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)	)	

NRC STAFF TESTIMONY OF PAUL E. NORIAN ON LOGIC FOR
REACTOR COOLANT PUMP TRIP IN SMALL-BREAK LOCA
(Additional Board Question 2)

Q1. Please state your name and position with the NRC.

A. My name is Paul E. Norian. I am Section Leader of the Systems Analysis Section, Analysis Branch, Division of Systems Safety. I have held this position since 1975 and am 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. From June through December 1979, I was assigned to the Bulletins and Orders Task Force as a member of the Analysis Group. I served as Alternate Group Leader and coordinated 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.

Q2. Have you prepared a statement of professional qualifications?

A. Yes. A copy of my statement is attached to this testimony.

Q3. Please state the purpose of this testimony.

A. The purpose of this testimony is to respond to the additional Board Question No. 2 which reads as follows:

> "We note (letter D. Ross to J. T. Mattimoe, December 14, 1979) that there is still some dispute as to the fundamental logic for Reactor Coolant Pump (RCP) trip in a small-break LOCA.
>
> a) What current instructions to reactor operators govern tripping of the pumps in small-break LOCA's and upon what theory of system behavior are those instructions based?
>
> b) What are the implications for safety of operating Rancho Seco until the exact behavior of the system in a small-break LOCA is well understood?

Q4. What are the current NRC instructions to licensees with regard to the tripping of reactor coolant (RC) pumps?

A. The instructions for tripping the RC pumps are stated in IE Bulletin No. 79-05C, Short Term Actions 1.A. This item states "Upon reactor trip and initiation of HPI caused by low reactor coolant system pressure, immediately trip all operating RCPs." These instructions have been included in the Rancho Seco Operating Procedures in accordance with the requirement in the bulletin.

Q5. Has there been a difference of interpretation of these instructions between the Licensee and the staff?

A. Yes. Rancho Seco's original interpretation of the bulletin requirement was that RC pump trip was not required when high pressure injection (HPI) is initiated manually rather than automatically. It is the staff's opinion that the RC pumps should be tripped whenever the system pressure decreases to the HPI actuation pressure regardless of whether HPI is initiated manually or automatically. The position is stated in the letter from D. Ross to J. Mattimoe dated December 14, 1979. We have been assured by Rancho Seco that they will comply with this requirement.

Q6. The logic to trip the RC pumps following reactor scram and initiation of HPI is based on what calculations or analyses?

A. The requirement to trip the reactor coolant pumps following reactor scram and initiation of HPI is based on the results of small-break LOCA calculations performed by Babcock and Wilcox (Reference 1). These calculations indicated that cladding temperatures in excess of the 2200°F licensing limit would occur for a range of small-break LOCAs if the RC pumps are tripped later in the transient with a large amount of steam in the system. The continued operation of the pumps maintains the fluid in a mixed condition (steam and water) and results in an increase in the total mass loss out of the postulated break If the pumps are initially tripped, the water and steam phase will eventually separate and the break flow will change from a mixture of water and steam and become essentially all steam. Consequently, more mass will remain in the primary system if the RC pumps are quickly tripped compared to the case where RC pump trip is significantly delayed. Small-break LOCA analyses performed assuming early pump trip show that the 2200 F licensing limit is not exceeded. Since unacceptable cladding temperatures were calculated for a range of small-break LOCAs assuming delayed RC pump trip, IE Bulletin 79-05C was issued to require prompt tripping of the RC pumps. The NRC generic assessment of the PWR vendor analyses assuming delayed pump trip is presented in NUREG-0623 (Reference 2).

Q7. Have any other analyses been performed to evaluate system performance in the event of a small-break LOCA?

A. Yes. Item (d) of the Commission's May 7 Order required Rancho Seco to complete analyses for potential small breaks and develop and implement operating instructions to define operator actions. In response to this requirement, B&W performed a series of small-break LOCA calculations which were submitted to the NRC in Reference 3. The staff has completed its review of those analyses and the models used to perform the calculations. The generic analyses performed in this report for the 177 fuel assembly lowered loop plant apply to Rancho Seco. Section 5 of this report, entitled "The Small Break Phenomena - Description of Plant Behavior," contains a qualitative description of plant response for various break sizes, break locations and assumed availability of systems such as high pressure injection and auxiliary feedwater. These LOCA scenarios describe the course of system behavior that could result in continuous depressurization of the primary system, stabilization of primary system pressure near or above the secondary side pressure, and breaks that result in system repressurization. The analytical bases for these accident scenarios are presented in Section 6 of the B&W report, and includes a series of small break LOCA calculations for breaks in the cold leg and pressurizer. The results of the staff review are presented in NUREG-0565 (Reference 4). This report concluded that the analysis methods used by Babcock and Wilcox are satisfactory for the purpose of predicting trends in plant behavior and that a sufficient spectrum of small-break LOCA analyses had been performed to identify the anticipated system performance. These analyses provided an adequate basis for the development of improved procedures, and demonstrate that operator actions together with a combination of heat removal by the steam generators, high pressure injection system, and the break ensure adequate core cooling.

Q8. Based on the above calculations and analyses, does the Staff believe that Rancho Seco can be operated safely in the event of a small-break LOCA?

A. Yes. The staff believes that Rancho Seco can be operated safely without knowledge of the exact behavior of the system in a small-break LOCA. Such understanding would enable the staff to determine the optimum RC pump operation following a small break LOCA. Since this cannot be determined at this time, the staff has required that the RC pumps be tripped since that operation would result in acceptable consequences.

The staff believes that improved understanding of small-break LOCA is necessary and supports the current NRC research programs at the Semi-scale and LOFT facilities during 1980 which will explore the sensitivity of RC pumps running and tripped. The results of these test programs could indicate the optimum mode for pump operation following a small-break LOCA. In the meantime, our current understanding assuming the RC pumps are tripped is sufficient to assure safe operation of the facility.

References:

1. Letter from J. J. Mattimoe, SMUD, to R. H. Engelken, dated September 19, 1979 transmitting "Supplemental Small Break Analysis", in response to IE Bulletin 79-05C.

2. NUREG-0623, "Generic Assessment of Delayed Reactor Coolant Pump Trip during Small Break Loss-of-Coolant Accidents in Pressurized Water Reactors," November 1979.

3. Letter from J. H. Taylor, B&W, to R. J. Mattson, NRC, transmitting report entitled, "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 Fuel Assembly Plants, "Volumes I and II, May 7, 1979.

4. NUREG-0565, "Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior for Babcock and Wilcox Operating Plants," January 1980.

[illegible]

PROFESSIONAL QUALIFICATIONS

I am Section Leader of the Systems Analysis Section, Analysis Branch, Division of Systems Safety. I have held this position since 1975 and am 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. From June through December 1979, I was assigned to the Bulletins and Orders Task Force as a member of the Analysis Group. I served as Alternate Group Leader and coordinated 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.

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 Laboratories 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 Section 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.