UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

CALIFORNIA ENERGY COMMISSION'S PROPOSED FINDINGS OF FACT AND CONCLUSIONS OF LAW IN THE FORM OF AN INITIAL DECISION

CALIFORNIA ENERGY COMMISSION

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CALIFORNIA ENERGY COMMISSION'S PROPOSED FINDINGS
OF FACT AND CONCLUSIONS OF LAW

The California Energy Commission hereby submits proposed Findings of Fact and Conclusions of Law ("Findings") in the form of an initial decision for the consideration of the Board.[1]

FINDINGS OF FACT

I. Introduction

1. The Sacramento Municipal Utility District ("Licensee" or "SMUD") is the holder of Facility Operating License No. DPR-54, which authorizes operation of the Rancho Seco Nuclear Generating Station (the "facility" or "Rancho Seco"). Sacramento Municipal Utility District

1. The California Energy Commission's Findings are being filed August 4, 1980, pursuant to the Board's oral Order of July 28, 1980.

(Rancho Seco Nuclear Generating Station), Commission Order, Docket No. 50-312 (May 7, 1979), 44 Fed. Reg. 27779 (1979) (hereafter the "May 7 Order").The facility includes a Babcock and Wilcox ("B&W") designed pressurized water reactor ("PWR"), located at the Licensee's site in Sacramento County, California. Id.

2. This proceeding is directly related to the March 28, 1979 accident at the Three Mile Island, Unit 2 ("TMI") nuclear power plant, which employs a B&W PWR of the same basic design as Rancho Seco. That accident raised questions about the safety of all PWR's designed by B&W, including Rancho Seco. Id.; California Energy Commission ("CEC") Ex. 1, Admission Response 1; CEC Ex. 2, Admission Response 1; CEC Ex. 26.

3. As a result of the TMI accident and the safety concerns which arose from it, the Nuclear Regulatory Commission ("NRC") on May 7, 1979, ordered that Rancho Seco be shut down until certain immediate actions were accomplished. The NRC also ordered that various long-term actions also be accomplished, albeit not before the facility might be permitted to restart. May 7 Order. SMUD actually had shut Rancho Seco down on April 28, 1979 in order to accomplish the short-term actions. The May 7 Order had the effect of confirming that shut down and

requiring SMUD to accomplish the short and long-term actions. CEC Ex. 25; May 7 Order.

4. The broad purpose of this proceeding is to consider whether the actions specified in the May 7 Order were adequate or whether other actions should also have been required.[2] As specified hereafter, however,[3] we view our job more broadly: to determine the adequacy of the May 7 Order and the safety of continued operation of Rancho Seco in the context of the full range of TMI-related developments which have taken place subsequent to the May 7 Order.

5. In addition to SMUD, there are two other active participants in this proceeding: the NRC Staff and the California Energy Commission, which is participating as a representative of an interested state pursuant to 10 C.F.R. 2.715(c). These parties participated actively by presenting testimony and documentary evidence and cross-examining at the hearings held February 26-28, March 3-6, April 8-11 and 14-17, and May 6-10 and 12-14, 1980.

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2. In the May 7 Order, the NRC stated that affected persons could request a hearing to test the adequacy of the May 7 Order. After petition requests were received, the NRC on June 21, 1979, specified in greater detail that the hearing would test the adequacy of the May 7 Order. <u>See</u> <u>Sacramento Municipal Utility District</u> (Rancho Seco Nuclear Generating Station), CLI-79-7, 9 N.R.C. 680 (1979), <u>motion to stay denied, Friends of the Earth, Inc. v. United States</u>, 600 F.2d 758 (9th Cir. 1979), <u>pet. review pending</u>.

3. <u>See</u> Section III.

6. Friends of the Earth, the Environmental Council of Sacramento and the Original SMUD Ratepayers Association (collectively "FOE") and Mr. Gary Hursh and Mr. Richard D. Castro were admitted as parties under 10 C.F.R. 2.714. Prehearing Conference Order, Aug. 3, 1979. Prior to the commencement of the hearing in this matter, however, FOE and Messrs. Hursh and Castro withdrew from the proceeding.[4] In the exercise of our discretion, we decided that certain contentions raised by these parties addressed serious safety questions, and accordingly, should be decided. We have rephrased many of these issues and designated them as Board Questions. Order Subsequent to the Prehearing Conference of February 6, 1980, dated February 14, 1980.[5]

7. We shall begin this Decision with a background discussion of the events giving rise to this proceeding. An understanding of this background is essential to consideration of the serious safety questions which have been raised. Thereafter, we shall address certain

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4. Messrs. Hursh and Castro withdrew at the second prehearing conference held February 6, 1980. FOE announced its withdrawal in a limited appearance statement. Tr. 170-71.

5. These issues are designated as Board Question H-C ___ and Board Question FOE ___.

procedural and substantive issues relating to the scope of this proceeding and participants' respective burdens and then shall make findings on the various issues presented.

II. Background: The TMI Accident and The May 7 Order

8. On March 28, 1979, TMI experienced a feedwater transient, together with subsequent equipment failures and operator errors, that resulted in fuel failure which exceeded design basis expectations. E.g., NRC Ex. 4 at 3-1. If Rancho Seco had experienced the same equipment failures and operator errors as did TMI, it would have experienced substantially the same accident.[6] Tr. 3032 (Rodriguez).

9. In the weeks immediately following the TMI accident, the NRC Staff intensively evaluated the causes and consequences of the TMI accident. As a result of this review, all holders of operating licenses for B&W reactors were required to take certain actions designed to preclude an accident similar to TMI from occurring at another B&W reactor. NRC Ex. 4, Appendix A.

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6. Rancho Seco likely would not have experienced the same off-site radioactive releases as occurred at TMI because its containment would have isolated earlier. See Section V.L., infra.

10. Notwithstanding the immediate actions required by the NRC in early April 1979, the NRC Staff, in late April 1979, concluded that all operating B&W plants, including Rancho Seco, should be shut down. The Staff's bases for this decision are contained in a document entitled "Status Report on Feedwater Transients in B&W Plants" ("NRR Status Report"), dated April 25, 1979 and prepared by NRC's Office of Nuclear Reactor Regulation. This report, admitted into evidence as CEC Exhibit 26, states:

> We conclude that we do not now have reasonable assurance that these B&W plants can continue to operate without undue risk. We believe that these plants should be shutdown now, and that the following information is necessary before restart can be permitted.
>
> In the short-term, we must take all reasonable steps to reduce the likelihood of occurrence of transients at B&W plants and to improve standing instructions, training and emergency procedures available to plant operators. This can be accomplished by:
>
> a. Reviewing and upgrading, as appropriate, auxiliary feedwater reliability and performance (timeliness);
>
> b. Reviewing results of FMEA analysis of ICS and taking actions, as to reduce its likelihood of initiating or exacerbating transients;
>
> c. Hard wiring anticipatory scram based on FW transients;
>
> d. Reviewing detailed analyses of plant response to transients to effects of HPI injection, and return to natural circulation cooling; and

e. Reviewing new and augmented standing instructions and emergency procedures for plant operators developed as a result of a-d above, and training plant operators and the new and augmented instructions and procedures including the stationing of a full-time dedicated operator to take appropriate prompt manual actions. CEC Ex. 26 at 1-7 (emphasis supplied).

The NRR Status Report then goes on to state:

> In the long-term, we must either reduce the sensitivity of the response of B&W plants to transients by design changes, or substantially upgrade the instrumentation and controls available to the plant operator and substantially upgrade plant operator education training and experience. Id. at 1-8.

11. At approximately the same time as the NRC Staff reached its shutdown decision, it conveyed this determination to SMUD and to other B&W licensees. The Staff presented no proposed criteria for allowing the B&W facilities to resume operation. Tr. 3253 (Rodriguez); NRC Ex. 4 at 3-3.

12. The day after learning of the NRC's shutdown decision, SMUD management received a telephone call from Harold Denton, Director of the NRC's Office of Nuclear Reactor Regulation. Mr. Denton informed SMUD that Duke Power Company, also the licensee of a B&W facility, had volunteered to shut down its facility and had proposed specific criteria for restart. Mr. Denton stated that the

Staff would accept Duke Power's restart criteria. Tr. 3254 (Rodriguez).

13. In late April, 1979, SMUD was extremely concerned that the NRC might shut down Rancho Seco without providing explicit criteria for restart. Tr. 2029 (Dieterich); Tr. 3253, 3260 (Rodriguez). To avoid this situation, SMUD decided to accept Mr. Denton's proposal and on April 27, 1979, SMUD authored a letter similar to that of Duke Power, volunteering to shut down Rancho Seco and setting forth specific restart criteria based upon the Duke Power letter, as modified to apply to Rancho Seco. CEC Ex. 25; Tr. 3254 (Rodriguez). In its April 27, 1979 letter, SMUD proposed to shut down Rancho Seco and complete the following actions prior to restart:

> a. Upgrade the timeliness and reliability of delivery from the Auxiliary Feedwater System by carrying out actions as identified in Enclosure 1 of [CEC Exhibit 25].
>
> b. To develop and implement operating procedures for initiating and controlling auxiliary feedwater independent of Integrated Control System control.
>
> c. Implement a hard-wired control-grade reactor trip that would be actuated on loss of main feedwater and/or turbine trip.
>
> d. Complete analyses for potential small breaks and develop and implement operating instructions to define operator action.

e. Provide for one Senior Licensed Operator assigned to the control room who has had Three Mile Island Unit No. 2 (TMI-2) training on the B&W simulator. CEC Ex. 25.

In addition, SMUD also proposed to undertake additional long-term modifications, albeit not prior to restart. These were:

a. The licensee will provide to the NRC Staff a proposed schedule for implementation of identified design modifications which specifically relate to items 1 through 9 of Enclosure 1 to the licensee's letter of April 27, 1979, and would significantly improve safety.

b. The licensee will submit a failure mode and effects analysis of the Integrated Control System to the NRC Staff as soon as practicable. The licensee stated that this analysis is now underway with high priority by B&W.

c. The reactor trip following loss of main feedwater and/or trip of the turbine will be upgraded so that the components are safety grade. The licensee will submit this design to the NRC staff for review.

d. The licensee will continue operator training and have a minimum of two licensed operators per shift with TMI-2 simulator training at B&W by June 1, 1979. Thereafter, at least one licensed operator with TMI-2 simulator training at B&W will be assigned to the control room. All training of licensed personnel will be completed by June 28, 1979. Id.

14. On May 7, 1979, the NRC issued its confirmatory order requiring SMUD to shut down Rancho Seco and to

complete the short-term items identified in SMUD's April 27, 1979, letter before returning the reactor to power operation. May 7 Order at 3, 6; NRC Ex. 4 at 1-2 and 3-2; Tr. 3696-97 (Capra).

