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December 11, 1981

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Washington, D.C. 20555

Dr. John H. Buck
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Christine N. Kohl, Esquire
Atomic Safety and Licensing Appeal Board
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Washington, D.C. 20555

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

Administrative Judges Rosenthal, Buck and Kohl:

Pursuant to the Atomic Safety and Licensing Appeal Board's Memorandum and Order, ALAB-655, 14 NRC ____ (October 7, 1981), please find enclosed: "SMUD's Responses to ALAB-655" and "Licensee's Memorandum of Law in Association with its Responses to the Information Requests in ALAB-655."

Licensee has not responded to information request items 5 and 7, which are directed only to the NRC Staff.

Respectfully submitted,

Thomas A. Baxter

Thomas A. Baxter
Counsel for Licensee

TAB:jah
cc: per Certificate of Service

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING APPEAL BOARD

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

CERTIFICATE OF SERVICE

I hereby certify that copies of "SMUD's Responses to ALAB-655" and "Licensee's Memorandum of Law in Association with its Responses to the Information Requests in ALAB-655" were served this 11th day of December, 1981 by deposit in the U.S. mail, first class, postage prepaid, to the parties identified on the attached Service List.

Thomas A. Baxter

Thomas A. Baxter

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING APPEAL BOARD

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

SERVICE LIST

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December 11, 1981

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

'81 DEC 14 P3:10

BEFORE THE ATOMIC SAFETY AND LICENSING APPEAL BOARD

OFFICE OF SECRETARY
DOCKETING & SERVICE
BRANCH

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

SMUD'S RESPONSES TO ALAB-655

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING APPEAL BOARD

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

AFFIDAVIT OF ROBERT A. DIETERICH

City of Washington)
: ss
District of Columbia)

ROBERT A. DIETERICH, being duly sworn according to law, deposes and states as follows:

1. My professional qualifications are set forth in the evidentiary record of this proceeding in my testimony, following Tr. 1988. My position with the Sacramento Municipal Utility District, however, has changed since the hearing. I am now Supervisor of Nuclear Licensing and Environmental Engineering in the Generation Engineering Department.

2. The information provided in SMUD's responses, dated December 11, 1981, to the Atomic Safety and Licensing Appeal Board's requests for information contained in an October 7, 1981 Memorandum and Order (ALAB-655), was prepared by me or under my supervision by Babcock & Wilcox personnel,

and is true and accurate to the best of my knowledge and belief.

Robert A. Dieterich

Robert A. Dieterich

Subscribed to and sworn before
me this 11th day of December, 1981.

Mela M. Fletcher

Notary Public

My commission expires My Commission Expires October 15, 1986.

INFORMATION ITEM NO. 1: Status reports from SMUD and the staff on the six recommendations in BAW-1564 to enhance AFW safety and reliability.

SMUD RESPONSE

The first question posed by the Appeal Board in its discussion of the reliability of the Rancho Seco auxiliary feedwater ("AFW") system dealt with the status of SMUD's response to the six recommendations contained in CEC Exhibit 3, "Integrated Control System Reliability Analysis," Babcock & Wilcox ("B&W") report BAW-1564. Initially, it should be observed that the recommendations set forth in CEC Exhibit 3 are not primarily concerned with the issue of AFW reliability; rather, these recommendations are geared toward reducing Integrated Control System ("ICS") and balance-of-plant ("BOP") failures which, in turn, could lead to AFW or Engineered Safety Features Actuation System ("ESFAS") actuations. See CEC Ex. 3 at 3-1. Therefore, these recommendations address the second concern with the ICS raised by the Staff following the TMI-2 accident (i.e., that the ICS could cause a loss of main feedwater transient) rather than the first Staff concern, that the ICS could impair the ability of the AFW system in responding to such a transient. (See 13 NRC at 569-570 (I.D., ¶¶22-25), for a delineation of the two Staff concerns.)

The B&W recommendations were drawn from both a failure modes and effects analysis of the ICS and from a review of the field data from all operating B&W plants. In view of the fact that this field data was derived from plants having different secondary system (or BOP) designs, B&W recommended that each recommendation be studied on a plant-specific basis, in order to determine their applicability to any particular plant, prior to implementation. See CEC Ex. 3 at 1-1, 3-1. As Staff witness Capra described in his testimony, SMUD reported on its evaluation of the B&W recommendations to the Staff by letter dated January 21, 1980. Tr. 3702. The status of SMUD's response to each recommendation is addressed separately below.[1/]

1. NNI/ICS power supply reliability.

Prior to the TMI-2 accident and subsequent Staff interest in the ICS, SMUD had identified two areas of concern with respect to the non-nuclear instrumentation ("NNI")/ICS power supplies. The first of the concerns dealt with the number of trips caused by problems with the 120 volt inverters, which were the sole power sources for NNI and the ICS. Inverter reliability was enhanced by the institution of an improved maintenance program and minor equipment changes. Further, Licensee installed automatic bus transfer ("ABT") devices on the NNI and ICS power sources. In the event of a primary power source failure, the ABT automatically shifts to a backup inverter power source.

1/ The B&W recommendations can be divided into two groups: the first three are ICS-related, while the remaining three recommendations are related to BOP equipment. CEC Ex. 3 at 3-1.

Secondly, as a result of SMUD's review of an NNI/ICS power supply transient at Rancho Seco, the NNI internal power sources have been modified. This modification involved the addition of two power supplies, allowing the complete separation of switching and control functions. In addition, fuses were added to all outgoing signals from the NNI. As Staff witness Capra testified, these modifications have increased the reliability of the power sources. Tr. 3703 (Capra).

The above NNI/ICS power supply improvements were planned prior to the performance of B&W's Reliability Analysis (CES Ex. 3), and, in fact, the recommendation on NNI/ICS power supply reliability in the B&W report, while generically applicable to all operating B&W plants, was derived in large part from the experience at Rancho Seco specifically. That experience was accumulated, however, prior to the implementation to the modifications, described above, which improved the reliability of the NNI/ICS power supplies at Rancho Seco. SMUD did review, nevertheless, the information regarding NNI/ICS power supply reliability in CEC Exhibit 3 and has determined that the modifications described above adequately alleviate any concerns with respect to power supply reliability.

2. <u>Reliability of input signals from the Nuclear Instrumentation/ Reactor Protection System ("NI/RPS") to the ICS - specifically the reactor coolant ("RC") flow signal.</u>

At the time of the January 21, 1980 response to the Staff, SMUD was considering two options to upgrade the

RPS flow signal to the ICS: changing the jack or hard-wiring the flow signal to the ICS or, alternatively, utilizing an auctioneered RCS flow input signal to the ICS. SMUD has not yet implemented either of these options, although both are still under active consideration. The reason no further action has been taken to date is that, while this particular recommendation from B&W's review of operating experience with the ICS could result in a spurious trip and therefore impact plant availability, it does not have a significant effect upon plant safety. Further, the RCS flow input signal to the ICS has no effect whatsoever upon the initiation or control of auxiliary feedwater.

