Wilson
3-2	Wilson
3-3	Wilson
5-1	James Wing
5-2	Thomas Greene
Castro-Hursh	
2	Rubin
3	Dale Thatcher
4	Thatcher
5	Norian
6	Norian
7	Matthews

Castro-Hursh

8	Wilson
9	Thatcher
10	Wilson, Norian
16	Thatcher
20	Greene
21	Matthews
22	Norian
24	Norian
25	Thatcher
26	Rubin
29	Wilson
30	Thomas Novak
31	Wilson
32	Wilson, Daniel Garner, Frederick Allenspach, Allen Johnson
34	Philip Morrill

Friends of the Earth

IIIa	Rubin
IIIc	Robert Capra
IIId	Garner, Allenspach, Wilson, Johnson
IIIe	Wilson

Additionally, the Staff may call Mr. Novak, who is serving as technical coordinator for this proceeding, as an additional witness on any of the contentions or issues.

With regard to the substance of Staff testimony, the Staff will testify that the actions taken pursuant to the Commission's Order of May 7, 1979 provide reasonable assurance that the facility will respond safely to feedwater transients pending completion of the long-term requirements of the Order. The Staff testimony will respond to each specific issue or contention in the context of the actions taken by the Licensee pursuant to the May 7, 1979

Order. Staff testimony has not yet been completed but will be served upon all parties on the schedule established by the Licensing Board.

This response was prepared by Stephen H. Lewis.

Respectfully submitted,

for Stephen H. Lewis
Richard K. Hoefling
Counsel for NRC Staff

Stephen H. Lewis
Stephen H. Lewis
Counsel for NRC Staff

Dated at Bethesda, Maryland
this 11th day of December, 1979.

1591 299

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

AFFIDAVIT OF STEPHEN H. LEWIS

Stephen H. Lewis deposes and says under oath as follows:

1. I am an attorney in good standing admitted to practice before the courts of the Commonwealth of Massachusetts and the District of Columbia. I hold the position of Staff Attorney in the Office of the Executive Legal Director of the Nuclear Regulatory Commission and serve as Staff Counsel in this proceeding.

2. The answer to the California Energy Commission Interrogatory 17 was prepared by me. I hereby certify that the answers given by me are true and accurate to the best of my knowledge.

Stephen H. Lewis
Stephen H. Lewis

Subscribed and sworn to before
me this 11th day of December, 1979.

Elizabeth Ann Tipton
Notary Public

My Commission Expires: July 1, 1982

1591 300

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

AFFIDAVIT OF DANIEL J. GARNER

Daniel J. Garner deposes and says under oath as follows:

1. I am a Project Manager in the Nuclear Regulatory Commission Staff's Operating Reactors Branch 4. I am responsible for the overall coordination of licensing actions as they apply to the operating license of the Rancho Seco Nuclear Generating Station. My professional qualifications are attached to the NRC Staff response to California Energy Commission Interrogatory 17 filed in this proceeding.

2. The answer to the California Energy Commission Interrogatory 16 was partially prepared by me. I hereby certify that the answers given are true and accurate to the best of my knowledge.

Daniel J Garner
Daniel J. Garner

Subscribed and sworn to before me this 11th day of December, 1979.

1591 301

Elizabeth Ann Tipton
Notary Public

My Commission Expires: July 1, 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 (SP)
	)	
(Rancho Seco Nuclear Generating Station)	)	

AFFIDAVIT OF RAYMOND DISALVO

Raymond DiSalvo deposes and says under oath as follows:

1. I am a risk assessment engineer in the Nuclear Regulatory Commission Staff's Office of Nuclear Regulatory Research.

2. The answer to the California Energy Commission Interrogatory 13 was partially prepared by me. I hereby certify that the answers given by me are true and accurate to the best of my knowledge.

Raymond DiSalvo

Raymond DiSalvo

Subscribed and sworn to before me this 11th day of December, 1979.

Elizabeth Ann Tipton

Notary Public

My Commission Expires: July 1, 1982

1591 302

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

AFFIDAVIT OF ROBERT A. CAPRA

Robert A. Capra deposes and says under oath as follows:

1. I am a Project Manager (Nuclear Engineer) in the Nuclear Regulatory Commission Staff's Project Management Group of the Bulletins and Orders Task Force. I am responsible for coordinating the review and evaluation of actions taken by the Babcock & Wilcox operating plant licensees in response to the Three Mile Island Unit 2 accident-related IE Bulletins and Commission Orders. My professional qualifications are attached to the NRC Staff response to California Energy Commission Interrogatory 17 filed in this proceeding.

2. The answers to California Energy Commission Interrogatories 1, 6, and 16 were prepared by me or under my supervision. I hereby certify that the answers given are true and accurate to the best of my knowledge.