15. On June 27, 1979, the NRC Staff issued a report (the "Staff Evaluation"), concluding that SMUD had satisfactorily completed the short-term items set forth in the NRC's May 7 Order and that it should be permitted to resume normal operation of the facility.[7] Shortly thereafter, Rancho Seco resumed power operation.

16. In addition to the requirements of the May 7 Order, the NRC has required SMUD to undertake other equipment, procedure, and personnel changes related to the TMI accident. These changes are primarily contained in Inspection and Enforcement (I&E) bulletins (Nos. 79-05A, 79-05B, and 79-05C), and two reports of the "TMI-2 Lessons Learned Task Force" (NUREG-0578 and NUREG-0585). NRC Ex. 4 at 3-1 through 3-8. A comprehensive list of these changes is included in the record. NRC Ex. 4, Appendix A.

III. Scope of the Proceeding and Allocation of Burdens

17. This proceeding has taken place in unusual circumstances, given the TMI accident and the unprecedented

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7. The Staff Evaluation is inserted in the transcript subsequent to page 362.

attention devoted to nuclear safety in the aftermath of that accident. Indeed, the Commission's May 7 Order was extraordinary in its confirmation of requirements proposed by the Licensee in response to a Staff conclusion that the Rancho Seco facility should be shut down. Consequently, the Commission's June 21 Order empanelling this Board was also unique in its authorization to review the adequacy of the already effective May 7 Order. As a result, the Board has several times been called upon to consider the scope of this hearing and the allocation of burdens among the parties.

18. The scope of this hearing was settled early in Board rulings based on the Commission's June 21, 1979 Order. That Order outlined the broad issues to be examined:

> 1. Whether the actions required by subparagraphs (a) through (e) of Section IV of the [May 7] Order are necessary and sufficient to provide reasonable assurance that the facility will respond safely to feedwater transients, pending completion of the long-term modifications set forth in Section II. A contention challenging the correctness of the NRC staff's conclusion that the actions described in subparagraphs (a) through (e) have been completed satisfactorily will be considered to be within the scope of the hearing. However, the filing of such a contention shall not of itself stay operation of the plant.
>
> 2. Whether the licensee should be required to accomplish, as promptly as practicable, the

long-term modifications set forth in Section II of the [May 7] Order.

3. Whether these long-term modifications are sufficient to provide continued reasonable assurance that the facility will respond safely to feedwater transients.

In addition, in a public meeting on July 11, 1979, to consider whether or not to amend its Order of June 21, 1979, the NRC determined that the Board was not precluded from inquiring into Licensee management competence and control, and voted to forward the transcript of that meeting to the Board.

19. After hearing argument on the scope of the hearing, the Board ruled that we would examine "all matters and issues which hinge upon the response to feedwater transients." Order Ruling on Scope and Contentions, October 5, 1979, at 3. In that context, we are, of course, compelled to examine the sufficiency of the actions confirmed by the Commission in its May 7 Order, as well as Licensee's implementation of the actions it proposed. As set forth in this Initial Decision, we have concluded that the actions proposed by SMUD were not sufficient to ensure that the facility would safely respond to feedwater transients. <u>See</u> Section V.O. However, we have not halted our inquiry at this conclusion. We have also considered the more current issue whether the additional measures

implemented at Rancho Seco since it resumed operation have provided the necessary reasonable assurance. It would not be fair to the Licensee nor useful to the Commission or the public for this Board to issue a ruling that ignores the events since Rancho Seco resumed operation. The Board has therefore viewed the scope of this hearing to be whether the measures implemented at Rancho Seco since the TMI accident, as well as those measures which are reasonably certain to be implemented in the near future, reasonably assure that the facility will safely respond to future feedwater transients.

20. The unusual context of this hearing also raised questions about the appropriate allocation of burdens among the parties. The Commission's Rules of Practice, 10 C.F.R. §2.732, provide that "unless otherwise ordered by the presiding officer, the applicant or the proponent of an order has the burden of proof." Here, the Licensee is the originator of the action confirmed by the May 7 Order. Accordingly, the Board held in our Prehearing Conference Order of August 3, 1979, that the burden of proof on all contentions would be placed on the Licensee. The Board also ruled that the burden of going forward on contentions would be placed on the party making the contention.

21. The California Energy Commission sought clarification of the Board's ruling on the burden of going forward, pointing out that as an interested state its issues were more akin to Board Questions than contentions. Thus, the Energy Commission argued that the burden of going forward on its issues should be shared by all parties. In our October 3, 1979 Order Ruling on Scope and Contentions, the Board adopted some of the Energy Commission's issues as Board Questions and left others as Energy Commission issues. On October 24, 1979, the Energy Commission restated its request for clarification of the burden of going forward on its issues. On December 17, 1979, the Board responded, stating that we viewed the Energy Commission's issues to be like contentions with regard to the burden of going forward. Thus, the Board held that the Energy Commission should carry that burden on its issues, with the exception of those adopted as Board Questions. The Energy Commission thereafter presented affirmative evidence on each of its issues in satisfaction of this burden.

22. Unlike the Energy Commission, the Licensee at no time sought clarification of its assigned burden of proof. Thus, the hearing went forward with that burden allocated to the Licensee as stated in the Board's Prehearing Conference Order of August 3, 1979. Following the

hearings, Licensee for the first time suggested that its assigned burden of proof was "altered" by the withdrawal of FOE and Messrs. Hursh and Castro and the Energy Commission's status as a representative of an interested state. Licensee's Proposed Findings of Fact and Conclusions of Law in the form of an Initial Decision (hereafter "Licensee's Findings") at 18, para. 26. The Licensee did not suggest in what manner or for what reason its assigned burden had been altered, although its subsequent findings suggested that it viewed the Energy Commision as having the burden of proof on its issues. E.g., Licensee's Findings at 175, para. 235 and 184, para. 247.

23. The Board views Licensee's suggested reallocation of the burden of proof as untimely and unwarranted. This burden was appropriately given to Licensee at the first prehearing conference, and no party thereafter asked the Board to reconsider or clarify that ruling. If the Licensee believed that events altered the bases of our ruling, it should have raised the issue prior to the hearing.

24. However, while we view the burden issue as settled, we shall briefly express our view regarding the merits of Licensee's assertion. Licensee is the logical

proponent of the May 7 Order, as it proposed the actions confirmed therein and has asserted that they sufficient to provide the necessary assurances that Rancho Seco can be safely operated. CEC Ex. 25. As the proponent, when other participants offer evidence which demonstrates that additional actions would enhance Rancho Seco's safety, it is appropriate for Licensee to have the burden to prove otherwise. We view this as particularly sensible in this case in view of the undesirable sensitivities of the B&W system. See Section V.A. Accordingly, the ultimate burden of proof rests on the Licensee on all issues in this proceeding.

IV. Summary of Contested Issues and Board Findings

25. While this Board has before it a great number of contested issues, they may conveniently be considered in several broad categories. First, there are questions relating to the design of the B&W nuclear steam supply system ("NSSS"), particularly the sensitivity of that system to upsets caused in the secondary system through use of a once-through steam generator ("OTSG").

26. With respect to the first category of issues, we find that there are certain design and operational sensitivities inherent in the B&W system which could

contribute to severe accidents at B&W facilities. In particular, the OTSG employed in B&W plants is extremely sensitive to feedwater perturbations in the secondary system, which events can cause rapid pressure and temperature changes in the primary system. This results in significantly less time for operator and equipment responses, a situation which can lead to greater safety system challenges and greater possibility of operator errors due to the reduced time in which to take action before safety systems are challenged.

27. The design of the B&W NSSS has not been altered to eliminate the system's inherent responsiveness. Accordingly, we find that Rancho Seco must have instrumentation, equipment and personnel adequate to respond to the events which may result from the design sensitivities. These questions relating to instrumentation, equipment and personnel comprise the second broad category of issues. In this regard, it bears repeating the NRC Staff's statement made soon after the TMI accident.

> In the long term, we must either reduce the sensitivity of the response of B&W plants to transients by design changes, or <u>substantially</u> upgrade the instrumentation and controls available to the plant operator and <u>substantially</u> upgrade plant operator education training and experience. CEC Ex. 26 at 1-8 (emphasis supplied).

Thus, in view of our findings that design changes to reduce sensitivities have not been made, the evidence must demonstrate <u>substantial</u> improvements in instrumentation, controls and personnel before we can find that no further modifications are necessary. We cannot make such a finding.

28. SMUD has made significant efforts since TMI to upgrade its capability to respond to feedwater transients. However, in certain respects, greater efforts can and should be made. In particular, additional work needs to be done on the integrated control system and the auxiliary feedwater system to ensure their reliability and substantial improvements need to be made in operator and management training.

29. The deficiencies which we identify in this Decision are not trivial. Rather, they relate directly to the long-term safe operation of the Rancho Seco facility. However, it is not our view that the deficiencies require an immediate shutdown of the facility. Rather, we believe it is consistent with the public health and safety to continue to operate Rancho Seco, provided prompt and responsible action is taken to comply with this Decision.

30. This Board is compelled to make one further general observation. While no participant specifically challenged the adequacy of the short- and long-term measures proposed by the Licensee and confirmed by the

May 7 Order, an obvious purpose of this proceeding is precisely to test whether those measures were, in fact, adequate. We find, without hesitation, that these measures were not adequate. The short- and long-term requirements set forth in the May 7 Order were not decided upon after careful analysis of necessary steps to improve Rancho Seco's safety. Rather, they were devised virtually overnight with a premium on actions that could be completed rapidly and thus ensure prompt restart of the facility. Steps which might take more than a few weeks to implement were not included in the short-term items even though at least one, the failure mode and effects analysis of the integrated control system, had already been identified by the NRC Staff as prerequisite to continued operation. It therefore is evident that the public interest was poorly served by the Licensee's proposal and the NRC Staff's support of the May 7 Order -- the public health and safety took second place to expeditious restart. Fortunately, there have been numerous efforts since May 7, 1979 to upgrade Rancho Seco safety, steps which have served somewhat to compensate for the initial inadequacy of that Order.

V. Findings of Fact on Contested Issues

A. The Sensitivity of B&W PWR's to Feedwater Transients

Additional Board Question No. 3:

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

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

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

31. The concerns expressed in Additional Board Question No. 3 mirror those stated by the NRC in its May 7, 1980 Order; namely, that an unusual B&W design characteristic, i.e., the OTSG results in a NSSS which is extremely sensitive to secondary side feedwater perturbations. See May 7 Order. We consider this matter to be one of fundamental importance in resolution of most of the safety issues raised concerning the B&W system since

it is primarily the alleged design sensitivities which purportedly create the unsafe situation. Id. This issue requires us to address several related matters: What are the design sensitivities; have the sensitivities been eliminated by post-TMI actions, including those required by the May 7 Order; and what safety implications arise if those sensitivities have not been eliminated? These questions will only be partially answered in this section of this Initial Decision because the entire Decision is basically devoted to the question of whether these sensitivities have, in fact, been sufficiently controlled to permit continued operation of the facility.