3. ICS/BOP system tuning, particularly feedwater condensate systems and the ICS controls.

In its letter of November 7, 1979 requesting that SMUD respond to the B&W report recommendations, the Staff stated that this concern, while related to tuning, appeared to point to more basic, operational problems and, therefore, requested that the following points be addressed: (1) particular operational problems experienced with respect to the ICS; (2) bases for operator intervention in place of automatic ICS action; (3) procedures used by the operator to perform the actions described in (2) above; and, (4) additional operator training. SMUD responded to these requests in its January 21, 1980 letter to the Staff and referenced the

Operating Experience section of the B&W Reliability Analysis in response to Staff request (1), above, as containing information on any such events. Rancho Seco has not experienced any particular operational problems with respect to the ICS.

SMUD identified three operating procedures as providing the bases for, and descriptions of, allowable operator actions to intervene with the automatic operation of the ICS. The procedures provide the primary guidance to the control room operator on the operation of the ICS. These procedures - A.71 Integrated Control, B.2 Plant Heatup and Startup, and B.4 Plant Shutdown and Cooldown - were provided to the Staff as attachments to Licensee's January 21, 1980 letter. Additionally, SMUD's response described the extensive training provided to control room operators during the on-site and simulator training sessions. The training provides detailed study of each major ICS subsystem, including their purpose, function, operating limits and interactions with other subsystems and plant equipment.

SMUD's review of ICS experience during plant operation and of operating procedures and training programs concluded that further actions are not necessary in this area.

4. Main feedwater pump turbine drive minimum speed control - to prevent loss of main feedwater or indication of main feedwater.

In its January 21, 1980 response to the Staff, SMUD indicated that it was considering the purchase of a new main feedwater pump control system. Since that time, SMUD has purchased and installed a new feedwater pump control system.

5. A means to prevent or mitigate the consequences of a stuck-open main feedwater startup valve.

At operations above fifteen percent reactor power, the main feedwater startup valves are in a full-open position and, therefore, a stuck-open valve would not represent a problem during normal operations. SMUD's January 21, 1980 letter described the means available to the control room operator to recognize and respond to a stuck-open main feedwater startup valve at power levels below fifteen percent rated power. SMUD does not believe that design changes, hardware or additional procedural modifications are necessary to respond to this recommendation.

6. A means to prevent or mitigate the consequences of a stuck-open turbine bypass valve.

As with the previous recommendation, SMUD has evaluated this recommendation and has determined that no additional actions are required. SMUD's January 21, 1980 response identified the methods available to the operator to mitigate the consequences of such an event. Further, the response identified an incident during Rancho Seco startup testing in which a turbine bypass valve became stuck-open and was successfully mitigated by actuating the manual isolation valve upstream of the turbine bypass valve.

In summary, SMUD has evaluated and responded to each of the recommendations contained in the B&W Reliability Analysis (CEC Ex. 3). As noted earlier, these recommendations were issued on a generic basis and were not necessarily applicable to all B&W plants. Where SMUD's evaluation has shown that actions in response to the recommendations were applicable and warranted, such actions have been taken. The Staff has reviewed SMUD's responses to the recommendations and, while no specific evaluation of these responses has been issued by the Staff, it is SMUD's understanding that the Staff will require no further actions with respect to the ICS specifically, in that the RPS will terminate plant transients caused by the ICS prior to any safety limits being exceeded.

INFORMATION ITEM NO. 2: Status reports from SMUD and the staff on SMUD's commitments to improve AFW reliability, as described in CEC Exhibit 21 (Enclosure 2).

SMUD RESPONSE

It should be noted that the short- and long-term AFW actions directed by the Commission in its Order of May 7, 1979 (short-term action (a) and the first long-term modification) have already been accomplished. See 13 NRC at 600-601, 634 (I.D., ¶¶ 110, 111, 194).

As the Appeal Board noted (ALAB-655, slip op. at 12), the Licensing Board found that the auxiliary feedwater system at Rancho Seco provides reasonable assurance that the plant can be safely shut down in the event of a loss of main feedwater. 13 NRC at 604-605 (I.D., ¶ 119). The Licensing Board proceeded to observe that "[d]espite its proven and improved reliability through the short term actions the Licensee has committed itself to make . . . additional long term modifications . . . ." Id. The Licensing Board further concluded ". . . that the timeliness and reliability of the AFW feedwater system at Rancho Seco is presently adequate to assure safe operation of the facility and will be further enhanced by completion of the long term modifications." 13 NRC at 605 (I.D., ¶ 120).

CEC Exhibit 21 is a letter of February 26, 1980, from the NRC Staff to SMUD, with the results of the Staff's review of SMUD's auxiliary feedwater reliability analysis and discussion of proposed actions on outstanding AFW items originating from other Staff review efforts. (SMUD's analysis and discussion are in the record as CEC Exhibit 20.) Enclosure 2 to CEC Exhibit 21 represents the Staff's established implementation schedule, as of February 26, 1980, for completing what the Staff viewed to be necessary actions with respect to the continued upgrade of the timeliness and reliability of the Rancho Seco AFW system.

SMUD first responded to the Staff's letter of February 26, 1980, on March 18, 1980, in a letter which is in the record as CEC Exhibit 22. Subsequently, there have been numerous submittals to the Staff on the effort to upgrade the AFW system. The following is the status of the actions identified in Enclosure 2 to CEC Exhibit 21:

PART A: <u>NRC STAFF COMMENTS ON RANCHO SECO RELIABILITY ANALYSIS</u>

1. <u>Revise success criterion defined in Section 1.5.</u>

The disagreement between SMUD and the Staff over the mission success criterion used in the AFW reliability analysis was addressed in testimony before the Licensing Board. <u>See</u> 13 NRC at 603-604 (I.D., ¶¶ 114-117). The Staff position was that the mission success criterion should be revised to include a requirement to deliver AFW to the steam generator before the steam generator boils dry, without regard for the behavior of other systems that are available to protect the reactor. SMUD's position was that the ultimate measure of AFW system reliability is the ability to remove decay heat from the core to prevent core damage.

This disagreement was discussed in the Initial Decision, where the Licensing Board found that: resolution of this controversy is not necessary in order to determine whether the short term actions required by the Commission's May 7 Order provide reasonable assurance that Rancho Seco will respond safely to feedwater transients; the Staff's revision of the mission success criteria for AFW system reliability is not crucial to the decision; in the Staff's opinion the revision of this criterion would probably not change the relative comparability of B&W to either Westinghouse or Combustion Engineering systems. 13 NRC at 603-604 (I.D., ¶ 116). The Licensing Board found that although steam generator dryout is an undesirable event because it results in challenging the plant's safety systems, it is not an event of great safety concern because later restoration of feedwater and/or actuation of HPI assures adequate core cooling under any circumstances. Id. at 604 (I.D., ¶ 117).[1/]

SMUD still considers avoidance of steam generator dryout to be an inappropriate mission success criterion for an AFW reliability analysis, and SMUD has not revised the original analysis to consider such a criterion. The

1/ See also, NRC Staff Proposed Findings of Fact 122 and 123, August 22, 1980.