Robert A. Capra
Robert A. Capra

Subscribed and sworn to before me this 7 day of December, 1979

[illegible]
Notary Public

My Commission Expires: July 1, 1982

1591 303

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

AFFIDAVIT OF DALE F. THATCHER

Dale F. Thatcher deposes and says under oath as follows:

1. I am a reactor engineer in the Nuclear Regulatory Commission Staff's Instrumentation and Control Systems Branch. I am currently responsible for the review and evaluation of the instrumentation and control systems as part of the Bulletins and Orders Task Force. My professional qualifications are attached to the NRC Staff response to California Energy Commission Interrogatory 17 filed in this proceeding.

2. The answers to the California Energy Commission Interrogatories 4 and 12 were prepared by me or under my supervision. I hereby certify that the answers given by me are true and accurate to the best of my knowledge.

Dale F. Thatcher
Dale F. Thatcher

Subscribed and sworn to before me this 11th day of December, 1979.

Elizabeth Ann Tipton
Notary Public

My Commission Expires: July 1, 1982

1591 304

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

AFFIDAVIT OF PAUL E. NORIAN

Paul E. Norian deposes and says under oath as follows:

1. I am the Alternate Group Leader of the Analysis Group, Bulletins and Orders Task Force. I coordinate the reviews of small break loss-of-coolant accidents (LOCA) and transient analyses submitted by vendor owner's groups since the Three Mile Island Accident. My professional qualifications are attached to the NRC Staff response to California Energy Commission Interrogatories 5 and 7 filed in this proceeding.

2. The answers to the California Energy Commission Interrogatories 5 and 7 were prepared by me. I hereby certify that the answers given by me are true and accurate to the best of my knowledge.

Paul E. Norian
Paul E. Norian

Subscribed and sworn to before
me this 17th day of December, 1979.

Elizabeth Ann Tipton
Notary Public
My Commission Expires: July 1, 1982

1591 305

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

AFFIDAVIT OF JAMES WING

James Wing deposes and says under oath as follows:

1. I am a senior nuclear engineer in the Nuclear Regulatory Commission Staff's Effluent Treatment Systems Branch. I am responsible for the review and evaluation of the radioactive waste treatment and effluent control systems described in the Final Safety Analysis Report for the Rancho Seco Nuclear Generating Station. My professional qualifications are attached to the NRC Staff response to California Energy Commission Interrogatory 17 in this proceeding.

2. The answers to the California Energy Commission Interrogatories 9 and 10 were prepared by me. I hereby certify that the answers given by me are true and accurate to the best of my knowledge.

James Wing
James Wing

Subscribed and sworn to before
me this 11th day of December, 1979.
Elizabeth Ann Tipton
Notary Public
My Commission Expires: July 1, 1979

1591 306

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

AFFIDAVIT OF THOMAS A. GREENE

Thomas A. Greene deposes and says under oath as follows:

1. I am an Senior System Engineer in the Nuclear Regulatory Commission Staff's Containment Systems Branch. I am responsible for the review and evaluation of the Containment and Secondary Containment Functional Design, Containment Subcompartment Analysis, Containment Heat Removal Systems, Containment Isolation System, Combustible Gas Control in Containment, and Containment Leak Testing Program as described in the Safety Analysis Report for Nuclear Power Plants. My professional qualifications are attached to the NRC response to California Energy Commission Interrogatory 17 filed in this proceeding.

2. The answer to California Energy Commission Interrogatory 11 was prepared by me. I hereby certify that the answer given by me is true and accurate to the best of my knowledge.

Thomas A. Greene
Thomas A. Greene

Subscribed and sworn to before
me this 11th day of December, 1979.

Elizabeth Ann Tipton
Notary Public

My Commission Expires: July 1, 1982

1591 307

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 (SP)
	)	
(Rancho Seco Nuclear Generating Station)	)	

AFFIDAVIT OF BRUCE A. WILSON

Bruce A. Wilson deposes and says under oath as follows:

1. I am a reactor engineer in the Nuclear Regulatory Commission Staff's Operator Licensing Branch. I am responsible for the preparation and administration of written, oral, and practical exams for operators' and senior operators' licenses at production and utilization facilities. Since May, 1979, I have been assigned to the Systems Group, Bulletins and Orders Task Force. My professional qualifications are attached to the NRC Staff response to California Energy Commission Interrogatory 17 filed in this proceeding.

2. The answers to the California Energy Commission Interrogatories 8, 15 and parts of 2, 3 and 13 were prepared by me. I hereby certify that the answers given by me are true and accurate to the best of my knowledge.

Bruce A. Wilson

Bruce A. Wilson

Subscribed and sworn to before
me this 11th day of December, 1979.