32. Unlike other PWR's, B&W facilities use a OTSG. Webb Testimony at 5-6, following Tr. 1801; Karrasch and Jones Testimony at 16-25, following Tr. 535; Rubin and Novak Testimony on Sensitivity of the Once-Through Steam Generator Design at 3, following Tr. 1163 ("Rubin and Novak OTSG Testimony"). The design of the OTSG makes B&W facilities unusally sensitive to the effects of feedwater transients. NRC Ex. 4 at 2-2. This sensitivity takes two related forms. First, because the OTSG has a smaller secondary side volume than other designs,[8] changes in

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8. The secondary coolant volume in a B&W OTSG is about one-third of the volume of the secondary side of a U-tube steam generator. Tr. 518 (Lewis).

feedwater flow cause relatively rapid changes in OTSG feedwater level. This also means that the OTSG boils dry more quickly than other designs in a loss of feedwater transient. Indeed, even with an anticipatory reactor trip (discussed <u>infra</u>), a B&W OTSG will boil dry in approximately four minutes compared to a boil dry time of 15 - 20 minutes for a Combustion Engineering PWR and 20 - 30 minutes for a Westinghouse PWR. Tr. 589 (Karrasch); 1608 (Matthews); CEC Ex. 26 at 1-1. Second, the feedwater level in the OTSG determines the amount of heat transfer between the primary and secondary systems. Thus, the OTSG closely couples the primary and secondary systems such that feedwater transients result in rapid changes in primary system pressure and temperature. May 7 Order at 1-2; Lewis Testimony at 12, following Tr. 477; Webb Testimony at 5-8; Rubin and Novak OTSG Testimony at 3-5; Tr. 1075 (Karrasch and Jones); CEC Ex. 5; NRC Ex. 4 at 5-15 through 5-19; NRC Ex. 2 at 4-11.

33. This close coupling quickly translates a secondary system malfunction into a gross disturbance of the primary system. One Rancho Seco operator expressed this effect as follows: "[T]he biggest response [to a feedwater transient] comes from the primary coolant side of the plant." CEC Ex. 37 at 14. Another operator put it more vividly:

> Feedwater does, you know, it is a big deal. But you can drop pressure like crazy by just adding a little bit of cold water. CEC Ex. 38 at 15.

34. The design of the OTSG has certain operational advantages, particularly in generating superheated steam and permitting rapid adjustment to load changes. Karrasch and Jones Testimony at 16; NRC Ex. 4 at 5-1, 5-18. However, in transient conditions the OTSG sensitivities have distinct disadvantages, particularly that secondary side disturbances are rapidly reflected in the primary system. Thus, for example, if a B&W OTSG boils dry due to a loss of feedwater transient, there will be a rapid heat up in the primary system -- indeed, far more rapidly than in other PWR's. CEC Ex. 26 at 1-2, 2-3 and 2-4.

35. Licensee has suggested that these sensitivities of the B&W system do not constitute safety concerns because they have been taken into consideration in the licensing of the plant. Karrasch and Jones Testimony at 16; Tr. 2010-11, 2088-89 (Dieterich). However, we find that these sensitivities, particularly the rapid boil dry time of the OTSG, do represent significant safety concerns chiefly because they require more rapid and precise operator and

equipment responses to feedwater disturbances. NRC Ex. 4 at 2-2 and 2-3. As a matter of logic, such requirements for more rapid response increase the likelihood of operator errors. Further, these sensitivities may "result in unnecessary challenges to pressure relief devices or the engineered safety features." Id.

36. The sensitivity of the B&W NSSS to feedwater transients was highlighted as a design deficiency in the NRR Status Report.

> We identify some design and analysis deficiencies of this class of plant and note some possible remedial measures.
>
> There are several design differences that distinguish a B&W plant in its response to feedwater transients:
>
> a. The mass of liquid in the secondary side of the steam generator is less than that for other PWRs. More importantly, the B&W design operates as a superheat boiler. Thus, the steam generator tubes are uncovered for a major portion of their length in steady operation. In this mode, changes in feed flow are quickly manifested as changes in heat transfer from the primary system. In this manner, absent prompt and remedial action by the control system (and in some cases a safety system), the steam generator will dry out. Ex. 26 at 1-1.

37. The evidence presented in this proceeding revealed that there have been no design changes implemented at Rancho Seco that reduce this sensitivity with regard to the coupling of the primary and secondary systems. Webb

Testimony at 12-13; Rubin and Novak OTSG Testimony at 8; NRC Ex. 4 at 2-2. However, the Staff's B&W Transient Response Task Force has recommended that licensees investigate design changes to reduce the sensitivity. NRC Ex. 4 at 5-19. NRC witness Capra described possible avenues that might be explored:

> I think, for instance, to have the facility operate with less superheat, operate at a different level, or a level control in the once-through steam generator which would be a high level.
>
> It is not operating at a specific level now, but based on steam pressure and the amount of superheat, one passive method that was discussed that we are not sure of the feasibility is possibly providing a surge tank effect, or a surge tank on the feedwater lines themselves, such that if you had a loss of feedwater, you would have a surge volume similar to a core flood tank which would provide passively feedwater for a certain period of time which would give you a longer time to get on the auxiliary feedwater system to prevent the steam generator from drying out.
>
> It is possible to change set points on the secondary side, either on the turbine bypass valves -- maybe I said steam generator bypass, turbine bypass valves, or steam generator safety valves.
>
> There are a lot of possibilities. Until sensitivity studies are done to see if they are feasible and what net effects they would have, it is not possible to be definitive on what the best way to go would be. Tr. 3732-33 (Capra).

Mr. Capra felt that such studies could be completed in two years. Id.

38. Given the fact that B&W design sensitivities continue to be present, it is appropriate to turn attention to the important subsidiary question raised in Additional Board Question 3 and the NRR Status Report: Whether instrumentation, controls, equipment and operator training at Rancho Seco have been substantially upgraded subsequent to TMI in a manner which satisfactorily compensates for the continued design sensitivities inherent in the B&W system. These subsidiary questions represent the heart of the remainder of this Initial Decision. However, at the outset, certain findings can be made.

39. In response to the TMI accident, certain actions have been taken to lessen the quick response of B&W systems. The primary action was the addition of an anticipatory reactor trip, as required by the May 7 Order. This device trips the reactor more quickly on a loss of main feedwater or turbine trip, extending the OTSG boil dry time from approximately 1-2 minutes to approximately four minutes. Tr. 589 (Karrasch); CEC Ex. 26 at 2-3. However, even with the anticipatory trip, B&W plants have a much shorter boil dry time than other PWR's. Finding 32. Thus, while this measure provides some additional time for

operator response to a feedwater transient, it does not substantially reduce the close coupling of the primary and secondary systems induced by the OTSG. Indeed, witnesses tended to downplay the importance of the anticipatory trip, stating that there wouuld be minor safety implications if it should fail. Thatcher Anticipatory Reactor Trip Testimony at 9; Dieterich Testimony at 16; Karrasch and Jones Testimony at 27; Tr. 2127-28 (Dieterich).

40. This Board also has raised questions regarding the reliability of the anticipatory trip, asking basically whether it is reasonable to take credit for the anticipatory trip before it is made safety grade.[9] However, no witness seriously questioned the reliability of the trip and, accordingly, we find that the control grade

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9. Board Question H-C 9:

Has the reliability of the recently installed control grade reactor trip from loss of feedwater/turbine trip been adequately demonstrated?

Additional Board Question 1:

At a meeting with owners of B&W reactors held on August 23 it was noted that, in the interim then elapsed since the TMI-2 accident, control-grade hard-wired anticipatory reactor trips (ART) have been called to respond four times and failed once:

a. Is it typical of performance by control grade trips?

b. What are the safety implications for operation of Rancho Seco before such trips are upgraded?

trip is satisfactory. Staff Evaluation 14-16; Dieterich Testimony at 15; Tr. 1126-27 (Karrasch and Jones); Tr. 1711-12 (Thatcher); Tr. 2128-29, 2332-33 (Dieterich). We note, however, that the anticipatory reactor trip will shortly be redesigned to safety grade which should ensure its reliability. Dieterich Testimony at 15.

41. In addition, there have been changes made to the high reactor coolant pressure trip setpoint (from 2355 to 2300 psig) and to the setpoint for the pressurizer power operated relief valve ("PORV") (from 2255 to 2450 psig). I&E Bull. 79-05B; NRC Ex. 4 at A-4. These actions, however, were designed primarily to reduce challenges to the PORV [Thatcher Anticipatory Reactor Trip Testimony at 3] and they do not eliminate the system's design sensitivities. Indeed, as discussed later [Section V.E.] these changes, while limiting PORV challenges, may increase challenges to safety valves. Finally, SMUD also has acted to upgrade the auxiliary feedwater system and the integrated control system, and new emphasis has been placed on operator training. These actions are discussed in detail in subsequent sections of this Decision.

42. The actions relating to B&W sensitivities point up the significant fact about B&W plants: they require a highly interactive and responsive control system. In

addition, operators at Rancho Seco may be required to take more rapid action and have a better understanding of instrument response than operators on plants having other designs. Thus, B&W reactors like Rancho Seco must have a substantially more responsive control system and substantially better trained operators than other PWR designs. NRC Ex. 4 at 2-3; Lewis Testimony at 12, following Tr. 477; Minor and Bridenbaugh Testimony at 13, following Tr. 3496; Rubin and Novak OTSG Testimony at 5; CEC Ex. 26 at 1-8.

43. In conclusion, with respect to Additional Board Question No. 3, we find as follows:

a. The OTSG so closely couples primary system behavior to secondary system disturbances that disturbances of the primary system are inevitable for feedwater transients.

b. Subsequent to the TMI accident, there have been no basic design changes which eliminate this close coupling. However, there are various possible design changes that might be investigated to reduce sensitivities.

c. The close coupling represents a serious safety concern, primarily because it reduces operator response time and increases potential challenges to other systems.

d. The addition of an anticipatory reactor trip has reduced somewhat the effects of OTSG sensitivity but even with this trip, the B&W NSSS may be subject to more severe primary system disturbances than other PWR's and have a more rapid boil dry time.

e. It is clear to us that the close coupling, at a minimum, requires that Rancho Seco have substantially better control systems, instrumentation and operators in order to ameliorate the effects of B&W design sensitivities.

