UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
Sacramento Municipal tility District	))	
(Rancho Seco Nuclear Generating Station	))	Docket No. 50-312 (SP)

NRC STAFF TESTIMONY OF MARK P. RUBIN AND THOMAS M. NOVAK REGARDING THE SENSITIVITY OF THE ONCE-THROUGH STEAM GENERATOR DESIGN

(Additional Board Question 3)

Q.1 Please state your name and position with the NRC.

A. My name is Mark P. Rubin. I am an employee of the U.S. Nuclear Regulatory Commission assigned to the Reactor Systems Branch, Division of Systems Safety, Office of Nuclear Reactor Regulation. However, from June through December, 1979, I was assigned to the Bulletins and Orders Task Force, Office of Nuclear Reactor Regulation.

My name is Thomas M. Novak. I am an employee of th U.S. Nuclear Regulatory Commission assigned to the Reactor Systems Branch, Division of Systems Safety, Office of Nuclear Reactor Regulation. However, from June through December, 1979, I was assigned as the Deputy Director of the Bulletins and Orders Task Force, Office of Nuclear Reactor Regulation.

Q.2 Have you prepared a statement of professional qualifications?

A. Yes, A copy of our statements is attached to this testimony.

Q.3 Please state the nature of the responsibilities that you have had with respect to the Rancho Seco Nuclear Generating Station.

A. The accident at Three Mile Island Unit 2 (TMI-2) on March 28, 1979 involved a feedwater transient coupled with a small break in the reactor coolant system. Because of the resulting severity of the ensuing events and the

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potential generic aspects of the accident on other reactors, the NRC staff initiated prompt action to: (1) assure that other reactor licensees, particularly those plants such as Rancho Seco which have a smiliar design to TMI-2, took the necessary actions to substantially reduce the likelihood of future TMI-2-type events from occurring, and (2) start comprehensive investigations into the potential generic implications of this accident on other operating plants. To accomplish some of this work, the Bulletins and Orders Task Force (B&OTF) was established within the Office of Nuclear Regulator Regualtion (NRR) in May 1979. The B&OTF was responsible for reviewing and directing the TMI-2-related staff activities associated with loss of feedwater transients and small breakloss-of-coolant accidents (LOCAs) for all operating plants to asssure their continued safe operation.

The initial priority of the B&OTF was placed on evaluating the actions taken by the B&W operating plant licensees in response to the Confirmatory Shutdown Orders issued in May 1979. We were assigned to the Task Force in mid-June 1979. Upon assuming those positions, we participated in the final preparation of the Staff Safety Evaluation which documented our evaluation of SMUD's compliance with the immediate requirements of the May 7, 1979 Order. On the basis of this report, issued on June 27, 1979, the Rancho Seco Facility was authorized to return to power operation.

Q.4 What is the purpose of your testimony?

A. The purpose of our testimony is to respond to Board concerns relative to the design of the steam generator and, particularly, Additional Board Question No. 3 which reads:

> "It appears from a Board Notification issued by R. H. Vollmer on December 5, 1979, that the basic design of the Once Through Steam Generator (OTSG) may so closely couple primary system behavior to secondary system disturbances that gross disturbance of the primary system is inevitable for feedwater transients. Further, it seems there are situations in which an operator may not be able to tell exactly what is wrong or what response is appropriate (e.g., overcooling vis-a-vis a small-break LOCA).

"a. What changes in the system and procedures have been made to ameliorate this situation?

"b. What are the implications for safety of operating Rancho Seco before any uncertainties are resolved?"

Q.5 Describe the once-through steam generator (OTSG) design of Babcock & Wilcox (B&W) facilities and compare it to those of other pressurized water reactor manufacturers.

A. B&W plants employ a once through steam generator (OTSG) design, rather than U-tube steam generators which are used in other pressurized water reactors. Each steam generator has approximately 15,000 vertical straight tubes, with the primary coolant entering the top at 603-608° F and exiting the bottom at about 550° F. Primary coolant flows down inside the steam generator tubes, while the secondary coolant flows up from the bottom on the shell side of the OTSG. The secondary coolant turns to steam about half way up, with the remaining length of the steam generator being used to superheat the steam. In the U-tube steam generators used by other reactor manufacturers, the primary coolant enters at the bottom. This coolant flows up inside the steam generator tubes, which are bent to form a U towards the top of the generator, and then run back down the other side of the generator. Consequently, the primary coolant exits at the bottom of the steam generator. The tubes, containing the heated reactor coolant, are continuously covered by the liquid on the steam generator's shell side. Therefore, steam produced by this type of generator is saturated, unlike the superheated steam available from the B&W OTSG.

Q.6 With specific reference to the Three Mile Island incident, does the OTSG design of B&W facilities make these plants more susceptible and sensitive to a loss of feedwater transient?

A. Yes.

Q.7 Explain the nature of the increased susceptibilities and sensitivities of Babcock & Wilcox once-through steam generators.