Staff was notified of this continuing SMUD position in a letter of May 14, 1980, from SMUD to the NRC Staff.

However, a reliability analysis has been performed for the planned upgraded AFW system (see item B.1.a, below) which addresses two aspects of AFW system reliability -- delivery of flow for only automatic operation of the system, and delivery with operator intervention. Delivery of auxiliary feedwater assuming only automatic system actions corresponds to a mission success criterion of preventing steam generator dryout.

2. Revise Section 2.4.2 and AFW system procedures with regard to AFW pump suction and discharge pressure instrumentation.

Section 2.4.2 of the reliability analysis erroneously indicated that AFW pump suction and discharge pressure instrumentation is provided in the control room. The Staff therefore requested that SMUD verify that this discrepancy does not affect the reliability study results and that there are no AFW system procedures that are dependent on such instrumentation.

In a letter to the NRC Staff of May 14, 1980, SMUD reported that the absence of AFW pump suction and discharge pressure in the control room does not significantly affect the unavailability of the Rancho Seco AFW system. SMUD reported that the Rancho Seco fault tree had been requantified to account for the lack of AFW pump suction and discharge pressure indication in the control room, and showed a maximum increase of 7% in an already very low system unavailability.

3. Discussion of ICS-NNI power supply as a potential single failure source.

In a letter to the NRC Staff dated May 14, 1980, SMUD reported that the ICS/NNI power supply was not identified as a potential single failure source for the AFW reliability analysis because this battery-packed 120 VAC power source was assumed to be available for all cases (as stated on page 3 of the report). This simplifying assumption was required by the Staff in order to make the Rancho Seco analysis conform with the Staff's analyses of Westinghouse and Combustion Engineering plants. The SMUD letter further explained the treatment of the integrated control system in the analysis as a single control device.

PART B: NRC STAFF POSITIONS/REQUESTS FOR ADDITIONAL INFORMATION ON AFW SYSTEM OUTSTANDING ITEMS.

1.a AFW Automatic Initiation and Control System.

In its letter to the NRC Staff of March 18, 1980 (CEC Exhibit 22), SMUD reported that it was working toward completion of a safety-grade AFW initiation system. SMUD stated, however, that the magnitude of the task associated with design and procurement of a safety-grade AFW initiation system was substantial -- especially when considered with other Category B (long-term) requirements of NUREG-0578 (TMI-2 Lessons Learned Task Force recommendations). SMUD reported that the 1981 refueling outage was a realistic date for incorporation of the system at Rancho Seco.

Several developments have occurred subsequent to the conclusion of the evidentiary hearings. In a meeting with the NRC Staff on September 4, 1980, SMUD presented its plan for the proposed upgrade of all Rancho Seco AFW system to reflect consideration of all of the applicable Inspection and Enforcement bulletins, and other orders, studies, reports and programs which had been developed subsequent to the TMI-2 accident. Rather than proceed piecemeal, SMUD proposed a single, integrated modification which would result in a safety-grade, single-failure-proof system for AFW initiation which would also control AFW flow to control steam generator level and the rate of steam generator fill. The result of this upgrade will be that the AFW system can be considered an engineered safety feature, and which will include features to insure natural circulation, to prevent over-cooling and over-filling, and to prevent steam line and feedwater line breaks. SMUD informed the Staff at that meeting, and in a letter of October 9, 1980, that hardware could not be procured for this improvement until the first quarter of 1982, and that SMUD did not anticipate installation until later that year. It was noted that this schedule is consistent with the completion of a new building which will house most of the equipment.

NUREG-0737, the so-called clarification of the Staff's TMI Action Plan requirements, was transmitted to SMUD by Staff letter of October 30, 1980. The generic requirement for a safety-grade AFW initiation system included a proposed implementation date of July 1, 1981.

On November 17, 1980, SMUD submitted the conceptual design for the system upgrade, and requested Staff approval prior to a design freeze and the preparation of equipment purchase specifications on December 5, 1980. In that letter, SMUD also informed the Staff that it could not meet the July 1, 1981 NUREG-0737 schedule for safety-grade initiation since the total system upgrade, to be implemented during the first extended outage following equipment delivery in 1982, includes this concept as well as safety-grade control.

In a letter dated January 22, 1981, the Staff approved SMUD's preliminary design for the upgraded AFW system. Subsequently, SMUD has provided the Staff with additional design information. SMUD is to submit a final design by January 1, 1982, for Staff review and approval. Equipment delivery is expected to begin in late 1982, and installation is planned for the first extended outage after the completion of equipment delivery.

1.b <u>Actuate AFW valves FV-2052' and FV-20528. Implement periodic testing procedure for automatic initiation circuitry of AFW pumps.</u>

In its letter of March 18, 1980 (CEC Exhibit 22), SMUD committed to test the AFW flow control valves during the current refueling outage, and to establish and implement a procedure (until the safety-grade control system is installed) for performing channel functional tests of the automatic initiation circuitry of the AFW pumps, using the loss of main feedwater signal, at least every 31 days. Both of these actions were accomplished during the 1980 refueling outage.

1.c Submit test results and analyses which support automatic loading of Pump P-319 on Nuclear Service Bus.

SMUD responded to this Staff request in CEC Exhibit 22. Subsequently, SMUD proposed that as a part of the complete AFW system upgrade, both AFW pumps will load automatically on new diesel generators to be installed at Rancho Seco. Because these new diesel generators will not be delivered in time for installation during an extensive 1982 outage devoted to TMI-related modifications, SMUD will provide, as an interim measure, for the automatic loading of Pump P-319 (the pump without a steam turbine drive) on the emergency bus (existing diesel generators) in the event of a loss of off-site power. This interim modification will be accomplished during the 1982 outage.

2.a AFW Flow Indication.

In its responsive letter of March 18, 1980 (CEC Exhibit 22), SMUD informed the Staff that the 1981 refueling outage was a more realistic schedule for the installation of safety-grade AFW flow indication. Subsequently, in

NUREG-0737, the Staff changed its proposed implementation date to July 1, 1981.

In its letter of November 17, 1980, transmitting to the Staff the preliminary design for the upgraded AFW system, SMUD stated that it was proceeding to implement safety-grade AFW flow indication independently of the overall system upgrade.

In a subsequent letter dated December 15, 1980, SMUD stated that this modification would be completed during the 1982 refueling outage. This remains the schedule, and the outage is expected to start in September, 1982.

On September 8, 1981, SMUD submitted to the Staff the final design for this modification.

2.b <u>Implement periodic testing procedure for performing tests on AFW flow and steam generator level indication.</u>

In its letter of March 18, 1980 (CEC Exhibit 22), SMUD stated that until safety-grade flow indication is installed, SMUD would establish and implement a procedure for performing channel functional tests of the AFW flow indication and OTSG level indication at least every 31 days, when the plant is operating. These test procedures have been established and implemented.

