UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

# NRC STAFF TESTIMONY OF PHILIP R. MATTHEWS ADEQUACY OF THE PRESSURIZER AND PRESSURIZER RELIEF TANK SIZE

(Board Question 21)

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

A.1. My name is Philip R. Matthews. I am an employee of the U. S. Nuclear Regulatory Commission assigned to the Auxiliary Systems Branch, Division of Systems Safety, Office of Nuclear Reactor Regulation. My position in the ASB is Section Leader responsible for the technical supervision of engineers conducting technical review of nuclear plant auxiliary systems.

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

A.2. Yes. A copy of this statement is attached to this testimony.



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Q.3. Please state the purpose of this testimony.

A.3. The purpose of this testimony is to respond to Castro-Hursh Contention 21:

Board Question 21

Rancho Seco, being a Babcock and Wilcox designed reactor, has a pressurizer tank and quench tank which are of inadequate size to accommodate the volume of gas or liquid that may be required to be stored in the event of a loss of feedwater transient, and therefore is unsafe and endangers the health and safety of Petitioners, constituents of Petitioners and the public.

Q.4. Describe the Rancho Seco pressurizer tank and pressurizer quench tank systems and functions.

A.4. The Rancho Seco pressurizer tank[1] maintains the reactor coolant system (RCS) pressure within a prescribed range during steady state operation and limits pressure changes during transients. The Rancho Seco pressurizer quench tank[2] (pressurizer relief tank, PRT) condenses, cools, and collects the steam discharged from the pressurizer electromatic relief valve (power operated relief valve, PORV) and 2 code safety valves.

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[1] The pressurizer tank, referred to hereafter as the pressurizer, is part of the reactor coolant system and is a vessel having the same design pressure as the RCS

[2] The pressurizer quench tank, referred to hereafter as the pressurizer relief tank (PRT), is a low pressure design vessel inside primary containment but is not considered part of the reactor coolant system.

The pressurizer is a vertical cylindrical vessel with a bottom surge line penetration connected to the reactor coolant system (RCS) piping at the reactor outlet. The pressurizer contains removable electric heaters in its lower section and a water spray nozzle in its upper section to maintain RCS pressure within desired limits.

An increase in plant electrical output results in a temporary decrease in average RCS coolant temperature and a contraction of RCS coolant volume which causes an outsurge from the pressurizer. During outsurges, as the RCS pressure decreases, some of the pressurizer water flashes to steam, thus assisting to maintain RCS pressure. The electric heaters are then energized to heat the pressurizer water to its initial saturation temperature and thus restore normal operating pressure.

A decrease in plant electrical output results in a temporary increase in average RCS temperature with an increase in RCS coolant volume which causes an insurge to the pressurizer. During insurges as system pressure increases, water from the reactor vessel inlet piping is sprayed into the steam space to condense steam and reduce pressure. Spray flow and heaters are controlled by the pressure controller. The pressurizer water level is controlled by the level controller.

Since all sources of heat in the system, the reactor core, pressurizer heaters, and reactor coolant pumps are interconnected by the RCS piping with no intervening isolation valves, overpressure protection is provided on the pressurizer. Overpressure protection consists of two ASME code safety valves and the PORV.

The pressurizer relief tank (PRT) is a vessel located inside the containment which condenses, cools and collects steam discharged from the pressurizer electromatic relief and code safety valves. After the PRT receives PORV and/or safety valve effluent, the tank's contents are cooled to normal temperatures by the component cooling water system. The PRT has a relief valve which vents to the flash tank of the coolant radwaste system. The flash tank is located outside the containment. This line is automatically isolated if the RCS pressure drops to 1600 psig or the containment pressure rises to 4 psig. The PRT is further protected against overpressure by a rupture disc sized for the total combined relief capacity of the two pressurizer code safety valves and the PORV. If the disc should rupture, the PRT contents would discharge into the containment. See Question 6 below for further discussion of containment discharge.

Q.5. What are the NRC staff criteria for an acceptable size of the pressurizer and pressurizer relief tank (PRT)?

A.5. The NRC staff criterion applicable to the pressurizer is General Design Criterion 15 of Appendix A to 10 CFR Part 50:

> "The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant system pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences."

The NRC staff criterion applicable to the PRT size is stated in Section 5.4.11 of the NRC Standard Review Plan (NUREG-75/087) "Pressurizer Relief Tank." The criterion states that the PRT volume and the quantity of water initially stored in the tank should be such that no steam or water will be released to containment under any normal operating conditions or anticipated abnormal occurrences.

Q.6. Do the Rancho Seco pressurizer and PRT meet these criteria?

A.6. Yes. In this discussion, the pressurizer, and associated heaters, spray, PORV, code safety valves and PRT are considered as an integrated pressure control system.