Elizabeth Ann Lipton

Notary Public
My Commission Expires: July 1, 1979

1591 308

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 (SP)
	)	
(Rancho Seco Nuclear Generating Station)	)	

AFFIDAVIT OF THOMAS M. NOVAK

Thomas M. Novak deposes and says under oath as follows:

1. I am a branch chief in the Nuclear Regulatory Commission Staff's Division of Systems Safety. The branch that I supervise is responsible for the review and evaluation of a variety of safety systems in addition to the review and evaluation of a large number of transients and accidents described in Chapter 15 of applicants' Safety Analysis Reports. Currently, I am assigned to the Staff's Bulletins and Orders Task Force serving as Deputy Director. My professional qualifications are attached to the NRC Staff response to California Energy Commission Interrogatories filed in this proceeding.

2. Identified portions of the responses to California Energy Commission Interrogatories 2 and 3 were prepared by me. I hereby certify that the answers given are true and accurate to the best of my knowledge.

Thomas M. Novak

Thomas M. Novak

Subscribed and sworn to before me this 11th day of December, 1979.

Elizabeth Ann Tipton
Notary Public

My Commission Expires: July 1, 1982

1591 309

Professional Qualifications

of

RAYMOND DISALVO

I am a risk assessment engineer in the Probabilistic Analysis Staff, Office of Nuclear Regulatory Research. I am responsible for developing, managing and applying the results of research programs related to the safety of nuclear fuel cycle facilities. My principal area of current involvement is research leading to improvements in reactor safety. This includes improvements in plant safety features and in the operator-machine interface.

I received a Bachelor of Arts Degree in chemistry from Rutgers University in 1967 and a Doctor of Philosophy Degree in Solid State Science from the Pennsylvania State University in 1973.

From January 1974 to March 1978, I was a nuclear engineer with the Fuel Behavior Research Branch, Office of Nuclear Regulatory Research. My principal responsibility was managing research related to the behavior of light water reactor fuel and to the release and transport of radioactive materials.

I assumed my current position in April 1978 and have since been responsible for the following tasks:

- technical and administrative coordination of NRC's research to enhance the capability of reactor operators
- technical and administrative coordination of NRC's research to improve reactor safety
- perform and review risk assessments of reactors, waste management and other fuel cycle facilities
- investigate methods to define and quantify acceptable risk
- develop and justify research programs and resources necessary to implement them
- staff liaison with Department of Energy, Advisory Committee on Reactor Safeguards and other groups

My responsibilities require me to maintain a current awareness of the most recent developments in the aspects of reactor safety related to the operator-machine interface and in studies performed which address improving the quality of that interface, such as those related to control room design and operating procedures.

BRUCE A. WILSON

PROFESSIONAL QUALIFICATIONS

I am a Reactor Engineer in the Operator Licensing Branch, Division of Project Management, Office of Nuclear Reactor Regulation. I am responsible for developing, preparing and administering examinations for applicants for reactor operator and senior reactor operator licenses. I am assigned to the Power and Research Reactor Group, which is primarily responsible for administering examinations on Combustion Engineering and Babcock & Wilcox designed reactors in addition to research reactors.

I received a Bachelor of Science Degree in Mechanical Engineering in 1966 from Syracuse University and a Master of Science in Nuclear Engineering in 1967 from the University of Washington.

In 1967, I entered active duty with the United States Air Force and was assigned to the 10 Megawatt Nuclear Engineering Test Facility (NETF), Wright Patterson AFB, Dayton, Ohio. From 1967 to 1968, I was a Project Engineer in the Experimental Branch where my primary function was to design and perform safety analyses of in-core irradiation test experiments.

From 1968 to early 1970, I was Chief, Reactor Engineering Section, where I performed safety analyses for reactor modifications and safety limit bases for technical specifications. During this period, I was certified as a Reactor Operator and Shift Supervisor at the NETF by the Air Force Directorate of Nuclear Safety.

From 1970 to 1971, I was assistant to the Chief, Operations and Maintenance Division during the final decommissioning and entombment of the facility.

In 1971, I was transferred to the Armed Forces Radiological Research Institute in Bethesda, Maryland. For eight months, I was Project Manager in the Accelerator Division and then transferred to the Reactor Division, where I was Assistant Physicist-in-Charge of a TRIGA Mark F reactor. I received a Senior Reactor Operator's License for this facility from the U.S. Atomic Energy Commission (AEC) and was primarily responsible for experiment safety review, technical specification revision and training.

In October 1973, I resigned my commission with the Air Force and joined the Operator Licensing Branch of the AEC. Since May of 1979, I have been assigned to the Systems Group of the Bulletins & Orders Task Force.

My functions on this Task Force have been to review and approve the Small Break Loss-of-Coolant Accident (SBLOCA) Guidelines developed by Westinghouse and B&W, and to insure that the applicable facilities have developed emergency procedures incorporating these Guidelines. Finally, I have audited the operators and training records to determine that sufficient training had been conducted regarding the SBLOCA phenomenon and the revised emergency procedures.