B. Integrated Control System

Board Question H-C 16

Is the failure mode and effects analysis for the Rancho Seco integrated control system complete and adequate?

44. The close coupling of the primary and secondary systems induced by the OTSG requires B&W facilities to use a different control system than other nuclear power plants. This control system is called the integrated control system or "ICS". The ICS is the principle control system for all the important parameters of the plant's operation, including reactor power, primary system temperature and pressure, feedwater flow and level, steam production and flow, and, ultimately, power production. Karrasch and Jones testimony at 7-12. In some B&W plants, including TMI and Rancho Seco, the ICS also controls auxiliary feedwater flow during a loss of main feedwater or loss of all reactor coolant pumps. Thatcher ICS Testimony at 3-4, following Tr. 1163. The ICS is designed to control plant parameters and to compensate for the sensitivity of B&W plants by responding automatically to changes in these important plant parameters. Id. at 2-3. Thus, our consideration of the ability of the Rancho Seco system to control the sensitivity of the OTSG begins with an

examination of the reliability of the ICS. See generally NRC Ex. 4 at 5-49 and 5-50.

45. The reliability of the ICS has been questioned because it is more complex than the control systems used at other PWR's and because it is not designed to meet the single failure criterion of IEEE standard 279. CEC Ex. 26 at 1-1 and 1-2; NRC Ex. 4 at 5-53; 10 C.F.R. 50.55a(h). Because of the complexity of the ICS, it would be extremely difficult to design it as a safety system. NRC Ex. 4 at 5-53 and 5-56.

46. The NRC Staff's concerns regarding the ICS were described in April, 1979, in the NRR Status Report, which set forth the following comments in response to the rhetorical question: "Does the ICS perform satisfactorily?"

> a. B&W has stated and we [NRC Staff] agree, that "we are not satisfied with the reliability of the integrated control system".
>
> b. The failure modes and effects have not been systematically analyzed. . . .
>
> c. The ICS may initiate a feedwater transient (on the order of 10-15% of all events in the past).
>
> d. The ICS controls AFW in some plants. . .and could contribute to loss of AFW.
>
> e. Even when the ICS works well there may be, in response to a feedwater transient, wide swings in reactor pressure, pressurizer level, and average reactor coolant temperature. CEC Ex. 26 at 1-5.

47. An additional concern expressed in the NRR Status Report with respect to the ICS was the combination of (c) and (d) in the previous Finding: that an ICS failure might cause a loss of both main and auxiliary feedwater. The Report stated:

> B&W was unable to state whether failures in the Integrated Control System could initiate a LOFW event and also inhibit AFW via the flow control valves. We have asked B&W to analyze this question promptly. If this common-mode failure can occur, and we see no reason why it is impossible, then the combined frequency AB (see Section 2.3.1) could be high because, for these events B=1. CEC Ex. 26 at 2-9.

The last sentence is particularly significant because, as the Report describes at Section 2.3.1:

> For a LOFW event, either AFW or HPI must function to protect the core. (There are some alternatives, such as restoring main feedwater flow, but they do not significantly change the picture.) The rate of accidents (fuel damage) would therefore be:
>
> A(BC)
> where A = challenge rate
> B = failure probability of AFW
> C = failure probability of HPI
>
> Hence, "failure" means insufficient functioning to cool the core, and involves consideration of performance, timing, and reliability. Given A = 2 per reactor year, the product BC must be adequately low; numerical guidance is not currently available. CEC Ex. 26 at 2-6 and 2-7.

Thus, if the ICS can both cause a feedwater transient and inhibit AFW flow, the probability of core damage may be high, depending upon the failure probability of HPI.

48. The NRC Staff described another concern with the ICS in a review of the design sensitivity of B&W facilities entitled "Primary System Perturbations Induced by the Once Through Steam Generator". CEC Ex. 5. This report was prepared after completion of the B&W reliability analysis of the ICS (discussed below). The report stated:

> The ICS appears to play a significant role in the plant's feedwater response. The staff is currently reviewing an FMEA study on the ICS. However, a review of operating experience suggests that the ICS often is a contributor to feedwater transients. In some cases the ICS appeared inadequate to provide sufficient plant control and stability. Some of the utility descriptions of feedwater transients (as summarized in the minutes of a meeting on August 23, 1979) emphasized the role of the operator in operating the MFW system. . . . CEC Ex. 5, Section IV.

This report also identified fluctuation in the main feedwater system (MFW) as a contributor to feedwater transients at B&W facilities. Id. Section II. Since MFW flow is controlled by the ICS, these fluctuations also raise concerns regarding its performance.

49. The concerns regarding the ICS led the NRC Staff to recommend in the April 25, 1979, NRR Status Report that a failure modes and effects analysis ("FMEA") be performed on the ICS and its results reviewed as a short-term item to be completed prior to further operation of the facility. CEC Ex. 26 at 1-7. An FMEA is a systematic procedure for identifying the modes of failure of a system and for evaluating their consequences. It is considered the first general step of a reliability analysis which can potentially provide some early useful information and a basis for later studies. Thatcher ICS Testimony at 6.

50. Although the NRR Status Report identified the FMEA as a measure that should be completed before restart of the facility, this was not proposed by SMUD as a restart requirement and was not made a short-term requirement of the NRC's May 7 Order. CEC Ex. 25; May 7 Order. SMUD resisted making the FMEA a restart requirement, at least in part because it could not be completed as quickly as the other short-term items. Tr. 2035 (Dieterich). At the time the restart criteria were being negotiated, however, B&W estimated the FMEA would be completed by later June, 1979. Tr. 1381 (Thatcher); Tr. 2036-37 (Dieterich).

51. The completion of an FMEA was made a long-term requirement of the May 7 Order. In response, B&W prepared an FMEA as part of a document entitled "Integrated Control System Reliability Analysis." In addition to the FMEA, this report included a review of the operating history of the ICS. The report was completed in August 1979. CEC Ex. 3.

52. The B&W FMEA concluded that "the reactor core remains protected throughout any of the ICS failures studied." CEC Ex. 3 at 2-1. The specific conclusions drawn from the FMEA were that:

1. The FMEA indicates that an inadvertently opened or stuck open turbine bypass valve could result in overcooling. (The plant data do not show a significant frequency of turbine bypass malfunctions, however.)

2. The FMEA also indicates that an inadvertently opened or stuck open main feedwater startup valve could result in steam generator overfill and overcooling.

3. The FMEA identifies feedwater pump speed control failure to both feedwater pumps as the only postulated failure that could adversely affect feedwater control to both steam generators after a reactor trip.

B&W's study of operating data from its plants concluded that "ICS hardware performance has not led to a significant number cf reactor trips (6 out of 310)". Id. The specific conclusions from this portion of the report were that:

1. The NNI/ICS power sources (external to ICS cabinets) have been vulnerable to single failures and human errors that have led to reactor trips and plant overcooling.

2. Failures of RC flow signals to the ICS have led to spurious reactor trips.

3. The ICS has shown a tendency to cause or to participate in feedwater oscillations, which have led to high RC pressure trips, low RC pressure trips, actuation of ESFAS, and loss of main feedwater. (Refer to Table 5-2, section 5, Operating Experience). Nonetheless, the ICS has prevented more reactor trips than it has caused and thus its net effect has been a reduction in the number of challenges to the reactor protection system.

4. When driven at a minimum speed, the main feed pump turbine has experienced a loss of oil pressure, causing loss of feedwater. The minimum speed stop and back up oil pressure should be examined so that unnecessary loss of or indication of loss of main feedwater is minimized. Id. at 2-2.

53. The Oak Ridge National Laboratory ("ORNL") reviewed B&W's ICS FMEA at the request of the NRC Staff. Board Ex. 1 at 2. ORNL found that:

> The B&W analysis . . . deals only narrowly with the ICS itself and not at all with the plant systems with which it interacts. With note of the concerns expressed and the guidance given in the NRC orders, the B&W analysis is more notable for what it does not include than for what it does include. . . .

* * * *

> In summary, the report deals only with a very limited scope of failures, essentially within the ICS cabinets; the only significant measure of response is whether a reactor trip would occur. Because of this limited scope, the results are necessarily of limited value. Board Ex. 1 at 3 and 4.

54. The ORNL review of the FMEA identified several specific inadequacies, the foremost of which was the choice of the systems which were analyzed. Rather than considering the ICS as including sensing, signal conditioning, actuating equipment, and power supplies, B&W limited its review to only the control system cabinets. Oak Ridge observed that a control system, especially one claimed to be "integrated" with other plant systems, cannot be meaningfully evaluated without consideration of the interaction between the cabinets and these other systems. Board Ex. 1 at 6.[10] Similarly, in NUREG-0667, the B&W Transient Response Task Force noted that the FMEA "did not address the very significant control board information problem encountered at Oconee 3 and Crystal River 3." NRC Ex. 4 at p. 5-59.

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10. The ORNL review states: "A control system, particularly one claimed as 'integrated,' should include sensing, signal conditioning, and actuating equipment and perhaps power supplies -- if not primary power sources. The system being controlled includes a number of process loops that are highly interactive and which must often operate within rather narrow individual constraints. The B&W analysis does not address these interactions. Id.

55. ORNL also criticized the FMEA because it examined failures of functional blocks rather than specific equipment. Board Ex. 1 at 6, 10. As the Licensee's witnesses admitted, an equipment block analysis is more detailed than the functional block analysis performed by B&W. Tr. 647 (Karrasch).[11] ORNL pointed out that a functional block analysis may miss undisclosed couplings or interactions between blocks such as power supplies or fuses, and therefore can be misleading. Board Ex. 1 at 6 and 10.

56. ORNL also found the FMEA deficient in that it seldom considered the effects of failures beyond reactor trip.

> While it is of interest to know that a failure causes a trip, it is also of interest to know whether a trip is actually needed and whether the trip lays all problems to rest. Board Ex. 1 at 6.

Oak Ridge added that the ICS controls the operation of equipment that is important during post-trip situations, but that the FMEA "does not pursue this necessary consideration". Id. ORNL illustrated this point by

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11. The evidence indicates that the less detailed analysis was selected to save time. Board Ex. 1 at 20.

pointing out that an ICS failure could possibly initiate a loss of main feedwater and inhibit auxiliary feedwater via the flow control valves, the same concern expressed by the NRC Staff in the <u>NRR Status Report</u>. Finding 47. Significantly, ORNL then stated "These possibilities are not addressed, presumably because they are plant specific". Board Ex. 1 at 6.