A. The sensitivity of the B&W design to loss of feedwater transients was recognized from evaluations of the TMI-2 accident. In the B&W once through steam generator design, secondary coolant turns to steam about half way up, with the remaining length of the steam generator being used to superheat the steam. The secondary-side heat transfer coefficient, in the steam space of the OTSG is much less than that in the bottom liquid section. This results in a heat transfer rate from the primary system which is quite sensitive to the liquid level in the steam generator. If a feedwater transient occurs, the liquid-vapor interface will move, resulting in a change in the overall heat transfer from the primary system. This tends to closely couple the primary system to the secondary side condition. The response of the primary system pressure and the pressurizer level to a change in feedwater flow rate is comparatively rapid, and therefore B&W designed plants are considered more sensitive and susceptible to feedwater transients than the other types of PWR reactors.

In its May 7 Order, the staff determined that "....B&W designed ractors appear to be unusually sensitive to certain off-normal transient conditions originating in the secondary system." In the Order the staff identified features of the B&W design that contributed to this sensitivity which could be improved upon to compensate for sensitivity to loss of feedwater events. These features are:

1. Design of the steam generators to operate with relatively small liquid volumes in the secondary side.

   This results in less total heat sink for the primary system, and less time till steam generator dry-out; following loss of main feedwater.

2. The lack of direct initiation of reactor trip upon the occurrence of off-normal conditions in the feedwater system.

   This results in the reactor continuing at power for 6 to 8 seconds after loss of feedwater or turbine.

3. Reliance on an integrated control system (ICS) to automatically regulate feedwater flow.

   This system can potentially experience failure modes which result in feedwater transients.

4. Actuation before reactor trip of a pilot operated relief valve on the primary system pressurizer.

   This valve, if stuck open when actuated, can aggravate the system transient.

5. Low steam generator elevation (relative to the reactor vessel) which provides a smaller driving head for natural circulation.

   This might result in less natural circulation cooling following the loss of main reactor coolant pumps.

The staff had determined that because of the five design features indicated above, "B&W designed ractors place more reliance on the reliability and performance characteristics of the auxiliary feedwater system, the integrated control system, and the emergency core cooling sytem (ECCS) performance to recover from anticipated transients, such as loss of offsite power and loss of normal feedwater, than do other PWR designs."

Q.9 Have any steps been taken with regard to the Rancho Seco facility to mitigate that facility's sensitivity to a loss of feedwater transient?

A. Yes.

Q.10 Identify what steps have been taken and when they were completed.

A. In the May 7 Order, the staff required a number of actions to mitigate the B&W sensitivity to loss of feedwater transients. These are indicated below.

   (a) Upgrade the timeliness and reliability of delivery from the Auxiliary Feedwater System by carrying out actions as identified in the licensee's letter of April 27, 1979.

   (b) Develop and implement operating procedures for initiating and controlling auxiliary feedwater independent of Integrated Control System control.

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

(d) Complete analyses for potential small breaks and develop and implement operating instructions to define operator action.

(e) Provide for one Senior Licensed Operator assigned to the control room who has had Three Mile Island Unit 2 (TMI-2) training on the B&W simulator.

Because of the small water volume, as well as the other factors noted above, the staff found that B&W designed reactors place more reliance on the reliability and performance characteristics of the auxiliary feedwater system. This occurs because the B&W steam generators contain less total water inventory than those of the other reactor vendors, and therefore, need more rapid actuation of the auxiliary feedwater system for continued decay heat removal following the loss of main feedwater. To increase the reliability and performance characteristics of the Rancho Seco auxiliary feedwater system, so as to compensate for the low SG water volume, the Commission required a number of actions in its Order of May 7, 1979. These actions are listed in enclosure 1 to this testimony.

In their June 27, 1979 Evaluation of the Licensee's Compliance with the May 7 Order, the staff found that Rancho Seco had satisfactorily completed the actions prescribed in the Order. Bulletin actions of reducing the high pressure reactor trip setpoint had also been completed.

Q.11 Do these steps provide an acceptable level of safety at the Rancho Seco facility relative to steam generator sensitivity to a loss of feedwater transient?

A. Yes.

Q.12 Provide your reasons.

A. The safety concerns regarding sensitivity of Rancho Seco to loss of feedwater transients is primarily related to a potential inability to cool the reactor core. Due to the B&W design, the reactor system responds more rapidly to loss of feedwater transients, than the other types of pressurized water reactors. Therefore the rapid availability of auxiliary feedwater is required. The staff required specific actions in the May 7 Order to increase the reliability and timeliness of this system. Additionally, operating procedures were modified to assure that auxiliary feedwater is supplied to the steam generator when required.