Allen Dale Johnson

Statement of Professional Qualifications

My name is Allen Dale Johnson. I was born July 22, 1931, at New Salem, North Dakota. I am employed by the United States Nuclear Regulatory Commission as a Reactor Inspector in the Reactor Operations and Nuclear Support Branch, Office of Inspection and Enforcement, Region V, Walnut Creek, California.

I was graduated from the University of Idaho in 1953 with a Bachelor of Science degree in chemistry and received a Juris Doctor degree from John F. Kennedy University, Orinda, California, in 1971. I am a member of the California State Bar and am duly licensed to practice law in the State of California.

I served as an officer in the U.S. Navy from July 1953 to July 1955.

From November 1955 through April 1963, I was employed by the Atomic Energy Division of Phillips Petroleum Company at the National Reactor Testing Station (NRTS) near Idaho Falls, Idaho. During my entire employment with Phillips Petroleum Company, I worked at the Material Testing Reactor (MTR) in the Operations Department. My job assignments were: Reactor Technician, Reactor Engineer, Shift Foreman and Shift Superintendent. As Shift Superintendent (3 years), I was responsible for the safe efficient operation of the reactor, associated supporting facilities, and experiments.

From May 1963 to the present, I have been employed by the NRC/AEC as a Reactor Inspector. My duties have included inspection and investigation of licensed facilities and activities for the purpose of ascertaining safety of facility operations and related activities. In addition, the duties include verification that activities conducted at licensed facilities have been performed in accordance with the rules and regulations of the Commission. I have been the principal inspector for power, test, and research reactors during all phases of construction, startup testing, and subsequent operations.

Frederick R. Allenspach

Statement of Professional Qualifications

June 1952	- Graduate - Polytechnic Institute of Brooklyn - Degree in Bachelor of Mechanical Engineering
July 1952 to August 1953	- New York Naval Shipyard
August 1953 to August 1954	- Republic Aviation Corporation
August 1954 to August 1956	- Military Service
September 1956 to June 1968	- Employed by the Brookhaven National Laboratory, Reactor Division. Approximately two years as operating shift supervisor in charge of an operating shift on the Brookhaven Graphite Research Reactor (BGRR). Approximately three years as BGRR day shift supervisor responsible for various reactor support activities. Approximately six years as BGRR Assistant Operations Group Leader primarily responsible for the temperature monitoring and reactor fuel management programs. One year as BGRR Operations Group Leader responsible for all operational aspects of the reactor. Included during this period at Brookhaven National Laboratory were several short term supplemental assignments to Brookhaven National Laboratory review and audit committees assigned the responsibility to determine if other Brookhaven nuclear reactors were being operated in accordance with the applicable rules and regulations.
June 1968 to June 1974	- Employed by the Atomic Energy Commission, Directorate of Licensing, Operational Safety Branch. My responsibilities included (as assigned); review and evaluation of applicants' organizational structure, and technical and administrative qualifications of applicants' proposed reactor operating organization, including emergency plans and industrial security plans; development of guides and codification of

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present and proposed practices with respect to administrative procedures for the operation of licensed reactors; the review of operating reports from licensed reactors for safety related items; and the preparation of reports relative to operating experiences at licensed reactors.

June 1974 to present - Employed by the Atomic Energy Commission (now Nuclear Regulatory Commission), Division of Project Management, Quality Assurance Branch. My responsibilities include review and evaluation of applicants' organizational structure, and technical and administrative qualifications of applicants' proposed reactor operating organization; development of standards, codes and guides with respect to administrative procedures for the operation of licensed reactors; and the development of uniform acceptance criteria for subjects required to be addressed by license applicants relating to operational safety matters.

Additional Educational Background: - I have attended the MIT course on Light Water Reactor Safety, a course in Industrial Defense and Disaster Planning for Privately Operated Facilities sponsored by the Dept. of Army at the Military Police School in Fort Gordon, Georgia, and a Babcock and Wilcox Simulator training course.

Professional Qualifications

Mark Phillip Rubin

My name is Mark Phillip Rubin. I am employed a' a Reactor Engineer, Reactor Systems Branch, Division of Systems Safety, U.S. Nuclear Regulatory Commission, Washington, D.C. The Reactor Systems Branch is responsible for evaluating the capability of reactor safety systems needed for safe shutdown during normal and accident conditions, including the performance of emergency core cooling systems. Currently, I am on temporary detail to the Bulletins and Orders Task Force where I am involved in the evaluation of operating reactor responses to the bulletins issued following the accident at Three Mile Island.

I attended the University of California at Los Angeles, California, receiving a BS degree in Nuclear Engineering in 1975 and an MS degree in Nuclear Engineering in 1976. I have also attended the graduate school at the University of Maryland and received an MBA degree in 1979.

Since 1976 I have been employed by the U.S. Nuclear Regulatory Commission in my present position. I have reviewed construction and operating license safety analyses in the reactor systems areas for compliance with NRC regulations as well as conducting studies on generic safety issues and developing staff positions.