57. On May 7, 1980, the NRC Staff released its conclusions regarding the B&W FMEA and the ORNL review of it. In a cursory six page review,[12] the Staff concluded that:

1. it concurred with the Oak Ridge report;
2. "the actions of the ICS as a result of failures in related systems can lead to major plant upsets";
3. "the recommendations made by B&W, <u>if implemented</u>, could reduce the probability or consequences of these failures";
4. "there is a need to perform a broader study of B&W control systems to more adequately assess the role these systems play in transient initiation and mitigation";
5. "the timing of this study will be dependent on manpower availability";
6. "the results will probably not be available until the latter half of 1981"; <u>and</u>

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12. The six pages include a summary and three pages describing the B&W and ORNL reports. NRC Ex. 5.

7. "this schedule is acceptable because of the system improvements <u>which we anticipate will result</u> from implementation of the recommendations made in BAW-1564 [the B&W Reliability Analysis]." NRC Ex. 5 at 6 (emphasis supplied).

58. To date, SMUD has implemented only one of the B&W recommendations contained in the FMEA. Tr. 3702-07 (Capra).

59. Notwithstanding the FMEA and other actions taken at Rancho Seco since TMI, the basic concerns expressed by the Staff in the <u>NRR Status Report</u> and in CEC Exhibit 5 have not been resolved. For example, Licensee witness Karrasch testified that despite the FMEA, they were still not certain that the ICS could not cause a loss of both main and auxiliary feedwater. Tr. 693-94 (Karrasch). The Staff recently acknowledged that the ICS can cause a loss of both feedwater systems, though not necessarily simultaneously. NRC Ex. 4 at 5-57.

60. The Licensee's proposed findings on this issue suggest that the Board should consider the operating history of B&W plants in evaluating the adequacy of the FMEA. <u>See</u> Licensee's Findings at 31-32, para. 50-52. The May 7 Order required only the FMEA, and not the operating history of the ICS. We do not, therefore, consider the operating history a substitute for an adequate FMEA. The issue before the Board is whether the FMEA was adequate; we

reject Licensee's suggestion that weaknesses in the FMEA should be forgiven because of the operating history section of the report.

61. The Board is equally unimpressed by Licensee's suggestion that a more thorough review of the ICS was precluded by the time allowed. Licensee's Findings at 34-35, para. 55, n. 30. The FMEA was not a short-term requirement, and restart of Rancho Seco was not conditioned on its completion. May 7 Order. We conclude that the FMEA should have been a short-term requirement of the May 7 Order, however, as proposed in the NRR Status Report. We cannot agree with the Licensee's and Staff's rejection of the need to perform the analysis before resuming operation of the facility. It appears to this Board that completion of the FMEA was a logical and necessary predicate to reasonable assurance that potential ICS failures had been identified and that operators had been made aware of the potential situation.

62. The importance of the ICS in maintaining stable plant operation and in compensating for the responsiveness of the B&W design to feedwater transients suggests that it should be considered a safety system. Since this appears infeasible, an in-depth analysis and understanding of this system is of even greater importance than if it were a

safety system. In this context, the Board concludes that, with respect to Board Question H-C 16, the FMEA cannot be viewed as adequate, particularly in satisfaction of a long-term requirement. The many serious faults in the analysis identified in the ORNL review, and in particular the scope of FMEA, allow no other conclusion. The Board agrees with the Staff that there is a need to perform a broader and more detailed study of the Rancho Seco control systems to assess more adequately the role these systems play in transient initiation and mitigation. This study should include an equipment block failure modes and effects analysis of the ICS and related systems assuming single failures as well as the more likely and most serious multiple failures.

C. Auxiliary Feedwater System Reliability

Board Question CEC 1-6:

> Will the modifications of subparagraphs a-e of Section IV of the Commission's Order of May 7 still leave the Rancho Seco emergency feedwater system in a condition of low reliability?

63. The auxiliary feedwater system ("AWF") represents an extremely important means by which the facility may cope with anticipated transients and control the sensitivities inherent in the OTSG. NRC Ex. 4 at 5-36, 5-41. The AFW is designed to deliver cooling water to the OTSG in a timely and reliable manner after a feedwater transient. The more timely and reliable the system, then the less severe will be disturbances in the primary system due to the secondary system perturbations. CEC Ex. 26, Sec. 2. Accordingly, the questions of AFW reliability is, in our view, crucial to our overall concern for the adequacy of means to ameliorate the sensitivities inherent in the B&W system.

64. Rancho Seco has two AFW trains, each capable of supplying necessary cooling water to both OTSG's. Matthews Testimony on Reliability and Timeliness of the Emergency Feedwater System at 2, following Tr. 1163 ("Matthews AFW Testimony"). One train is motor driven and one has dual drives, both motor and steam. Id. at 3, Tr. 1491-93 (Matthews). The primary electric power source for the pump motors is offsite power but there are two diesel generators to provide power if offsite supplies are lost. Id. at 1495-97. The dual drive motor derives its steam from the steam lines

outside the OTSG. Id. at 1492-93. A more complete description of the Rancho Seco AFW system as it existed just after the TMI accident is contained in CEC Exhibit 26.

65. As documented by the May 7 Order, subsequent to the TMI accident there was not sufficient assurance that B&W reactors, including Rancho Seco, could be operated safely. An important reason for this lack of assurance was that B&W reactors place high reliance on the AFW Matthews AFW Testimony at 7; CEC Ex. 26 at 2-3 and 2-4. This is particularly true because of the rapid boil-dry time of the OTSG. Id. As a result, the NRC Staff concluded in late April, 1979, that the performance, i.e., timeliness, of AFW systems at B&W plants was marginal and that the reliability needed improvement at some plants. Id. at 2-10.

66. The NRC Staff determined that one short-term item necessary for safe operation of B&W plants, including Rancho Seco, was "[r]eviewing and upgrading, as appropriate, auxiliary feed reliability and performance (timeliness)". Id. at 1-7 (emphasis supplied).

67. As noted earlier, in late April 1979, SMUD was concerned that the NRC might shut down Rancho Seco without providing explicit criteria for restart. Finding 14. To avoid this situation, SMUD proposed to take various actions in response to the concerns expressed by the NRC Staff, particularly those set forth in CEC Exhibit 26, including several actions designed to upgrade the timeliness and reliability of the Rancho Seco AFW system. Tr. 2028

(Dieterich); CEC Ex. 25. The specific actions proposed by SMUD to upgrade its AFW system are set forth as items (a) and (b) on page 1 of CEC Exhibit 25, including items 1 through 9 in Enclosure 1 to CEC Exhibit 25. These actions were proposed to be completed prior to restart of the facility. CEC Ex. 25.

68. Despite the recommendation in CEC Exhibit 26 that licensees review and upgrade their AFW systems [finding 66], SMUD performed no detailed analysis or review of its AFW system to determine what short-term steps should be taken to upgrade the timeliness and reliability of its AFW system. Rather, SMUD reviewed proposals made by Duke Power Company with respect to the Oconee nuclear power units and then thought up similar items which might be applicable to Rancho Seco. Tr. 3254 (Rodriguez). An important criterion for determining what items should be included in the list proposed by SMUD were items which could be completed by early June, 1979, and therefore would ensure rapid restart of the facility. Id. at 3261-65.

69. In the Board's opinion, the AFW items set forth in CEC Exhibit 25 did not materially upgrade the timeliness or reliability of the Rancho Seco AFW system.[13] Rather, these items, for the most part, merely refined procedures

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13. This does not mean that certain AFW items were not more important than others. Id. at 3256-57. It does mean, as described in succeeding paragraphs, that the AFW upgrade items, taken as a whole, have not been shown to be very significant.

which already existed at Rancho Seco and covered actions which operators already were capable of performing. For example, the first item on Enclosure 1 to CEC Exhibit 25 states:

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

Testimony in this proceeding confirmed that Rancho Seco operators already knew how to do this and that Rancho Seco management personnel would have expected operators to take appropriate action even before this requirement was completed. Tr. 1526 (Matthews); 2044 (Dieterich); 3247 (Rodriguez).[14]

The second item states:

> To assure that AFW will be aligned in a timely manner to inject on all AFW demand events when in the surveillance test mode, procedures will be implemented and training conducted to provide an operator at the necessary valves in phone communications with the control room during the surveillance mode to carry out the valve alignment changes upon AFW demand events. CEC Ex. 25.

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14. Significantly, the short-term items did not provide for automatic loading of the AFW pumps in the event of offsite power loss. Such automatic loading would have increased the timeliness of AFW delivery [Tr. 1526-27] and the manual loading has subsequently been identified as a dominant contributor to AFW failure. CEC Ex. 20 at 13. If an AFW reliability study, discussed <u>infra</u>, had been performed prior to restart, the value of automatic loading could have been identified and this possibly would have been determined to have been a necessary short-term action.

This item ensures that the AFW will be available during the short test procedures which are run quarterly on the AFW system. Tr. 2045-46 (Dieterich). It does not affect AFW reliability or timeliness during all other occasions.

The third requirement specified that operators be trained to take manual control of the AFW system to control steam generator level in the event there was an ICS failure.[15] Testimony in this proceeding established that SMUD expected its operators to have been able to perform these actions even before the changes proposed in SMUD's April 27 letter. Tr. 1540 (Capra); 2039, 2047-48 (Dieterich); 3248 (Rodriguez).

The remaining items (4-9) in Enclosure 1 to CEC Exhibit 25 also did not materially improve AFW timeliness or reliability. Items 4 and 9 pertained to verification of certain facts and resulted in no AFW system changes or upgrade. Tr. 1541, et seq., 1559 (Matthews), 2048 (Dieterich). Item 6 related to rpocedures for providing alternate AFW water sources. Mr. Rodriguez testified that operators already knew how to do this [Tr. 3250], although there is doubt whether the promised procedural changes ever have been adequately completed. CEC Ex. 21, Enc. at 6. Finally, Items 5, 7 and 8 pertained to instrumentation for AFW flow verification and annunciation. While these items provide added instrumentation relating to AFW operation and hence reduce somewhat the possibility of operator error, Rancho Seco

15. The second item of the April 27, 1979 letter also pertained to manual control of AFW independent of the ICS. CEC Ex. 25 at 1. This requirement is basically redundant of the AFW upgrade set forth as item 3 of Enclosure 1 to CEC Exhibit 25. Tr. 1537-39 (Matthews, Novak); 2046 (Dieterich).

operators already had methods to verify the proper functioning of the AFW system. Tr. 1482, 1549, 1552, 1554 (Matthews); 2053 (Dieterich); 3249-52 (Rodriguez). Accordingly, the short-term AFW items contained in CEC Exhibit 25 and subsequently adopted in the May 7 Order cannot be viewed as a significant overall enhancement of that system.