3. <u>System Modification for Periodic AFW Pump Testing.</u>

In a letter to the Staff of May 14, 1980, SMUD committed to provide position indication in the Rancho Seco control room for the motor-operated valve that will replace FWS-055 (a manual valve in the test flow path), to modify the test procedures for full flow AFW testing when the change is completed, and to provide the Staff with final design information. In the letter of March 18, 1980 (CEC Exhibit 22), SMUD stated that it would continue to use the surveillance procedure which requires stationing an operator at flow control valve FWS-055 during pump testing until the full flow test modification is installed.

Final design information was provided to the Staff on September 8, 1981, and the modification is expected to be completed during the outage scheduled to begin in September, 1982.

4. Review procedures and verify that they are adequate for supplying water from both the canal and the plant reservoir.

In CEC Exhibit 22, SMUD committed to review its procedures for providing alternative water sources to the AFW system and to ensure, during the refueling outage then in progress, that the procedures were revised to describe how to obtain water from the Folsom South Canal or the Plant Reservoir. These actions were accomplished during the 1980 outage.

5. Submit Technical Specification modification on AFW system flow path verification.

On April 30, 1980, SMUD proposed a revised Technical Specification related to full flow testing to the steam generator. The revision has been approved by the Staff and implemented.

6. Condensate Storage Tank Level Indication and Alarm.

In its responsive letter of March 18, 1980 (CEC Exhibit 22), SMUD repeated its intention to install safety-grade condensate storage tank level indication and alarms. SMUD confirmed that the safety-grade design will include redundant sensors, detectors, readouts and alarms from the tank to the control room, including power supplies. SMUD also confirmed that Class 1E components will be used except for the alarm annunciators, for which qualified Class 1E equipment is not available.

Control-grade condensate storage tank level indication and alarms were installed at Rancho Seco in response to NUREG-0578. The safety-grade modification is expected to be accomplished during the September, 1982 outage.

7. Submit requested information on AFW endurance test.

The information requested was provided in the SMUD's letter to the Staff of May 14, 1980.

8. [Not applicable.]

9. AFW system operation during loss of all AC power.

While Rancho Seco can be safely shut down in the event of a loss of all AC power, SMUD maintains its position that it is imprudent to provide procedures or to design for such an event because it is beyond the design basis of the plant. See Tr. 2354-2355 (Dieterich).

10. Submit revision to proposed Technical Specification for AFW Limiting Condition for Operation.

The requested proposed changes to the Technical Specifications were submitted to the Staff on April 30, 1980. The modifications have since been approved by the Staff and incorporated into the Rancho Seco technical specifications.

PART C: AFW SYSTEM STANDARD REVIEW PLAN-SECTION 10.4.9

The design information submitted to the Staff by SMUD letters of November 17, 1980, and September 8, 1981, includes an evaluation of the proposed upgraded AFW system for conformance to section 10.4.9 of the Standard Review Plan. SMUD has concluded that the system design conforms to the guidelines of that section of the plan.

PART D: DESIGN BASIS FOR AFW SYSTEM FLOW REQUIREMENTS

SMUD has evaluated the flow requirements for the AFW system to determine that the existing flow capacity is adequate to meet the system requirements. A copy of the evaluation is attached.

RANCHO SECO AUXILIARY FEEDWATER FLOW EVALUATION

The design basis event for sizing the Auxiliary Feedwater System (AFWS) is Loss of Feedwater (LOFW) with a concurrent Loss of Offsite Power (LOOP), and subsequently loss of reactor coolant pumps. The pertinent parameters for this accident relative to the AFWS are design flowrate and required time to full AFWS flow. These parameters reflect the functional requirements of the AFWS to a) remove decay heat, and b) provide a smooth reactor coolant flow transition from RC pump operation to natural circulation. The design values which resulted from this analysis are 780 gpm deliverable to the steam generators within 40 seconds of the initiation signal. The 40 second time was chosen to allow the AFWS to inject feedwater and begin increasing SG level to the 50% operating range level, required for natural circulation, prior to completion of the RC pump coast-down. At that time, the design flowrate was selected to be equal to or greater than the decay heat generation rate. Since decay heat rate changes with time, other values than 40 seconds and 780 gpm could have been used and been acceptable. All other transients which either require or assume the availability of AFW in the Safety Analysis use the design values derived from the LOFW analysis. The results of these other analysis are acceptable and are referenced in Table 1, attached.

Subsequent to this original analysis, additional analysis was done indicating that a required AFW flowrate of 760 gpm was sufficient to meet the decay heat generating at time of AFW initiation. These results were described in the B&W, May 16, 1979 letter to the NRC, following the Three Mile Island accident.

Accidents 1, 2 and 3 of Table 1, which specifically require AFW for mitigation, were analyzed using the original AFWS performance criteria established by the LOFW accident. The results of these analyses were acceptable and are described in the FSAR sections noted in Table 1. The other accidents listed in Table 1 (4-12) do not require AFW for mitigation though the availability of the AFWS, a defined by the performance criteria established by the LOFW accident, is assumed. The results of those analyses were acceptable and are described in the FSAR sections noted in Table 1.

The accidents listed in Table 1 have not been reanalyzed using the revised AFW flow requirement (760 gpm). However, more recent analysis on identical plants indicate a significantly lower flowrate is adequate for all accidents addressed in the Rancho Seco FSAR.

Addressing the events included in the NRC letter of February 26, 1980, which have not been included in Table 1, we have the following comments:

LMFW w/Loss of Onsite and Offsite AC Power - This event was not a design basis of the plant and subsequently is not included in Chapter 14 of FSAR. The B&W Report, "Auxiliary Feedwater Systems Reliability Analyses" (BAW-1504) indicates however, that the SMUD AFW System will provide injection under these conditions.

Plant Cooldown - Plant cooldown with AFW is a new issue as stated in Reg. Guide 1.139 and not a design basis for this plant. The NRC has not indicated how Reg. Guide 1.139 is to be applied to operating plants. The extent of plant cooldown for which the AFWS is designed is discussed in FSAR Section 14.1.2.8.4D.

Turbine Trip with and without Bypass - This event does not affect the AFWS unless MFW fails, in which case the loss of MFW event previously addressed would bound the AFWS design.

Main Steam Isolation Valve Closure - Again, this event does not directly affect the AFWS unless MFW is lost as discussed above.

Main Feed Line Break - This event was not a require analysis for this plant and is not included in FSAR Section 14. Main Feedline Break is a more abrupt case of LOFW and results of an analysis would be approximately the same.

Small Break LOCA - The AFW criteria assured for this event is described in Topical Report BAW-10052 updated by letter report, J. H. Taylor (B&W) to S. A. Varga (NRC), 7/18/78, and B&W report entitled, "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 FA Plant", 5/7/77.

The RCS cooling rate is not a limit relative to accident acceptance criteria. The safety limit for all transients which use AFW for mitigation is that the core remain cooled with ultimate acceptance criteria being those addressed in Table 1. For transients which result in draining the pressurizer or for which natural circulation is slowed or interrupted, restoration of pressurizer level and subcooling is accomplished by swelling due to core heat input and inventory restoration by HPI.

Steam Generator level is not based on decay heat removal rate or cooldown capability. SG level is set low for decay heat removal and high for natural circulation. It is also set high for a small LOCA as described in Topical Report BAW-10052, and in the B&W report, "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks".