THOMAS A. GREENE

PROFESSIONAL QUALIFICATIONS

CONTAINMENT SYSTEMS BRANCH

OFFICE OF NUCLEAR REACTOR REGULATION

I am a senior Systems Engineer in the Containment Systems Branch, Office of Nuclear Reactor Regulation. In this position I am responsible for the technical review, analysis and evaluation of the containment and secondary containment functional design; containment subcompartment analysis, containment heat removal systems, containment isolation system, combustible gas in containment, and containment leak testing program as described in Safety Analysis Reports to assure that nuclear power plants can be built and operated without undue risk to the health and safety of the public. I also assist in the preparation of standards, guides, and codes for the design and operation of reactors which deal with the containment system.

From 1973 to present, I have been employed with the Atomic Energy Commission and the Nuclear Regulatory Commission which was established by the Energy Reorganization Act of 1974. I have been the principal reviewer for a number of uclear power plants and am a former member of the American Nuclear Society Standard Committee 56.1, "Design Basis for Hydrogen Treatment in Containments."

From 1969 to 1973 I was employed as a Nuclear Safety Engineer at Combustion Engineering, Windsor, Connecticut. My major area of responsibility was the analysis of the thermal-hydraulic response of a nuclear power plant during a hypothetical loss of coolant accident by the use of computer blowdown codes. During this time period, I closely followed the LOFT semiscale test program at the National Reactor Testing Station, Idaho Falls, Idaho.

From 1968 to 1969, I was employed as a Nuclear Engineer at the San Francisco Bay Naval Shipyard, Vallejo, California. My duties were in all areas of engineering related to the overhaul and refueling of nuclear power systems on naval submarines and surface vessels. Also, at this time I taught a night course in Basic Nuclear Engineering at John F. Kennedy University, Martinez, California.

I received a Master of Science Degree in Nuclear Engineering from the University of Arizona in 1969 and a Bachelor of Science Degree in Engineering Physics from the University of Oklahoma in 1966. While at the University of Arizona, I was a graduate assistant and instructor in the radioisotopes and instrumentation course.

Since graduating from college, I have attended various courses in reactor technology and safety.

I have been a member of the American Nuclear Society since 1965.

PHILIP J. MORRILL
PROFESSIONAL QUALIFICATIONS
REGION V - WALNUT CREEK, CALIFORNIA
OFFICE OF INSPECTION AND ENFORCEMENT

My name is Philip J. Morrill. I am employed by the United States Nuclear Regulatory Commission as a reactor inspector in the Reactor Operations and Nuclear Support Branch, Office of Inspection and Enforcement, Region V, Walnut Creek, California. My primary responsibility in this position is the inspection of nuclear power plants during the operating phase to determine compliance with NRC rules and regulations.

I received a Bachelor of Science degree from the U.S. Naval Academy in 1966. I was employed by the U.S. Navy in the Naval Nuclear Power Submarine program from 1966 until 1971. During this time, I became qualified as Engineering Officer of the Watch for the AIW pressurized water nuclear propulsion plant prototype and was later qualified as Engineering Officer of the Watch on board the USS John Marshall (SSBN 611 (G)), a nuclear powered polaris missile submarine (1969 through 1971). I was also the ship's Main Propulsion Assistant (responsible for maintenance and administration of the nuclear reactor and power generation equipment) for one and one-half years of this time. In 1971, I joined the Bechtel Corporation in San Francisco, California and was assigned to the Susquehanna Steam Electric Station project mechanical group. From August 1971 through September 1972, I was responsible for the design and development of the radioactive waste disposal system. From September 1972 through January 1974, I was assigned duties of the project licensing engineer. From January 1974 through March 1976, I was the project nuclear group leader responsible for managing and supervising the efforts of 8 to 10 engineers.

In March 1976, I was hired by the U.S. Nuclear Regulatory Commission, Office of Inspection and Enforcement, Region V, in Walnut Creek, California, as a reactor inspector for the Reactor Construction and Engineering Support Branch. In this position, I participated in several construction inspections of the San Onofre Nuclear Generating Station and successfully completed a nondestructive examination school at Convair Division of General Dynamics (San Diego, California), as well as a quality assurance and inspection course in Bethesda, Maryland. In January 1977, I transferred to the Reactor Operations and Nuclear Support Branch of Region V, Office of Inspection and Enforcement and was assigned as back-up inspector for the Trojan Nuclear Plant. In succeeding months I participated in inspections of the Rancho Seco. Humboldt, and Trojan nuclear plants in addition to completing five weeks of pressurized water reactor systems and operations training. For about one year I was then assigned as principle inspector for the Trojan Plant. In the fall of 1978, my assignment was again changed to follow-up the preoperational testing of the Diablo Canyon

Nuclear plant. Although these have been my principal assignments, I Have participated in a variety of research and power reactor inspections during the last two years.

I am presently a registered Professional Mechanical Engineer and Nuclear Engineering in the State of California.