1) This pressure control system meets the requirements of GDC 15. Section 4.2.4.1 of Reference 1 states that the pressurizer code safety valve capacity is determined on the basis of the maximum pressure transient imposed on the reactor coolant system. The safety valves are sized to prevent a pressure in excess of 110 per cent of RCS design pressure at the highest pressure point in the system (RC pump discharge) when the code safety valves are relieving at 100 per cent capacity at 103 per cent of set pressure. The following events were considered in determining the

maximum pressure transient: (1) control rod withdrawal, (2) turbine trip, (3) complete loss of power, and (4) loss of feedwater flow. Control rod withdrawal from zero power results in the maximum pressure increase and is the limiting case anticipated maximum pressure transient.

Section 4.2.4.2 of Reference 1 states that the PORV was sized to maintain the RCS pressure below the reactor high pressure trip set point for expected operating transients including (1) step and ramp load changes at specified rates and power changes, (2) turbine trip, (3) full load electrical output load rejection, and (4) reactor trip. Its capacity is also based on proper sequencing of the normal control functions of the ICS, turbine bypass system, turbine system and feedwater system during the normal transients. Subsequent to TMI-2 in accordance with Item 3 of Reference 2, the set point of the PORV was raised from 2255 psig to 2450 psig and the reactor high pressure trip set point was reduced from 2355 psig to 2300 psig. This action was taken to reduce the likelihood of automatic actuation of the PORV and thus the likelihood of it sticking open. However, this revised set point of the PORV does not cause the RCS pressure boundary design conditions to be exceeded during normal operation or anticipated operational occurrences. Thus, the setpoints and the capacity of the PORV and of the code safety valves with the existing pressurizer size assure that GDC 15 is met.

2) The Rancho Seco PRT meets the sizing criteria of Standard Review Plan Section 5.4.11. Section 4.2.4.5 of Reference 1 states that effluent from the PORV and code safety valves discharges into the PRT which condenses and collects the effluent. The PRT is sized to accept the combined discharge, without release to containment, of both the relief and safety valves which would result from anv of the plant transients indicated above which were used for the sizing of the capacity of the valves.

The PRT, however, is not sized to accommodate the continuous flow from a PORV or a safety valve which might stick open following termination of an anticipated transient. However, the PRT is protected against overpressurization in such an event by a rupture disc which is sized for the combined relief capacity of the two code safety valves and the PORV. RCS liquid and gases would then be exhausted into the containment. The containment automatically isolates when the RCS pressure drops to 1600 psig or the containment pressure rises to 4 psig. In the event the PORV should stick open, the operator can then close the block valve upstream of the PORV to stop the flow. Also, as a result of TMI-2 and discussed in Question 8 below, modifications have been made which reduce the likelihood of PORV operation following transients and which provide the operator better control of the PORV and its block valve. Also, opening of the safety valves will not occur for either loss of feedwater or turbine trip transients even if the PORV fails to open at the new setpoint. In the unlikely event that a safety valve opened and stuck open, a small break LOCA would result. Analysis for this case shows that the core can remain covered and be adequately cooled as discussed in the NRC Staff Testimony of Paul Norian in response to Board Question CEC 1-4.

Q.7. With specific reference to the Three Mile Island Unit 2 (TMI-2) incident, do the size of the pressurizer and PRT at the Rancho Seco facility pose a safety concern to the NRC staff? If so, what is the nature of the concern?

A.7. As discussed in Question 6 above, the pressurizer and PRT meet the established size criteria as related to providing overpressure protection and are not considered a safety concern. However, since Rancho Seco is a B&W reactor system, the size of the pressurizer can be an indirect concern related to the safety concern described in Reference 2. Reference 2 describes the high sensitivity of RCS pressure and pressurizer level responses to possible RCS overcooling due to changes in feedwater flow rate or temperature in conjunction with the once-through steam generator design. See the "NRC Staff Testimony of Mark P. Rubin and Thomas M. Novak Regarding the Sensitivity of the Once-Through Steam Generator Design.

Q.8. What steps have been taken at the Rancho facility following the TMI-2 incident relative to the sizing of the pressurizer and the PRT?

A.8. No steps have been taken to directly change their size; however, action has been taken at Rancho Seco that affects functions which are related to the size of the pressurizer and PRT and are summarized below:

1) The following actions were taken to mitigate the effect of transients which could result in reductions in RCS pressure or abnormal pressurizer level excursions:

   a) As evaluated in Reference 3 and directed by item 4d of IE Bulletin 79-05A, the licensee has provided additional guidance and training to operators 1) to make use of available RCS indications in addition to pressurizer level in order to determine RCS water inventory as part of the loss of coolant procedure, and 2) to use the pressure/temperature relationship of the RCS to assure proper RCS subcooling prior to securing high pressure injection.

   b) As evaluated in Reference 3 and directed by Item 1 of IE Bulletin 79-05B, the licensee established procedures and trained operators in methods of establishing and maintaining primary system natural circulation which includes maintaining sufficient pressurizer level and heater capacity to maintain RCS subcooling.