As discussed above, the design basis event regarding AFWS design requirements is loss of main feedwater with concurrent loss of RC pumps; the analysis assumptions for this event are listed below. Corresponding technical justification, where not specifically listed below, is based on licensing requirements and prudent engineering judgement at the time of the analysis.

a) Maximum Rx Power - 100%

b) Time Delay Initiating Event to Rx Trip - The reactor will trip on high RCS pressure approximately 5-10 seconds after a LOFW event. The initiation signal for AFW is loss of main feedwater.

c) AFWS Initiation Signal and Time Delay - The AFW initiation signal for the LOFW event is loss of both main feed pumps as sensed by steam inlet valve positions on the two main feed pump turbines. The design basis time delay from initiation event to full flow of AFW flow into SG is 40 seconds.

d) SG Level at Initiation Event - Steam Generator Inventory is dependent on power level. In all cases, AFW flow within 40 seconds will avoid steam generator dryout.

e) SG Inventory and Decay Heat - For discussion of water inventory see d) above. Reactor decay heat rate is shown in FSAR Table 14, 1-14.

f) Maximum SG Pressure - 1103 psig.

g) Minimum Number of SG - The number of generators was not specified in the analysis, heat removal capability is the pertinent parameter and can be accommodated by one SG.

h) RC Flow Condition - Both natural circulation and RC pump operation were analyzed.

i) Maximum AFW Inlet Temperature - The maximum AFW inlet temperature assumed was 90°F.

j) Steam, Feedline Break Time Delay - The feedwater line break was not a required analysis for this plant. Refer to FSAR Section 14.2.2.1 for steam line break analytical information.

k) Main Feedline Volume and Temperature Between SG and AFWS - N/A - There is no piping connection between the MFWS and AFWS.

l) SG Normal Blowdown - N/A - The OTSG's do not have a blowdown system.

m) Water and Metal Sensible Heat Used - Plant Cooldown was not considered in the design basis analysis. $1x10^6$ BTU/°F was used for the water and metal sensible heat from normal full power Tave to the post-trip Tave setpoint.

n) Time at Hot Standby, etc. Relative to AFW Inventory - The AFW inventory was sized for decay heat removal for the day after Rx trip as discussed in FSAR Section 14.1.2.8.4D. The design basis for AFWS is not plant cooldown; the NRC Reg. Guide 1.139 requirements for operating plants have not yet been established.

TABLE 1

	ACCIDENT DESCRIPTION	FSAR SECTION	ACCEPTANCE CRITERIA[1]
1)	Loss of Coolant Flow	14.1.2.6	A, B
2)	Loss of Electric Power	14.1.2.8 & 14.3.2	A, B, D
3)	Steam Line Break	14.2.2.1 & 14.3.3	D
4)	Uncompensated Operating Reactivity Changes	14.1.2.1	A, B
5)	Startup Accident	14.1.2.2	A, B
6)	Rod Withdrawal Accident at Rated Power Operation	14.1.2.3	A, B
7)	Moderator Dilution Accident	14.1.2.4	A, B
8)	Cold Water Accident	14.1.2.5	A, B
9)	Stuck-Out, Stuck-In, or Dropped Control Rod Accident	14.1.2.7	A, B
10)	Steam Generator Tube Failure	14.2.2.2 & 14.3.4	B, D
11)	Rod Ejection Accident	14.2.2.4 & 14.3.7	C, D
12)	Loss of Coolant Accident	14.2.2.5 & 14.3.8	D, E

NOTE: (1)

KEY	ACCEPTANCE CRITERIA	TECHNICAL BASIS
A	Max. RCS Pressure - 110% Design	ASME Code
B	DNB > 1.3 with BAW-2	SRP 4.4
C	280 Cal./Gram Fuel Limit	Reg. Guide 1.77
D	Acceptable Doses	10CFR100
E	Fuel Cladding < 2200°F	10CFR50.46

INFORMATION ITEM NO. 3: Status reports from SMUD and the staff on the installation of the safety-grade anticipatory reactor trip.

SMUD RESPONSE

The second general area of inquiry raised by the Appeal Board dealt with the status of SMUD's commitment to install a safety-grade anticipatory reactor trip upon loss of main feedwater and/or turbine trip. Specifically, the Appeal Board questioned whether the trip had been installed (noting that, at the time of the Licensing Board hearing, the trip was due to be installed in approximately June, 1980), and, if not, to explain the basis for the delay and provide a projected completion date.

As noted by the Appeal Board, the Commission's May 7, 1979, Order required, as a short-term item, that SMUD implement a hard-wired, control-grade anticipatory reactor trip upon loss of main feedwater and/or turbine trip. ALAB-655, slip. op. at 14. The Commission's Order also required, as a long-term item, that the anticipatory reactor trip be upgraded to safety-grade as promptly as practicable. Id. at 15. The control-grade anticipatory trip was installed prior to the restart of Rancho Seco in July, 1979, and, as recognized by the Licensing Board, has been successfully operated and tested since that time. 13 NRC at 582. (I.D, ¶56). SMUD has not yet installed the safety-grade anticipatory reactor trip at Rancho Seco, but anticipates doing so during the extended outage for TMI-2 modifications currently scheduled to begin in September, 1982. Several factors have combined to cause this delay from the originally projected completion date, as set forth below.

The installation of the safety-grade trip was initially delayed by the need to perform a new seismic analysis of the Reactor Protection System cabinets, due to the additional mass of equipment being added to these cabinets by this modification. As stated in SMUD's July 21, 1980 letter to the Staff,[1/] the seismic analysis would require twenty weeks to perform, thereby delaying the implementation until the first outage of sufficient duration following completion of the seismic analysis.

In October of 1980, the Staff issued NUREG-0737, "Clarification of TMI Action Plan Requirements," which revised earlier post-TMI requirements imposed by the Staff, including the safety-grade anticipatory reactor trip requirement. Under Item II.K.2.10 of NUREG-0737, B&W licensees were to submit their final design for the safety-grade trip by January 1, 1981, with installation due by July 1, 1981. SMUD responded to NUREG-0737 by letters to the Staff dated December 15, 1980 and February 26, 1981; in these responses, SMUD committed to provide the Staff with final design information by October 1, 1981 and to install the safety-grade anticipatory reactor trip during the April, 1982 outage. SMUD's December 15, 1980 letter provided the following justification for delaying the modifications:

1/ A copy of this letter was served on the Licensing Board and parties by letter dated July 22, 1980 from Thomas A. Baxter, counsel for Licensee. A copy of the letter is attached for the information of the Appeal Board.

(1) The July 1981 installation date coincides with the period of peak electrical demand in Northern California and it would not be advisable to remove Rancho Seco from service at that time.

(2) Design efforts and equipment delivery schedules for the TMI-2 modifications would preclude installation prior to January 1982. Equipment delivery delays are being encountered due to SMUD's attempt to purchase equipment qualified to the criteria set forth in NUREG-0588.