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DANIEL J. GARNER

PROFESSIONAL QUALIFICATIONS

OPERATING REACTORS BRANCH NO. 4

OFFICE OF NUCLEAR REACTOR REGULATION

I am a Project Manager in Operating Reactors Branch No. 4, Office of Nuclear Reactor Regulation. In this position I coordinate the licensing activities for two nuclear power plants, including Rancho Seco, and for various research reactors. For these facilities I am the principal point of contact between the licensees and the NRC.

I received a commission as a line officer in the United States Naval Reserve in April 1972. For the duration of my four year commission I was assigned as an instructor at the Naval Nuclear Power School, Mare Island Naval Shipyard. My duties included teaching physics and reactor principles to both enlisted and commission officer students. I was also assigned during my last year as a Division Director in one of seven major academic divisions in the school.

In 1976 I took a position in the Division of Naval Reactors Headquarters, U. S. Department of the Navy as a Technical Assistant to the Deputy Director. In that capacity I was responsible for the curriculum and technical operation of the Naval Nuclear Power Schools. My duties also included responsibility for the training requirements of civilian shipyard test engineers who supervise testing of naval nuclear propulsion plants, and for the initial training phase of all new engineers at Division of Naval Reactors.

In March of 1979, I joined the staff of the NRC in the capacity in which I now serve.

My professional education includes a Bachelor of Science in Electrical Engineering from the University of Kansas and a Master of Science in Nuclear Engineering from the Massachusetts Institute of Technology.

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ROBERT A. CAPRA

PROFESSIONAL QUALIFICATIONS

BABCOCK & WILCOX PROJECT MANAGER

PROJECT MANAGEMENT GROUP

BULLETINS & ORDERS TASK FORCE

Since June 1979, I have served as the Babcock & Wilcox (B&W) Project Manager for the Bulletins & Orders Task Force (the Task Force), Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission (NRC). In this capacity, I coordinate and establish priorities for the work being done by the Task Force which is associated with the B&W designed operating nuclear power plants (except Three Mile Island Nuclear Station, Units 1 and 2). I coordinate the scope and schedule of the work required of the B&W licensees by the Task Force. I also serve as the principal liaison between the Task Force, the licensees and the B&W Owners' Group.

I enlisted in the United States Navy in July 1964 and served in that capacity for three years. During that time my duties included attending the Enlisted Naval Nuclear Power School, Mare Island, California followed by subsequent study and qualification as a reactor operator and staff instructor on the Navy's "DIG" reactor located in West Milton, New York.

Following enlistment, I attended the United States Naval Academy where I graduated in June 1971 with a Bachelor of Science degree in Marine Engineering and was commissioned as a line officer in the United States Navy. Additional graduate level studies in nuclear reactor theory, thermodynamics, electrical engineering, health physics and other related engineering fields were completed in 1972 at the Officer Naval Nuclear Power School, Bainbridge, Maryland. I subsequently returned to West Milton, New York where I studied and qualified as a Senior Reactor Operator on the Navy's "DIG" reactor.

From 1973 to 1976, I served aboard an operating nuclear submarine, during which time my duties included standing watch as a Senior Reactor Operator and directing, training and supervising technicians in the operation, maintenance and repair of various equipment and systems primarily associated with the ship's nuclear reactor. During this period, my assignments included supervision of the Operations Department, Electrical Division, Reactor Controls Division, Main Propulsion Division, and the Chemistry and Radiological Control personnel. In addition, I qualified as Chief Engineer for the supervision of operation and maintenance of Naval Nuclear Propulsion Plants.

From 1976 to 1978, I was assigned as a Company Officer at the United States Naval Academy where my duties included supervising, directing and evaluating the training and activities of 130 officer candidates (midshipmen).

I joined the NRC staff in July 1978, where I served as a Licensing Project Manager in the Division of Project Management. In this capacity, I coordinated the safety review for two construction permit applications (New Haven, Units 1 and 2 and Haven, Unit 1) and served as the Project Manager for one plant under construction (North Anna, Units 3 and 4). In addition, I served as the Licensing Topical Report Manager for General Electric Topical Reports. I remained in that position until my assignment to the Bulletins & Orders Task Force.

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DALE F. THATCHER

PROFESSIONAL QUALIFICATIONS

INSTRUMENTATION & CONTROL SYSTEMS BRANCH

DIVISION OF SYSTEMS SAFETY

Since May of 1979, I have been assigned to the Bulletins and Orders Task Force as a technical reviewer in the area of instrumentation and control. Just prior to this assignment I was a member of the NRR team which aided in the Three Mile Island Recovery Operation.

My previous position was that of Senior Reactor Engineer, Section B, of the Instrumentation and Control Systems Branch, Office of the Assistant Director for Plant Systems, Division of Systems Safety.