70. The most important means of assessing and upgrading the timeliness and reliability of the Rancho Seco AFW system would have been through the performance of a thorough AFW reliability study. A significant value of an AFW reliability study would be the identification of success criteria and dominant failure contributors of that system and thus would permit identification of important AFW upgrade items and proper training of Rancho Seco operators in response to potential failures. Tr. 1560-61 (Matthews). Further, such a study would have accomplished the "review" of the AFW system called for in CEC Exhibit 26. Finding 66. Such a study was not proposed by SMUD in its April 27 letter, nor was one performed prior to restart of the facility, despite the fact that the NRC Staff prepared such analyses for Westinghouse and Combustion Engineering plants in a month or a little less in late Spring, 1979. Tr. 1578-79 (Matthews). If time had been no factor in restart of the facility, it would have been preferable to prepare an AFW reliability study prior to restart. Tr. 2078 (Dieterich).

71. SMUD has performed an AFW reliability study subsequent to Rancho Seco's restart. The study, using fault-tree techniques, compares Rancho Seco's reliability to that

of a Westinghouse PWR for three scenarios: Loss of main feedwater ("LOMF"); LOMF in conjunction with loss of off-site power; and LOMF with loss of all AC power. That study was communicated to the NRC on December 17, 1979 and was introduced into evidence as CEC Exhibit 20.

72. The NRC Staff has reviewed the Rancho Seco AFW study and has generally found it to be complete. CEC Ex. 21. However, the Staff found the study incomplete in its definition of success criteria. The study defined success as delivery of AFW to at least one OTSG with no time factor stated. Id. at 2. The NRC Staff stated that the success criteria should include the requirement to deliver AFW flow to one OTSG before the steam generator boils dry since that is a primary function of the AFW system. CEC Ex. 21, Enc. 1 at 1; Tr. 1597 (Matthews).

73. It is important to safe operation that AFW be delivered to the OTSG prior to boil dry. If boil dry occurs, there is a loss of heat sink, plus a rapid primary system response. This also means that more rapid operator actions may be required, which, in turn, increase the probability of operator errors. Tr. 1667 (Matthews); CEC Ex. 26 at 2-4; NRC Ex. 4 at 2-3; Finding 35. Such conditions should be avoided. Tr. 1488-89 (Novak). The avoidance of steam generator boil dry is a reasonable criterion to insist upon. As early as April, 1979, the NRC Staff had expressed its serious concerns on this subject:

Once the steam generator substantially dries out, the reactor system will heat up. The potential for voids in the primary system increases. The reactor pressure may go up to the point where the PORV lifts. Eventually, if natural circulation is not restored or if auxiliary feedwater is not made effective, then core cooling will be dependent on initiation (manually) of the high pressure injection (HPI) system of ECCS. CEC Ex. 26 at 1-2.

74. SMUD initially committed to the NRC to revise the AFW study to include NRC's recommended success criterion of avoiding steam generator boil dry. CEC Ex. 22, Attach. at 1; Tr. 2107 (Dieterich). Subsequently, SMUD has determined that it will not upgrade the study, apparently based on the view that OTSG boil dry is not a safety concern. Tr. 2088-89 (Dieterich).

75. The NRC Staff also has requested SMUD to verify and revise, as necessary, procedures for AFW operation in the event of a loss of all AC power. CEC Ex. 21, Enc. 1 at 8. Although, SMUD originally committed to do so [CEC Ex. 22, Attach. at 3], SMUD Subsequently has refused to carry out this commitment because the loss of all AC power is allegedly beyond design basis. Tr. 2355 (Dieterich).

76. The loss of all AC power in conjunction with a LOMF is admittedly a low porbability event. However, we believe procedures should be prepared to guide operators in that situation, due to the potential severe consequences. On loss of all AC power, the only way to cool the core would be with the heat sink provided by the OTSF since HPI would be lost. Tr. 1587-88 (Matthews). If all AC power were lost, then only the steam driven AFW train could

provide that heat sink. If that train were, for some reason, unavailable (as due to routine maintenance), there would be no way to cool the core on loss of all AC power. Tr. 2366-67 (Dieterich). It would appear sensible for SMUD to have explicit procedures addressed to this situation, such as to require both diesels to be available whenever the steam drive for the AFW is not available.[16] This would even further reduce the risk that a loss of all AC power might occur. We decline to order any particular procedure but do believe that SMUD, as requested by the NRC in CEC Exhibit 21, should establish procedures relating to loss of all AC power.

77. The rapid boil-dry time of the OTSG and consequent quick response of the primary system demands an "<u>extremely reliable</u>" AFW system. NRC Ex. 4 at 5-36, 5-41; Tr. 1489-90 (Novak). In our view, the need for an extremely reliable AFS system makes it appropriate to require the Rancho Seco system to be better than that at PWR's of other design. Tr. 1487 (Matthews). This is further supported by the NRC Staff finding that "the timing requirements for AFW delivery are substantially more stringent for B&W plants than for others." CEC Ex. 26 at 2-4. Indeed, Licensee witness Dieterich stated that a

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16. Rancho Seco is permitted to operate with one diesel out of service for 30 days and with one AFW train out of service for 48 hours. Rancho Seco procedures do not distinguish between steam and motor driven trains in terms of being out of service. Tr. 1499, 1509-10, 1753 (Matthews); 1512 (Capra); 2065-66, 2366-67 (Dieterich).

B&W AFW system should have quicker water delivery than a non-B&W PWR. Tr. 2041 (Dieterich).

78. The AFW reliability study compares the Rancho Seco AFW system to systems employed by Westinghouse PWR's. The results of the comparison, as set forth in CEC Exhibit 20, demonstrate that the Rancho Seco AFW system is no more reliable than that of other PWR's and, indeed, is less reliable for certain cases. Those results are:

	Rancho Seco Reliability Compared to Westinghouse PWR
Case 1 (loss of main feed):	medium to high
Case 2 (loss of main feed plus loss of offsite power)	low to medium
Case 3 (loss of main feed plus loss of offsite power)	medium

If the AFW success criterion were revised to be no boil dry of the OTSG, the Rancho Seco results set forth above would tend to move toward less reliability in comparison to the Westinghouse PWR, given the fact that Westinghouse PWR's have about 30 minutes to steam generator boil dry while B&W PWR's have only 4 minutes. Tr. 1608, 1660-61 (Matthews); 1490 (Novak).

79. Licensee has stressed that Rancho Seco's AFW system has had a perfect operating history and, therefore, that no further AFW upgrading needs to be accomplished. Rodriguez Testimony at 49; Tr. 3255 (Rodriguez). We do not question that the AFW system has had a good record. But we are not convinced that this operating history justifies

a conclusion that no further improvement is necessary. The extreme importance of the AFW system to a B&W NSSS makes us inclined to order improvements even in light of a good operating history. However, we do not accept, without qualification, the assertion that Rancho Seco's AFW operating record is perfect. During the well-known "light-bulb" incident at Rancho Seco, there was a boil dry of at least one, and perhaps both, OTSG.

Whether this was an AFW "failure" or not, it indicates that the AFW system or systems which affect that system may impede AFW delivery. Tr. 3308 (Rodriguez).

80. Licensee has committed to further upgrade its AFW system so that it is entirely safety grade. Tr. 2098-99 (Dieterich). This upgrade, while certainly important to the overall reliability of the Rancho Seco AFW, does not substitute for the need for a revised AFW study using realistic success criteria. However, it does lead us to believe that the revised study which we order in this decision should be delayed until the AFW upgrades are completed in the first half of 1981, so that the new study will accurately analyze the AFW system as it, in fact, will exist.

81. We reach the following conclusions regarding AFW system timeliness and reliability:

(a) The short-term AFW items enumerated in SMUD's April 27, 1979 letter and confirmed in the NRC's May 7 Order were not adequate to ensure timely and reliable AFW performance. These items were not chosen with careful analysis of AFW strengths and weaknesses and did not materially upgrade the Rancho Seco AFW system. A thorough reliability study should have been performed prior to restart so that appropriate actions could have been identified.

(b) The AFW reliability study is not complete and should be upgraded in accordance with the comments of the NRC Staff set forth in CEC Exhibit 21, particularly with revision of the success criterion to provide for AFW delivery prior to steam generator boil dry. This revised study should be completed within six months of the anticipated upgrade of the AFW system to safety grade. If the revised study reveals deficiencies that keep the Rancho Seco system from being more reliable than the systems at Westinghouse PWR's (as "reliable" is used in the AFW study), then the Rancho Seco system will promptly be upgraded so that it is more reliable than the Westinghouse systems.

(c) SMUD should verify and revise, as necessary, procedures for AFW operation in the event of a loss of all AC power.

D. Frequency of Feedwater Transients

Board Question FOE III(a):

> The NRC orders in issue do not reasonably assure adequate safety because the orders fail to evaluate or comment upon the acceptability of 27 feedwater transients over the past year in nine Babcock & Wilcox (B&W) reactors, a frequency which is 50 percent greater than the corresponding rate for other pressurized reactors.

82. This contention was apparently based upon a study conducted by the NRC Staff shortly after the TMI accident. This study was cursory in nature and was conducted to see if a vast difference in feedwater-related malfunctions existed between B&W facilities and other PWR designs. The study revealed that the nine operating B&W facilities had experienced 27 feedwater transients in the year preceding TMI. Rubin and Novak Testimony on Acceptability of Feedwater Transients Referenced in NUREG-0560 at 3, following Tr.1163 ("Rubin and Novak Feedwater Transients Testimony").

83. The results of this study suggest that feedwater transients occur somewhat more frequently in B&W facilities than in other facilities. The 27 such events identified in the Staff study were "somewhat larger" than the number of such events experienced by other reactors. Id. NUREG-0560, which describes the study, states that B&W facilities experience three such events per year, compared to two for other PWR designs. Webb Testimony at 5.

84. Witnesses for the Licensee testified that the frequency of feedwater transients in B&W plants causing a reactor trip in the year preceding TMI was not higher than other facilities, but

they could not testify as to the relative frequencies of such events at B&W facilities in other years. Karrasch and Jones Testimony at 13-14; Tr. 741-42 (Karrasch). However, due to various changes in Rancho Seco operation (the revised setpoints for PORV actuation and high pressure reactor trip plus the anticipatory reactor trip), feedwater transients are more likely to cause a reactor trip at Rancho Seco now than before TMI. Id. at 756. For this reason, a comparison of feedwater transients causing reactor trips before TMI is not a valid indication of the frequency of such events at Rancho Seco today. Nor is it valid to compare such transients causing reactor trips at B&W facilities to those of other vendors before TMI, since at that time most non-B&W reactors had anticipatory trips and B&W facilities did not. CEC Ex. 26 at 2-3.

85. Mr. Capra testified that in response to an interrogatory he had compared the number of feedwater transients in various PWR designs since TMI. During this eight month period, Mr. Capra's review revealed that Combustion Engineering plants experienced more such events than B&W plants, and that Westinghouse plants had the fewest of the three. Tr. 3754 (Capra).