(3) Due to the large number of post-TMI equipment modifications being required, SMUD must construct a Nuclear Service Electrical Building at Rancho Seco to house the modifications. Construction of this building will not be completed until late 1981.

Upon review by the Staff of SMUD's responses to NUREG-0737, the Commission, on July 10, 1981, issued an order requiring, among other actions, that SMUD submit its final design for the safety-grade anticipatory reactor trip to the Staff within sixty days. In accordance with this order, SMUD submitted its final design to the Staff on September 8, 1981; additional information was provided in response to Staff questions on October 19, 1981. On November 2, 1981, the Staff issued its Safety Evaluation accepting

SMUD's safety-grade anticipatory trip design. The modifications will be implemented during the 1982 outage, which has been rescheduled from April to September to allow for generating capability during the peak usage season, to meet the fuel cycle reload criteria and to coincide with additional delays in equipment delivery schedules.

It is SMUD's opinion that the delay encountered in implementing the safety-grade anticipatory reactor trip should not be viewed in isolation, but in the context of all of the post-TMI requirements imposed by the Staff. These post-TMI modifications have resulted in the required installation of extensive new equipment which has, in turn, resulted in the need for additional construction at the Rancho Seco site. SMUD has moved forward in implementing these modifications as promptly as practicable in view of other considerations (i.e., generating capacity, etc.) which must be taken into account. Further, as the Licensing Board found, there are no adverse safety implications for the operation of Rancho Seco with the hard-wired control-grade anticipatory reactor trip prior to the upgrade to safety-grade. 13 NRC at 582, (I.D., ¶57).

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July 22, 1980

Elizabeth S. Bowers, Esquire
Chairman
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dr. Richard F. Cole
Atomic Safety and Licensing
Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Mr. Frederick J. Shon
Atomic Safety and Licensing
Board Panel
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

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

Dear Mrs. Bowers and Members of the Board:

Please find enclosed, for your information, a letter, dated July 21, 1980, from Licensee Sacramento Municipal Utility District (W. C. Walbridge) to the NRC Staff (R. W. Reid), which includes information relevant to the matters before the Board.

Sincerely,

Thomas A. Baxter

Thomas A. Baxter
Counsel for Licensee

TAB:jah
cc: Service List attached

DUPE OF
8007250345

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
	)	
(Rancho Seco Nuclear Generating Station)	)	
	)	

SERVICE LIST

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Atomic Safety and Licensing Board
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Washington, D.C. 20555

SMUD
SACRAMENTO MUNICIPAL UTILITY DISTRICT ☐ 6201 S Street, Box 15830, Sacramento, California 95813; (916) 452-3211

July 21, 1980

Director of Nuclear Reactor Regulation
Attention: Mr. Robert W. Reid, Chief
Operating Reactors, Branch 4
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Docket No. 50-312
Rancho Seco Nuclear Generating
Station, Unit No. 1

Dear Mr. Reid:

On October 5, 1979, the Sacramento Municipal Utility District provided you with information concerning the safety grade anticipatory reactor trip to be installed at Rancho Seco Unit No. 1. In that letter we indicated that we felt this trip could be installed within 26 weeks of design approval. Your letter of December 20, 1979, provided approval of our preliminary design.

It has been determined that the equipment being added to the reactor protection system cabinets is of sufficient mass to require a new seismic analysis of these cabinets. We presently estimate 20 weeks for the completion of this seismic analysis and will, therefore, be unable to install this equipment until that time. We still intend to install this modification during the first outage of sufficient duration following completion of this analysis. As requested in your letter of December 20, 1979, the District will provide the additional information required in sufficient time to allow staff approval of the final design prior to system operation. If you have any further questions on this matter, please advise.

Sincerely,

Wm. C. Walbridge

Wm. C. Walbridge
General Manager

DUPE OF
8007250248

INFORMATION ITEM NO. 4: Status reports from the staff and SMUD on the need for the additional analyses identified in the Staff Evaluation at 19, 23.

SMUD RESPONSE

This portion of the Appeal Board inquiry directs SMUD to provide a status report on the additional LOCA analyses identified in the Staff Evaluation on pages 19 and 23. The identified analyses are: (1) the more detailed small break loss-of-coolant accident (LOCA) analyses discussed in Sections 8.4.1 and 8.4.2 of NUREG-0560, and (2) (a) confirmation that the AFW can be restored in a reasonable period of time, and (b) description of the thermal-mechanical behavior of vessel materials under small break LOCA conditions.

Prior to describing how SMUD addressed the additional analyses identified in the Staff Evaluation, it is important to note that both NUREG-0560 and the Staff Evaluation were issued shortly after the TMI-2 accident, in May and June of 1979, respectively. Since that time, and as more particularly described below, SMUD has performed, and is performing, additional small-break LOCA analyses to satisfy subsequent Staff recommendations and has undertaken other actions related to the issues described above. Therefore, the additional analyses identified by the Appeal Board must be viewed as only a part of the larger effort currently underway in the small-break LOCA area.

1. Analyses identified in NUREG-0560.

As part of the Commission's May, 1979 Order, SMUD was directed to "[c]omplete analyses for potential small breaks and develop and implement operator instructions to define operator action." In the Staff Evaluation, it was concluded that SMUD had complied with this portion of the Order. However, the Staff Evaluation also concluded that the small break analyses discussed in Sections 8.4.1 and 8.4.2 of NUREG-0560 needed to be performed in order to support longer term operation of the facility. Licensee's Testimony of Bruce A. Karrasch and Robert C. Jones, ff. Tr. 535, includes, in Section 10 entitled Small-Break Loss-of-Coolant Accidents, a description of the small-break LOCA analyses which were performed for Rancho Seco. These analyses satisfy the small break LOCA analysis recommendations of Sections 8.4.1 and 8.4.2 of NUREG-0560. The subsequent paragraphs summarize the analysis recommendations of these sections in NUREG-0560 and describe how the analyses discussed in the testimony satisfies them. Additionally, further actions currently underway for SMUD which are related to the NUREG-0560 recommendations are discussed in the testimony.

In Section 8.4.1 of NUREG-0560, the Staff recommended that the analysis of feedwater and other transients should be performed for conditions beyond the design basis of the plant. It also recommends that these analysis results be incorporated within the plant emergency procedures. Specific recommendations in this section related to small break analyses are:

a. The failure of a PORV should be analyzed.

b. The effect of delayed or no auxiliary feedwater should be analyzed.

c. Studies to clearly define the significance of the steam generators as a heat sink for all breaks should be performed (i.e., can adequate core cooling be maintained via "bleed-and-feed" in the RCS using the HPI and PORV).

d. The effect of degraded heat transfer in the steam generators due to the presence of non-condensible gases in the system should be studied.

The small break LOCA analysis requirements of Section 8.4.2 of NUREG-0560 are similar to those of Section 8.4.1. The main emphasis of Section 8.4.2 is the recommendation to perform additional analyses of breaks smaller than 0.05 ft^2.