In the ICSB, my primary responsibility was to perform technical reviews of the design, fabrication, and operation of instrumentation and control systems for nuclear power plants. This review encompasses evaluation of applicant's safety analysis reports, generic reports and other related information on the instrumentation and control designs.

I graduated from Lehigh University with a Bachelor of Science Degree in Electrical Engineering in June 1971.

From my graduation in June 1971 until my employment at the Commission, I was an Instrumentation Engineer with Gilbert Associates, Inc., an Architect - Engineering company located in Reading, PA. My responsibilities included the design and evaluation of various instrumentation and control systems including primarily the areas of reactor protection systems and other safety systems for various domestic nuclear power plants.

I joined the Regulatory staff of the Atomic Energy Commission in March 1974 as a Reactor Engineer. Since then I have participated in the review of instrumentation control and electrical systems of numerous nuclear power stations and standard plant designs. In addition, I have participated in the formulation of related standards and regulatory guides.

I am a member of the Institute of Electrical and Electronics Engineers (IEEE) and have participated in the development of IEEE Standard 379-1977, "IEEE Standard Application of the Single Failure Criterion to Nuclear Power Generating Station Class IE Systems" and other proposed standards.

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THOMAS M. NOVAK

PROFESSIONAL QUALIFICATIONS

I am employed as Chief, Reactor Systems Branch, Division of Systems Safety, Office of Nuclear Reactor Regulation. My responsibilities include supervising the performance of safety reviews and evaluations of applications for nuclear power plant construction permits and operating licenses.

I graduated from Rutgers University in 1958 with a Bachelor of Science degree in Mechanical Engineering. In 1968 I received a Master's degree in Mechanical Engineering from Catholic University of America.

In June of 1958, I joined the Westinghouse Electric Corporation and participated in the Graduate Student Training Program. In November of 1958, I accepted a permanent assignment at the Bettis Atomic Power Laboratory, West Mifflin, Pennsylvania. While at the laboratory, I worked in a section responsible for thermal and hydraulic design and analysis of submarine reactor cores. As a member of this section, I performed design and accident analyses for a variety of core designs. Prior to leaving the laboratory, I was assigned project responsibility for a series of tests to determine heat transfer and fluid flow characteristics for a potential fuel element configuration.

In November of 1964, I undertook employment with the U.S. Marine Engineering Laboratory, Annapolis, Maryland. As a senior project engineer, I was given the responsibility for developing and carrying out R&D programs for improving the performance of naval steam generators. My duties included supervising the design, construction, and operation of high pressure heat transfer and water treatment test facilities. I supervised the development of computer codes to predict steady-state and dynamic thermal performance characteristics of naval propulsion boilers involving both pressure-fired and forced-draft systems.

In November of 1968, I accepted a position with the Atomic Energy Commission in the Division of Reactor Licensing (DRL). While a member of DRL, I participated in the review of both pressurized and boiling water reactors.

Following a reorganization within the Regulatory Staff, I was assigned on March 5, 1972, to the Reactor Systems Branch as a senior nuclear engineer. My primary work assignments involved the review of emergency core cooling systems.

In March of 1973, I was promoted to my present position. This branch has the responsibility for the reviews of core thermal and hydraulic behavior for normal operations, anticipated transients, and accidents. In May of 1979, as part of the Office of Nuclear Reactor Regulation interim organizational changes to deal with Three Mile Island 2, I was assigned to the Bulletins and Orders Task Force. My present assignment is Deputy Director, Bulletins and Orders Task Force.

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JAMES WING

EFFLUENT TREATMENT SYSTEMS BRANCH
DIVISION OF SITE SAFETY AND ENVIRONMENTAL ANALYSIS

My name is James Wing. I am a senior nuclear engineer in the Effluent Treatment Systems Branch, Division of Site Safety and Environmental Analysis, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission.

I received a Bachelor of Science degree in Chemistry from the University of Tennessee in 1949, a Master of Science degree in Chemistry from Purdue University in 1953, and a Ph.D. degree in Chemistry from Purdue University in 1956.

Before joining the Commission, I was employed by Argonne National Laboratory from 1955 to 1969 as first, Assistant Chemist and then, Associate Chemist. In this capacity, I performed basic research in nuclear chemistry. From 1969 to 1975, I was employed by National Bureau of Standards as a research chemist and computer programmer. In these two positions, I did research work on radiochemistry and wrote computer programs for laboratory automation.

I have written 28 technical papers and 10 laboratory reports on various topics, including nuclear chemistry, radiochemistry, air pollution, applied mathematics, and food technology. In the academic year of 1964-1965, I was a Fulbright Lecturer. I am a member of the American Chemical Society.

I have been a staff member of the Nuclear Regulatory Commission since January 1975. From 1975 to 1978 I was a member of the Accident Analysis Branch. Since 1978, I have been a senior nuclear engineer in the Effluent Treatment Systems Branch. My duties in this position include review and evaluation of radioactive waste treatment and effluent control systems of nuclear power plants.