86. We believe that determination of whether B&W plants have or do not have more feedwater transients than other plants is not terribly crucial in view of the fact that the evidence indicates that the numbers are roughly comparable to other designs. More important, in our view, is the question of whether the number occurring at B&W plants is acceptable. In this regard, the evidence supports a finding that there is genuine cause for

concern. One of the Staff witnesses co-authored a document that concluded "[r]egardless of the reasons, B&W plants are currently experiencing a number of feedwater transients which the staff feels are undesirable". CEC Ex. 5, section VI, Conclusions. And NRC Staff witness Capra summarized remarks on this subject made by Harold Denton, Director of NRC's Office of Nuclear Reactor Regulation, at an April 3, 1980 meeting with B&W licensees as follows:

> In the relatively short period of commercial operation of B&W plants, approximately 38 reactor years, there have been too many undesirable incidents involving B&W design reactors.
>
> * * * * *
>
> . . . he [Denton] encouraged B&W and the licensees to personally pursue ways to improve their safety record and, in particular, improve plant response to operational transients, such as loss of feedwater events.
>
> <u>This will be necessary to support long-term operation of the licensed plants</u> . . ." Tr. 1266-67 (Capra) (emphasis supplied).

87. When asked if he concurred with Mr. Denton's statement that B&W facilities had experienced an undesirable number of transients, Mr. Capra testified that the NRC has no criteria for acceptable numbers of such transients. When asked if he found the transient history of B&W facilities acceptable, Mr. Capra testified:

> I can't really say whether that is acceptable or not. Personally, I don't think it is . . . a good idea to me to have transients of that nature, such as Crystal River Three or TMI. Tr. 1268 (Capra).

88. The evidence on this contention must be viewed as somewhat inconclusive. The Board does not agree with the Licensee's view that only feedwater transients causing reactor trips merit concern, especially since the presence of anticipatory reactor trips on non-B&W facilities before TMI invalidates Licensee's comparison. In the face of the evidence before us, the Board must decide the issue against the party with the burden of proof, in this case the Licensee. Thus we conclude that Rancho Seco is somewhat more prone to feedwater transients than reactors of different designs. More important, we reiterate our view that these transients must be viewed in the context of the sensitivities of the B&W NSSS. Therefore, it is highly desirable to reduce their numbers so that there is less risk of safety system challenges.

E. <u>Safety System Challenges</u>

<u>CEC 1-1</u>:

Despite the modifications and actions of Subparagraphs (a) through (e) of Section IV of the Commission's Order, will reliance upon the High Pressure Injection System to mitigate pressure and volume control sensitivities in the Rancho Seco primary system result in increased challenges to safety systems beyond the original design and licensing basis of the facility?

<u>CEC 1-12</u>:

Despite or because of the modifications and actions of Subparagraphs (a) through (e) of Section IV of the Commission's Order of May 7, will Rancho Seco experience 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?

89. These issues raise concerns regarding the frequency of challenges to the high pressure injection system and the frequency of reactor trips. As a general rule, challenges to safety systems should be minimized. Lewis Testimony at 12; Tr. 499 (Lewis). This is because increased challenges proportionally increase the probability of safety system failure. <u>E.g.</u>, CEC Ex. 26 at 2-7; Tr. 757 (Jones); Tr. 3693-3694 (Capra). Additionally, these systems are designed for a finite number of challenges, and if that number is reached before the facility is to be decommissioned, it cannot continue operating unless the system is shown to be capable of safely withstanding further challenges. <u>E.g.</u>, Tr. 2013-19 (Dieterich).

90. With respect to the increased number of reactor trips (CEC 1-12), the Licensee submitted direct testimony.

Karrasch and Jones at 39-41. However, although the Licensee bears the burden of proof on this issue, neither its direct testimony nor its proposed decision addresses the concern set forth in Issue CEC 1-1: that the frequency of high pressure injection system operation at Rancho Seco is in excess of that assumed during the design and licensing of the facility.

91. Witnesses for the Licensee testified with respect to reactor trips that the measures required by the May 7 Order will increase these events. Karrasch and Jones at 39.[17] They further testified that the increase was not expected to cause B&W facilities to experience more such events than the industry average or to exceed the frequency assumed in the design and licensing of the plant. Id. at 40-41.

92. The prediction that reactor trips would increase at Rancho Seco as a result of the May 7 Order requirements was shared by the other witnesses in the proceeding. Rubin and Novak Design Basis Testimony at 3; Webb Testimony at 5. It was further confirmed by a Staff survey of the rate of such events in B&W facilities after implementation of these measures. This survey indicates that these measures have increased reactor trip frequency by 15 percent in all the B&W facilities. NRC Ex. 4 at 4-14. For Rancho Seco, the survey indicates that the changes

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17. The measures expected to increase reactor trips are the anticipatory reactor trip and the lowering of the high pressure reactor trip set point. Rubin and Novak Testimony on "The Design Basis for Rancho Seco Safety Systems" at 3, following Tr. 1163 (Rubin and Novak Design Basis Testimony").

have increased reactor trip frequency by more than 100 percent. B&W facilities have been operating with the revised set points and the anticipatory trip for only a short time, and therefore these numbers will undoubtedly change. But it is clear that the measures required by the May 7 Order have significantly increased the frequency of reactor trips at Rancho Seco. Id. At the 95 percent confidence level, Rancho Seco shows a significantly higher total trip frequency as a result of the May 7 Order measures. Id. at 4-12.

93. Whether this significant increase in reactor trips will result in a higher frequency of reactor trips than the design basis for the reactor protection system is unclear.[18] But any significant increase in trips increases the risk of a scram failure accident. As Dr. Lewis testified:

> [T]he inversion of the PORV set point and the scram set point may, if we are sitting here 10 years from now, turn out to have not been a wise thing to do because the extra challenges to scram are acceptable because they haven't had any scram failures, they would sure become unacceptable if we had one. Tr. 523 (Lewis).

94. Based on this evidence, the Board finds with respect to CEC 1-1 that the measures required by the May 7 Order have significantly increased the frequency of

18. As noted in Finding 91, Licensee's witnesses testified it would not. On the other hand, the Staff survey suggests otherwise. The survey shows that Rancho Seco is now experiencing 0.88 trips per month, compared to a design basis frequency of 0.83 trips per month. NRC Ex. 4 at 4-14.

reactor trips at Rancho Seco, and therefore increased the likelihood of a scram failure accident. While it is unclear whether the increased frequency is in excess of the design basis of the facility, it is evident that the increased frequency is undesirable.

95. Although these challenges to scram are undesirable, the Board is mindful that the anticipatory trip and revised set points were intended to serve a safety function. We have already described the short dry-out time of the OTSG when feedwater is lost, and noted that the anticipatory trip serves to extend this time somewhat. The Board believes this additional margin is sufficiently desirable to warrant the increase in reactor trips, although we reiterate the need to explore other remedies that will reduce the OTSG sensitivity. Unless such remedies are found, however, the Board believes the anticipatory trip does more good than harm.

96. The revised set points for the PORV and high pressure reactor trip, on the other hand, do not appear wise. While these changes have the benefit of reducing challenges to the PORV, which decreases the possibility of PORV failure, they increase challenges to scram. In addition, the revised PORV setpoint also means that the PORV will no longer provide an effective venting device to avoid reaching the pressurizer safety valves' setpoint of 2500 psig. Rather, it is likely that when pressure reaches a level high enough to actuate the PORV, the safety valves may also be actuated. The Board will

discuss in the next section its concern with challenging safety valves in discussing feed and bleed cooling.

97. The record shows that there is a method of reducing PORV failures without increasing challenges to scram or the safety valves. It appears possible to increase the reliability of the PORV by making the PORV and related systems safety grade. Tr. 1647-48 (Novak); 2123-24 (Dieterich). A proposal for such a PORV fix has been made by Consumer's Power but has not yet been acted upon by the NRC. NRC Ex. 4 at 5-29. There is no reason that Rancho could not implement such a PORV fix. Tr. 2123-24 (Dieterich). The Board therefore further finds, with respect to CEC Issue 1-1, that SMUD should upgrade the PORV to safety grade and shall then seek NRC permission to return the PORV and high pressure trip setpoints to their original pressure values.

98. With respect to the frequency of challenges to the high pressure injection system (CEC 1-12), it appears that the post-TMI changes at Rancho Seco have increased these events as well. Staff witness Novak testified that "[S]ince there has been an increase in reactor trips, an increase in HPI actuation is also likely." Rubin and Novak Design Basis Testimony at 3.

99. Even if HPI actuation had not become more frequent since TMI, this safety system is being used much more often than its design basis frequency. The HPI system was designed for 40 challenges over the 40-year

life of Rancho Seco, or one per year. Tr. 995, 997 (Karrasch). In fact, it has been called upon some thirty times in only the first six years of operation. This corresponds to a frequency some five times greater than was expected when Rancho Seco was licensed. Tr. 1159 (Rubin); 2013-18 (Dieterich); 3358 (Rodriguez).

100. As discussed previously, the overall probability of system failure is a product of the number of challenges times the reliability of the system. Accordingly, the probability of HPI failure can be fairly estimated to be several times greater for Rancho Seco than was expected when it was licensed.

101. It is also evident, if the present rate of HPI challenges continues, that Rancho Seco will exhaust the allowable cycles for this system long before the plant is to be decommissioned. Whether this will mean installing new HPI nozzles is unclear; the Licensee's witnesses suggested that there may be other, less drastic remedies. Tr. 2014-18 (Dieterich).

102. The cause of this dramatically higher frequency of HPI actuation is that operators are routinely using this safety system to control the primary system disturbances caused by the OTSG design. The operators acknowledged that it is ordinary practice to manually actuate an HPI pump following a reactor trip. CEC Ex. 37 at 57; CEC Ex. 38 at 57; Webb Testimony at 9; Tr. 2012, 2019 (Dieterich). This routine use of HPI is undesirable and should be remedied. NRC Ex. 4 at 5-13.

103. Therefore, with regard to CEC Issue 1-12, the Board concludes that the measures required by the May 7 Order have increased the use of HPI. The Board also finds that Rancho Seco is experiencing a much greater frequency of challenges to the high pressure injection system than was envisioned when it was licensed.

F. Loss of Coolant Accidents and Natural Circulation Cooling

Board Question CEC 1-4:

We note (letter D. Ross to J.J. Mattimoe, December 14, 1979) that there is still some dispute as to the fundamental logic for Reactor Cooling 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?

Board Question CEC 1-2:

Can poor understanding of natural convection in the Rancho Seco system result in a situation that will lead to inadequate cooling despite the modifications and actions of Subparagraphs a-e?