In the Karrasch and Jones Testimony, issues related to small-break LOCA were addressed in Section 10. Presented within the testimony was a summary of the analyses which had been performed subsequent to the TMI-2 accident. These analyses, which include analyses performed following the issuance of the Staff Evaluation, satisfy the recommendations of Sections 8.4.1 and 8.4.2 of NUREG-0560. To help illustrate this, a comparison of the NUREG-0560 recommendations and the analyses described in the testimony is provided below.

Analyses of PORV failures which were performed are summarized in Tables 3 and 4 of the Karrasch and Jones Testimony. These analyses include a loss of main feedwater with the assumption of a PORV failure following its actuation (consequential failure) and the failure of a PORV with the assumption of a loss of all (main and auxiliary) feedwater. This directly satisfies the PORV analysis recommendation of Section 8.4.1 of NUREG-0560.

As stated previously, Section 8.4.1 of NUREG-0560 recommended analyses of small break LOCAs with the assumption of delayed or no auxiliary feedwater. Such analyses have been performed and are described in Tables 1 and 2 of the Karrasch and Jones Testimony. These analyses demonstrated that auxiliary feedwater is not significant f[illegible] larger sized small-breaks and, in fact, demonstrated [illegible] adequate core cooling can be provided in a "bleed-and-fee[illegible] using the HPI.

The effect of non-condensible gases on the steam generator heat removal during small break LOCAs, a recommendation of NUREG-0560, was also addressed in the Karrasch and Jones Testimony at 47. It was found that noncondensible gases would not significantly impair the steam generator heat removal during a small break LOCA.

The recommendation of Section 8.4.2 of NUREG-0560 stated that analyses of break sizes smaller than 0.05 ft^2 should be performed. Tables 3 and 4, describing the analyses of a PORV failure under different assumptions, fall into this category since the effective leak area of a stuck open PORV is 0.007 ft^2. Additionally, the analyses of a small break LOCA with a loss of all feedwater (Table 2) include analyses of break sizes as low as 0.01 ft^2. Analyses of very small break LOCA (0.005-0.01 ft^2) are summarized in Table 5 of the Karrasch and Jones Testimony.

In addition to the analyses described above, additional analyses of small break LOCAs with delayed reactor coolant pump trip are described in the testimony (Table 6). This analysis addresses the effect of a delayed loss of offsite power on a small break accident. NUREG-0560 also recommended that the effects of loss of offsite power be considered in the analysis. Thus, SMUD has also complied with this recommendation of NUREG-0560.

Although it is clear from the discussions above that small-break analyses have been performed which comply with the recommendations of Sections 8.4.1 and 8.4.2 of NUREG-0560, further work is underway at SMUD in these areas. Relative to the recommendation of Section 8.4.1, SMUD is participating in the ATOG (Abnormal Transient Operating Guidelines) program underway at B&W. The purpose of this program is to provide an improved set of emergency operator guidelines for feedwater and other transients based upon plant symptoms. As part of the program, event trees are prepared to address the effect of consequential failures and operator actions and/or errors during the transient. Plant responses for each of these event trees are assessed to assure that proper guidance is given to the operator. These guidelines are presently scheduled to be implemented at Rancho Seco during the extended outage for TMI modifications scheduled to begin in September, 1982.

In the area of small break LOCA analyses, the Staff and SMUD have taken several actions. Based upon the analyses discussed in the testimony, and described above, the Staff issued NUREG-0565, "Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox Designed 177-FA Operating Plants." That document reflects the Bulletin & Orders Task Force review of the small break analyses which were performed subsequent to the TMI-2 accident and recommends that the small break methods be upgraded. During the Licensing Board hearing, extensive cross-examination was conducted based upon this document, which is in the record as Staff Exhibit 2. The recommendations of NUREG-0565 were implemented by the NRC as Item II.K.3.30 of NUREG-0737, "Clarification of TMI Action Plan Requirements." In response to this item, SMUD is participating in a generic Small Break LOCA Methods Program underway at B&W. This program is presently scheduled to be completed in mid-1982.

In summary, the small-break LOCA analyses recommendations of Sections 8.4.1 and 8.4.2 of NUREG-0560 have been performed and were discussed during the Rancho

Seco hearings. In addition, further analyses are continuing in this area in response to Staff recommendations set forth in NUREG-0737.

2.(a). Confirmation that AFW can be restored in a reasonable period of time.

SMUD believes that the Rancho Seco AFW System Reliability Analysis (CEC Exhibit 20), which was submitted to the Staff in December, 1979, is responsive to the statement in the Staff Evaluation, issued in June, 1979, recommending analyses to confirm that auxiliary feedwater, if lost, can be restored within a reasonable period of time.

2.(b). Thermal-mechanical behavior of vessel materials.

As requested by the NRC Staff, and later included in NUREG-0737 as item II.K.2.13, SMUD submitted a report, "Thermal-Mechanical Report - Effect of HPI on Vessel Integrity for Small Break LOCA Event with Extended Loss of Feedwater," BAW-1648, by letter dated January 16, 1981. SMUD feels this report satisfies the Staff's request for an evaluation of the effects of extended use of HPI with a loss of all feedwater (implying a loss of natural circulation flow) following a small break LOCA. However, since the report was generic, it contained several overly conservative assumptions to insure its applicability to all operating B&W plants. In addition, no technical justification existed to determine the amount of mixing which would occur in the reactor vessel downcomer between the cold HPI injection fluid and the hot return flow from the reactor vent valves. As a result, the report concludes that fracture mechanics acceptance criteria of the reactor vessel could be exceeded prior to the design life of the vessel.

The results of this report were discussed during meetings of the utility Regulatory Response Group PWR vendors, and the NRC Staff on March 31 and April 29, 1981, on the generic issue of "Pressurized Thermal Shock." As a result of these meetings, SMUD submitted further information on May 12 and May 15, 1981, showing that immediate corrective actions were not necessary and committing to perform a plant-specific analysis to demonstrate that considerable time exists before reactor vessel brittle fracture is of any concern at Rancho Seco.

INFORMATION ITEM NO. 6: SMUD and staff schedules for HPI analyses.

SMUD RESPONSE

Two of the issues heard by the Licensing Board, Issues CEC 1-1 and 1-12, questioned whether or not the actions directed by the Commission's Order of May 7, 1979, will result in an increase in reactor trips resulting from feedwater transients that will increase challenges to safety systems beyond the original design and licensing basis of the facility. During the oral examination of witnesses at the hearing, interest arose with respect to the number of High Pressure Injection (HPI) thermal cycles permitted on each injection nozzle during the life of the plant. While the Licensing Board expressed concern that the cycling criterion was being approached, the Licensing Board also found that the limit may be overly conservative and that there are several ways to cope with the matter should it become evident that a real safety limit is being approached. 13 NRC at 607 (I.D., ¶ 125).

Expressing the view that the record does not support the Licensing Board's appraisal, the Appeal Board retained jurisdiction to enable supplementation of the record with further information. ALAB-655, slip op. at 18-21. Rather than providing the proposed schedule for supplying the requested information, SMUD provides at this time the following information in response to the Appeal Board's requests.