To further reduce the sensitivity of Rancho Seco to loss of feedwater transients, direct reactor trips were installed for loss of feedwater and turbine trip. This provides additional time for the auxiliary feedwater system to respond, when a loss of feedwater transient occurs. Analytical studies have also shown that a temporary failure of all feedwater systems can be dealt with by alternate procedures such as feed and bleed. The Staff believes that these actions provide the necessary assurance that the Rancho Seco facility will respond safely to a loss of feedwater transient.

Q.14 Does the staff contemplate any other actions at the Rancho Seco facility to make that facility less susceptible and sensitive to loss of feedwater transients?

A. Yes.

Q.15 Identify what additional steps will be taken and the timeframe for their completion.

A. The anticipatory reactor trip system is being upgraded to safety quality, and should be completed this year. A reliability study of the integrated control system (ICS) is continuing. Coming out of this study there may be recommendations which will further reduce the susceptibility of Rancho Seco to loss of feedwater transients. For more details on this study see the testimony of Dale Thatcher on Board Question 16. Also, an Interim Reliability Evaluation Program (IREP) will be carried out on all B&W reactors, to provide a limited risk assessment of the design. Details of current staff activities, studies and requirements in these areas, can be found in Draft 2 of the NRC TMI-2 Action Plan, NUREG-0660.

Additionally, besides the question of sensitivity to loss of feedwater transients, the staff has begun a study of the sensitivity of B&W plants to excess feedwater transients. These studies are currently being pursued on B&W construction permit holders. The staff's program for this issue will be discussed in Section II.E.5 of Draft Three of the Commission Action Plan. Pending completion of these studies, the staff recommended in a Commission memo dated January 22, 1980, that construction not be halted on B&W plants under construction.

Q.16 What changes in the system and procedures have been made to ameliorate B&W overcooling sensitivity?

A. No changes in systems and procedures have yet been taken to completely damp out the response of the primary system to secondary side transients, specifically in regard to secondary side overcooling transients. Following the issuance of the Order, studies have continued on the general topic of B&W system response and sensitivity, specifically in respect to overcooling transients. These studies are currently being pursued with B&W construction permit applicants and have identified areas where various system modifications could be performed to reduce system sensitivity and transient frequency.

The various possible modifications and their potential effect are being evaluated by the staff. Modifications which are required by the staff will also be considered for backfit to Rancho Seco. Details will be available in Section III.E.5 of NUREG-0660 (Draft 3).

Q.17 What are the implications for safety of operating Rancho Seco before any uncertainties with regard to overcooling transients are resolved?

A. Concurrent with the work described above, studies have been conducted to determine if there exists any operational safety problems from overcooling transients. While reactor overcooling transients are undesirable, the staff has determined that these events will not result in loss of adequate core cooling or exceed the fuel damage criteria. Analyses performed both by the construction permit holders and the staff demonstrate that a severe overcooling transient will only generate some void in the primary system and will not interrupt natural circulation cooling. Therefore, the staff does not believe that an operational safety problem exists for Rancho Seco from this issue. The inability to quickly differentiate between small break LOCAs and overcooling transients is tolerable since immediate required automatic manual actions are the same for both events. Therefore, this similarity of response does not compromise core cooling.

Yet, the staff recognizes that while the current level of safety in a plant may be acceptable, changes may be possible which will provide even greater levels of protection. The staff has determined that a reduction in overcooling transient frequency and severity will enhance the defense in depth concept, even though current plant performance meets all current safety criteria in regard to core cooling. Therefore, while design changes on Rancho Seco to reduce sensitivity will be considered, there are no significant safety problems with deferring these actions until staff review is completed.

Enclosure (1)

Auxiliary Feedwater System Upgrade

1. Review procedures, revise as necessary and conduct training to ensure timely and proper starting of motor driven auxiliary feedwater (AFW) pump(s) from vital AC buses upon loss of offsite power.

2. To assure that AFW will be aligned in a timely manner to inject on all AFW demand events when in the surveillance test mode, procedures will be implemented and training conducted to provide an operator at the necessary valves in phone communications with the control room during the surveillance mode to carry out the valve alignment changes upon AFW demand events.

3. Procedures will be developed and implemented and training conducted to provide for control of steam generator level by use of safety grade AFW bypass valves in the event that ICS steam generator level control fails.

4. Verification that Technical Specification requirements of AFW capcity are in accordance with the accident analysis will be conducted. Pump capacity with miniflow in service will also be verified.

5. Modifications will be made to provide verification in the control room of AFW flow to each steam generator.

6. Review and revise, as necessary, the procedures and training for providing alternate sources of water to the suction of the AFW pumps.

7. Design review and modification, as necessary, will be conducted to provide control room annunication for all auto start conditions of the AFW system.

8. Procedures will be developed and implemented and training conducted to provide guidance for timely operator verification of any automatic initiation of AFW.

9. Verification will be made that the air operated level control valves (a) Fail to the 50% open position upon loss of electrical power to the electrical to pressure converter, and (b) Fail to the 100% open position upon loss of service air. The AFW bypass valves are safety grade.

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 syst s. 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 t e 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.