2) The following actions were taken to reduce the potential for automatic operation of the PORV following transients thus reducing or eliminating discharge of RCS liquids or gases to the PRT:

   a) As evaluated in Reference 3 and directed by Item 3 of IE Bulletin 79-05B, the licensee raised the PORV set point from 2255 psig to 2450 psig and reduced the reactor high pressure trip setpoint from 2355 psig to 2300 psig.

   b) As evaluated in Reference 3 and directed by Item 4 of IE Bulletin 79-05B, the licensee modified procedures and trained operators to manually trip the reactor following transients that result in increases in RCS pressure.

   c) As evaluated in Reference 4 and required by item (c) of the Commission Order of May 7, 1979, the licensee installed a control grade reactor trip that is actuated on loss of main feedwater and/or turbine trip.

Q.9. Based on the staff's review, are any further steps planned for the Rancho Seco facility relative to sizing of the pressurizer and the PRT? If so, what are these steps and when will they be implemented?

A.9. No steps are planned to directly change the size of the pressurizer and PRT; however, action is planned at Rancho Seco that will affect functions which are related to the size of the pressurizer and PRT and are summarized below:

1) Item 2.1.1 of Reference 5 (NUREG-0578 - TMI-2 Lessons Learned) recommended that all PWR plants provide capability to supply emergency power in the event of loss of offsite power to (1) the minimum number of pressurizer heaters required to maintain natural circulation conditions, and (2) the control and motive power systems for the PORV, PORV block valve and pressurizer level indication instrument channels. This will maintain the capability of the pressurizer to control RCS pressure in the event of loss of offsite power, thus decreasing the frequency of challenges to the emergency core cooling systems.

As committed in Reference 6, the licensee will provide the capability to make available power from a diesel generator supplied emergency bus for 126 kw of pressurizer heaters as backup to the power presently supplied from an offsite power source. Also the licensee will shift the power supply for the PORV block valve to a bus which can be supplied by the diesel generator. These modifications will be completed during the refueling outage scheduled for January, 1980. The power supplies for the PORV and pressurizer level instrumentation already comply with Item 2.1.1 since their power comes from a battery and inverter/battery that can be charged by a diesel generator.

2) Item 2.1.2 of Reference 5 recommended that all operating plants commit to provide performance verification by full scale prototypical testing for all RCS relief and safety valves under test

conditions calculated to occur for anticipated operation occurrences, and accident conditions.

As committed in Reference 6, the licensee will participate in the EPRI/NSAC program to conduct qualification testing for design basis accident conditions of PWR relief and safety valves. It is expected that substantive test data can be obtained by July, 1981.

3) Item 2.1.3.a of Reference 5 recommended that all operating plants provide in the control room either a reliable direct position indication for relief and safety valves or reliable flow indication devices downstream of the valves.

   As committed in Reference 6, the licensee will provide PORV and safety valve positive status indication which indicate and alarm in the control room. This modification will be accomplished during the refueling outage scheduled for January, 1980.

4) In compliance with the long term requirements of and in follow up to item (c) of the Commission Order of May 7, 1979, the licensee has proposed a safety grade design of the reactor trip discussed in my response to Question 8 above (items (2)(a) and (c)). By NRC letter from R. Reid to J. J. Mattimoe (SMUD) dated December 20, 1979, the licensee was given preliminary design approval for the proposed upgrade. It is anticipated that final design,

procurement of equipment and installation will take approximately 6 months from the date of our preliminary design approval.

Q.10. For each of the steps identified in response to Question 9 above, explain why the continued operation of the Rancho Seco facility is permissible prior to complete implementation?

A.10. As I have stated, the pressurizer and PRT meet the applicable staff criteria. In addition, as described in my response to Questions 8 and 9 above, a number of actions have been taken at Rancho Seco to enhance the performance of the pressure control system. In my judgment, these actions are considered sufficient for the continued safe operation of the plant. However, additional modifications have been directed by the staff which can be reasonably implemented and can provide additional safety margin. These modifications are being implemented consistent with staff schedule requirements.

References


1. Amendment 20 to Rancho Seco FSAR.
2. NRC Memorandum dated November 16, 1979, from D. Eisenhut to S. Scott, Subject: "Rancho Seco Board Notification 10 CFR 50.54 Request Regarding Design Adequacy of Babcock and Wilcox NSSS."
3. NRC letter dated November 23, 1979, to SMUD, "Evaluation of Licensee Responses to IE Bulletins 79-05A and 79-05B, SMUD, Rancho Seco Nuclear Generating Station, Docket No. 50-312.
4. NRC letter dated June 27, 1979, H. Denton, NRC to J. J. Mattimoe, SMUD - transmits Notice of Authorization to Resume Operation.
5. NUREG 0578 - "TMI Lessons Learned Task Force Status Report and Short Term Recommendations."
6. SMUD letter dated October 18, 1979, J. J. Mattimoe, SMUD to D. Eisenhut, NRC.

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.