All Rancho Seco reactor coolant system components are designed to withstand the effects of cyclic loads due to system temperature and pressure changes, as discussed in Section 4.1.2.4 of the Rancho Seco Final Safety Analysis Report. A description of the original design basis transients and the allowable number of each is given in Table 4.1-1 of the FSAR (attached). As discussed during the hearing (Tr. 2015-2017), however, these numbers of allowable transients do not identify the number of transient cycles which components are physically capable of safely withstanding. The numbers are based on design specification and analyses, versus the generally less severe transients which will be actually experienced. Also, the numbers are generated from analyses which consider the integrated effect of the various transients on the various RCS components. That is, a particular transient such as HPI actuation will normally impact multiple RCS components. And, conversely, each RCS component such as an HPI nozzle is affected by more than one of the design basis transients. Therefore, the calculation of the maximum number of a particular transient allowable for a particular component is not a meaningful exercise because the impact of the transient on other components and the impact on the component of other transients would not be considered.

Included in the number of original allowable operating transient cycles were 40 design cycles for rapid depressurization of the reactor coolant system (i.e. depressurization which would result in automatic HPI actuation), and 40 design cycles for HPI testing. Through approximately mid-1980 it was operating practice at Rancho Seco to manually actuate HPI following a reactor trip for the purpose of maintaining reactor coolant volume within its normal operating range. This procedure accounted for approximately 90% of the thermal cycles to which each HPI nozzle had been subjected up to the time of the hearing. Such an operating transient was not specifically included in the original reactor coolant system design specification. Therefore, each post-trip manual initiation of HPI was conservatively inventoried against the 40 allowable rapid depressurization cycles mentioned above.

Since the hearing, two actions have been taken:

1. Operating procedures for manual post-trip coolant system volume control have been changed and operators are now directed to use only the HPI nozzle used for system makeup. Since flow through this nozzle is continuous during normal operation, the nozzle is not subjected to a thermal cycle if injection flow is increased following a trip. In fact, the HPI nozzles have not been subjected to any additional thermal cycles since this procedure was instituted.

2. A reevaluation of the original allowable operating transient cycles design basis was performed in 1980 after the hearing. This effort resulted in including within the total number of allowable reactor trips (400) a specific provision for up to 70 actuations of HPI following such a trip. The dual impact of this action is to both increase the total allowable number of HPI actuations and to reduce the number of cycles previously listed as HPI actuation due to rapid system depressurization.

In summary, it can be seen that the design basis limit for HPI nozzle cycles is not being approached more quickly than anticipated, and that added stress on the nozzles due to an increase in HPI actuations is not likely.

With regard to each of the specific items on which the Appeal Board requested the record to be supplemented, the following information is provided:

1. <u>Maximum allowable number of thermal cycles on the HPI nozzles</u>. As discussed above, the original maximum allowable number of thermal cycles on the HPI nozzles was established in the design basis for Rancho Seco as 40 automatic actuations due to rapid depressurization of the RCS, plus 40 HPI test transients. The design basis has been reevaluated to include an additional 70 allowable cycles due to manual actuation of HPI following reactor trip.

The number of allowable transient cycles could be increased still further. For example, in order to simplify the calculations the design basis assumes that each transient of a particular type is the same. Actual events, however, do not proceed precisely according to these idealized, and generally conservative, assumptions. Therefore, by evaluating the actual transients versus assumed events as allowable limits are approached, a revised, increased maximum number of cycles can be computed. Also, as previously discussed, the current analysis does not represent the number of transient cycles which the nozzles are physically capable of safely withstanding. However, based on the reevaluation of allowable transients which has been performed and the change in operating procedures, also discussed above, to limit normal post-trip use of HPI, it is not considered necessary or appropriate to perform additional analysis at this time to determine a further increase in the number of allowable cycles.

2. Methods of detecting thermal cycle effects on the HPI nozzles. Possible effects of thermal cycles on the HPI nozzles are the initiation and/or increase in the size of flaws. Such effects may be detected by non-destructive examination techniques, and the nozzles are periodically examined as part of the NRC and ASME required inservice inspection programs at Rancho Seco.

3. Possible means of prolonging the useful life of the HPI nozzles. As discussed above, two actions have

already been taken to prolong the useful life of the HPI nozzles: a reevaluation of the design basis calculations and a change in operating procedures. Also, as discussed above, the useful life of the existing nozzles could be further prolonged, if necessary, by additional analysis.

4. Technical Specifications or operating procedures that might reduce the use of HPI without endangering the core. As discussed above, operating procedures have been changed at Rancho Seco such that HPI nozzles are no longer thermally cycled following a normal reactor trip. The presently existing analyses provide an adequate number of available cycles for the remainder of the plant life.

TABLE 4.1-1

OPERATING TRANSIENT CYCLES

Transient Number	Transient Description (ASME Category)		Design Cycles	
1A	Heatup from 70 F to 8% full power (normal)		240	10
1B	Cooldown from 8% full power (normal)		240	
2	Power change 0 to 15% to 0% (normal)		1,440	
3	Power loading 8% to 100% power (normal)		18,000	15
4	Power unloading 100% to 8% power (normal)		18,000	
5	10% Step load increase (normal)		8,000	
6	10% Step load decrease (normal)		8,000	
7	Step load reduction (100% to 8% power) (upset)			
	Resulting from turbine trip	160		
	Resulting from electrical load rejection	150		
	Total		310	
8	Reactor trip (upset)			
	Resulting from complete loss of reactor coolant flow	40		
	Resulting from turbine trip w/o automatic control action	160		10
	Resulting from complete loss of main feedwater flow	88		
	Resulting from trips included in transient numbers 11, 15, 16, 17 & 21	112		
	Total		400	
9	Rapid depressurization (emergency)		40	
10	Change of flow (upset)			
	Resulting from loss of one or more reactor coolant pumps		20	
11	Rod withdrawal accident (upset)		40	
12	Hydrotests (test)		35	
13	Steady-state power variations (normal)		∞	
14	Control rod drop (upset)		40	
15	Loss of station power (upset)		40	

TABLE 4.1-1

OPERATING TRANSIENT CYCLES

10

Transient Number	Transient Description (ASME Category)	Design Cycles
16	Steam line failure (faulted)	1
17A	Loss of feedwater to one steam generator (upset)	20
17B	Stuck open turbine bypass valve (emergency)	10
18	Loss of feedwater heater (upset)	40
19	Feed and bleed operations (normal)	40,000
20	Miscellaneous (normal)	
	Resulting from makeup flow perturbations (Type A)	30,000
	Resulting from spray flow perturbations (Type B)	20,000
	Resulting from makeup flow perturbations (Type C)	$4x10^6$
21	Loss of coolant (faulted)	1
22	Test transients (test)	
	High pressure injection system	40
	Core flooding check valve	240
23	Steam generator filling, draining, flushing and cleaning (normal)	
	Steam generator secondary side filling	
	Condition 1	120
	Condition 2	120
	Steam generator primary side filling	
	Condition 1	120
	Condition 2	120
	Flushing	40
	Chemical cleaning	20
	Total	540
24	Hot functional testing (test)	1