Philip R. Matthews

Professional Qualifications

I am employed by the U.S. Nuclear Regulatory Commission as a Section Leader in the Auxiliary Systems Branch, Division of Systems Safety, Office of Nuclear Reactor Regulation. I am responsible for supervision of technical personnel engaged in analysis and safety evaluation of nuclear power plant auxiliary systems including the main steam and feedwater, auxiliary feedwater, component cooling water, service water, new and spent fuel storage and handling, plant ventilating and air conditioning, and fire protection systems.

I attended the University of California, Berkely, California and received a Bachelor of Science degree in Chemistry in 1947. Subsequently, I have completed several graduate courses in mechanical and nuclear engineering.

In 1947, I commenced work at the Knolls Atomic Power Laboratory, General Electric Co., Schenectady, N.Y. I worked there until 1968 on various naval nuclear submarine and surface ship propulsion power plant projects. I had technical and management responsibility for nuclear plant mechanical and fluid systems design, testing, performance evaluation, prototype and shipboard reactor plant start-up and sea trials.

In 1968, I transferred to the General Electric Co., Nuclear Energy Division in San Jose, California. I was Quality Assurance manager for the Atomic Power Equipment Department responsible for quality assurance of APED purchased engineered equipment and installation of APED equipment at BWR nuclear plant sites.

I joined the Nuclear Regulatory Commission in 1973 as a nuclear engineer in the Office of Standards. In 1975, I assumed my present duties as Section Leader in the Auxiliary Systems Branch. In this position, I have had two major special assignments; namely, 1) to direct the technical preparation, issuance and plant specific implementation review of nuclear plant fire protection guidelines following the 1975 fire at Browns Ferry Nuclear Plant and 2) in 1979, to direct a Task Force in reviewing the design and operation of Auxiliary Feedwater Systems of operating nuclear plants with Westinghouse and Combustion Engineering designed reactors and provide specific recommendations for improving Auxiliary Feedwater System reliability.

PAUL E. NORIAN

PROFESSIONAL QUALIFICATIONS

I have been assigned to the Bulletins and Orders Task Force as a member of the Analysis Group since June 1979. I serve as Alternate Group Leader and coordinate the reviews of small break loss-of-coolant accidents (LOCA) and transient analyses submitted by the vendor owner's groups since the Three Mile Island accident. From 1975 until this assignment, I was Section Leader of the Systems Analysis Section, Analysis Branch, Division of Systems Safety. I was responsible for supervising the review of reactor vendor transient and LOCA analysis methods, the improvement of NRC analysis methods used in related accident analyses and the performance of staff audit calculations for transients and LOCAs.

I graduated from Lehigh University in June 1955 with a Bachelor of Science Degree in Engineering Physics. I also attended Drexel Institute of Technology, Catholic University of America, and the University of Maryland where I have taken various graduate courses in mathematics, physics, and electrical engineering.

In July 1955, I began work as a physicist with the duPont Company at the Savannah River Plant in Aiken, South Carolina. From that time until March 1962, I worked in the Works Technical Department on operational physics problems associated with the heavy water production reactors at Savannah River. This work included such assignments as the development of monitoring systems, performance of physics calculations required in reactor operation and in the development of new fuel elements, the review of operating procedures, and the analysis of various operating problems. In March 1962, I was transferred to the duPont Company's Chestnut Run Laboratoreis in Wilmington, Delaware, and worked for its Film Department on the development of industrial applications for plastic films.

In December 1963, I accepted a position with the Division of Reactor Licensing of the U.S. Atomic Energy Commission, and was project leader in the construction permit review of Consolidated Edison's Indian Point No. 2 reactor and Wisconsin-Michigan's Point Beach No. 1 reactor. I was assigned as a nuclear engineer in the Systems Performance Branch of the Division of Reactor Standards in March 1967. My responsibilities included analyzing and evaluating the performance of engineered safety systems and performing computer calculations for the evaluation of containment response and loss-of-coolant accidents. In March 1971, I participated in the Regulatory Task Force reappraisal of emergency core cooling systems for light water reactors. My main responsibility for the task force was the review of computer codes and input assumptions for LOCA analyses. In May 1973, I was assigned to the Core Performance Branch in the Directorate of Licensing. I served as Section Leader in the Thermal Hydraulics Sections and supervised the review of portions of reactor vendor model changes to conform with the new requirements for LOCA models specified in Appendix K to 10 CFR Part 50.

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

CERTIFICATE OF SERVICE

I hereby certify that copies of "NRC STAFF RESPONSES TO CALIFORNIA ENERGY COMMISSION'S FIRST SET OF INTERROGATORIES TO THE NUCLEAR REGULATORY COMMISSION," in the above-captioned proceeding have been served on the following by deposit in the United States mail, first class, or, as indicated by an asterisk, through deposit in the Nuclear Regulatory Commission's internal mail system, this 11th day of December, 1979:

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