Board Question CEC 1-10:

Is the physical configuration of the Rancho Seco primary system such as to permit unsafe accumulation of steam or other gases despite the modifications and actions of Subparagraphs a-e?

Board Question H-C 24:

Rancho Seco, being a Babcock and Wilcox designed reactor, is unable to avoid or control bubble formation in the primary system which may occur subsequent to a loss of feedwater accident, and therefore, is unsafe and endangers the health ar . safety of Petitioners and the public.

104. During this proceeding, a great deal of direct testimony and cross-examination addressed the broad question of maintaining adequate core cooling where for ed circulation is not available. This examination covered a wide range, including natural circulation cocl ng where no system failures are present, natural circulation in

conjunction with loss of coolant accidents ("LOCA's"), core cooling where there are significant voids in the primary system, core cooling where there is no secondary side heat sink, and adequacy of operator training to respond to the requirements of these situations.

105. A brief background narrative helps put these issues into perspective. The genesis for the extended inquiry into natural circulation and small break LOCA's was, of course, the TMI accident, which involved a small break LOCA and the failure to establish natural circulation under highly voided conditions.[19] That accident demonstrated, at a minimum, that greater emphasis needed to be placed on understanding small break LOCA's and natural circulation. It was in response to this accident and NRC investigations that the May 7 Order included the requirement that SMUD "[c]omplete analyses for potential small breaks and develop and implement operating instructions to define operator actions". May 7 Order.

106. In response to the May 7 Order requirements, new small break analyses were performed and operator training and procedures particularly relating to natural circulation, were developed. Norian Testimony at 4. These new procedures, as well as directives contained in

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19. Licensee in its proposed findings argued that there is no indication that TMI operators had inadequate understanding of natural circulation that contributed to the severity of the accident. Licensee Finding 104. We disagree. The operators' failure, though due to voided conditions, was a failure to understand necessary preconditions for natural circulation which resulted in their failure to recognize inadequate cooling.

I&E Bulletins 79-05A and 79-05B,[20] all were in effect when Rancho Seco resumed operation in early July, 1979.

107. Subsequent to restart of the facility, the NRC issued I&E Bulletin 79-05C, which fundamentally changed existing operating instructions. This Bulletin was based upon revised analyses which revealed that adequate core cooling could not be assured if certain size small breaks were to occur and the reactor coolant pumps were tripped more than a few minutes into the transient. I&E Bulletin 79-05C; Norian Testimony at 3-4; Karrasch and Jones Testimony at 63. Due to the difficulty of distinguishing between small break LOCA's and severe overcooling events, the NRC ordered that reactor coolant pumps be tripped immediately whenever primary system pressure falls to the high pressure injection ("HPI") setpoint. I&E Bulletin 79-05C. At that point, cooling must be provided by natural circulation until 50° subcooling can be verified and stable conditions exist. Id.

108. The upshot of the TMI accident and I&E Bulletin 79-05C is that there is significantly greater reliance on natural circulation cooling. While such a cooling mode has always been considered in the licensing of plants, it has heretofore been considered a distinctly secondary cooling technique. Tr. 885 (Karrasch).

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20. The requirements of the I&E Bulletins described in this decision appear in NRC Ex. 4, Appendix A.

109. The accepted issues and contentions quoted at the outset of this section typify the debate that has arisen from these events. Those questions which we deem most significant and which we address below are as follows:

-- When the primary system is in a subcooled state, is natural circulation a reliable means of providing cooling? We answer in the affirmative.

-- When the primary system is in a voided condition, as after a small break LOCA, is natural circulation a reliable means of providing cooling? We answer in the negative.

-- Whether the existing emergency core cooling analyses for Rancho Seco required by 10 C.F.R. §50.46 are still adequate in view of the analyses underlying I&E Bulletin 79-05C. We answer in the negative.

-- Whether there is a need to revise I&E Bulletin 79-05C criteria to avoid unnecessary reactor coolant pump trips. We answer in the affirmative.

-- Whether the increased reliance on natural circulation cooling modes and the reactor coolant pump trip requirement have imposed significant new responsibilities on operators. We conclude that these have imposed new responsibilities. The ability of operators to handle these responsibilities is analyzed in succeeding sections.

(i) <u>Natural Circulation Cooling in a Subcooled Primary System</u>

110. When the reactor coolant pumps are not operating, coolant flow must occur naturally. To date, Rancho Seco has never used natural circulation cooling. CEC Ex. 1, Admission No. 65. However, natural circulation has been successfully achieved on several occasions in lowered loop B&W reactors like Rancho Seco, twice following unplanned losses of off-site power. Karrasch and Jones Testimony at 35.

111. Natural circulation results from the density difference between the coolant heated by the core and that cooled in the steam generator. If the thermal center of the steam generator is elevated above that of the core, gravity will pull the cooled coolant down toward the core because it is more dense. The cooled coolant forces the heated coolant ahead of it up toward the steam generator. The coolant pulled down to the core is heated, and the coolant pushed up to the steam generator is cooled, perpetuating this process and creating a continuous flow equal to approximately 2 to 4 percent of that achieved through use of the reactor coolant pumps. Id. at 33-34; Norian Testimony at 2-3.

112. Natural circulation depends upon three things. First, there must be cooling in the steam generators, which means there must be auxiliary feedwater on the secondary side of the OTSG. Second, there must be a sufficient elevation difference between the thermal center of the core and that of the steam generators. Third, there must be an unbroken train of liquid between the steam generator and the core. Lewis Testimony at 9-11. Provided these conditions are satisfied, natural circulation provides a reliable method of core cooling. No witness disputed this fact. At the same time, no witness stressed that natural circulation is a preferred mode of cooling. It always would be preferable to be able to rely upon forced circulation cooling because such

reliance provides greater defense in depth. Webb Testimony at 9-12.[21]

113. A further reason that natural circulation cooling is not a preferred cooling mode is that it provides additional possibility for operator errors. While operator action is not normally required to establish natural circulation cooling (assuming AFW is established), operators must verify that it has occurred and take appropriate action if it cannot be verified. Karrasch and Jones Testimony at 37-38; Rodriguez Testimony at 52-53. This is not necessarily a simple matter, as demonstrated by the fact that Rancho Seco operators initially exhibited poor understanding of verification of natural circulation cooling when audited by the NRC Staff in early June, 1979. Wilson Testimony at 7.

(ii) <u>Natural Circulation in a Voided Primary System</u>

114. Operators might be called upon to establish natural circulation when voids are present in the primary system. Such voiding could be caused by a severe overcooling event which might lead the pressurizer to empty by a severe overheating transient such as an extended feedwater loss which would cause steam to be created in the primary system, or by a LOCA due to inventory and pressure reduction. Karrasch and Jones Testimony at 43;

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21. In addition, when the coolant pumps are tripped, operators also lose the pressurizer sprays which greatly improve plant pressure control. NRC Ex. 4 at 5-30, 5-31.

Lewis Testimony at 10.[22]

115. When gas is introduced into the primary coolant from the boiling of the coolant, the gas will form "voids" in the reactor coolant system. Natural circulation has not been tested in a PWR during conditions of significant voiding, and both the Staff and Licensee admitted that natural circulation is unreliable once significant voiding occurs in the primary system. CEC Ex. 1, Admission No. 36; CEC Ex. 2, Admission No. 34; Lewis Testimony at 9-11; Norian Testimony at 3; Tr. 803 (Karrasch and Jones).

116. Force circulation cooling (that is, use of the reactor coolant pumps) is much more reliable than natural circulation cooling when there is significant voiding in the primary system. CEC Ex. 1, Admission No. 37; CEC Ex. 2, Admission No. 35; Tr. 1329 (Norian).

117. Voiding can also occur from the introduction of noncondensible gases into the primary system. Typical sources include the nitrogen used to pressurize the core flooding tanks, hydrogen dissolved in the primary system and borated water storage tank fluid, hydrogen produced by the zirconium-water reaction, and helium used to pressurize the fuel rods. Norian Testimony at 3.

118. Witnesses have testified that even in voided conditions, a form of natural circulation cooling can be

22. As a result of the inversion of the setpoints for PORV actuation and high pressure reactor trip, as well as the anticipatory reactor trip, challenges to the PORV have been substantially reduced. This will decrease the likelihood of a small break LOCA resulting from a stuck open PORV. NRC Ex. 2 at 2-1.

maintained. If there are only a few voids, circulation will continue in its normal form. Id. at 3. If natural circulation is blocked by steam voids, it may be possible to cool the core by "pool boiling" or "reflux boiling". Lewis Testimony at 10-12. In this circumstance, the coolant near the core boils and the steam circulates to the OTSG, where it is cooled and condensed and returns as liquid to the core. Norian Testimony at 3; Tr. 797 (Karrasch and Jones)

119. Reflux boiling has never been tested in a PWR. Id. at 803. The PWR industry has not provided any data to experimentally verify analytical predictions of reflux boiling. NRC Ex. 2 at 2-7; CEC Ex. 1, Admission No. 51; Norian Testimony at 4.

120. When attempting reflux boiling, it is prudent to raise the secondary (feedwater) level in the OTSG to 95 percent on the operating range. The effectiveness of reflux boiling is uncertain when the secondary level is only 50 percent on the operating range. Tr. 820 (Karrasch and Jones).

121. The ICS automatically controls the feedwater level in the OTSG to 50 percent on the operating range when natural circulation is attempted. Operators are instructed to raise the level to 95 percent manually if natural circulation does not occur. Id. at 834-5.

122. If there is no heat removal available through the OTSG's, either because there is no feedwater or there are non-condensibles preventing heat transfer,

the only remaining method of core cooling is the so-called "feed and bleed" method. This is available only in plants with high head pressure HPI systems, such as Rancho Seco. Lewis Testimony at 11. This mode of core cooling relies upon heat rejection through the PORV and/or safety valves, which is accomplished by allowing the RCS to pressurize to the set points for these valves. The lost coolant is replaced by the HPI system. Id. at 11-12; Norian Testimony at 7.

123. Energy Commission witness Lewis testified that the feed and bleed concept is theoretically effective, but has not been thoroughly analyzed. He also testified that "as a long term cooling mode, it is not clear how many actuations of these various valves are prudent." Dr. Lewis concluded that feed and bleed cooling "must be regarded as a theoretically practical means of core cooling, to be used in extremis, until secondary cooling is restored." Lewis Testimony at 11-12.

124. Dr. Lewis envisioned feed and bleed as the discharge of "steam through the PORV and, perhaps, the safety valves, at a rate sufficient to remove decay heat from the system." Id. He did not approve of feeding and bleeding through the safety valves: "Finally, to the extent that the core (sic) safety valves would be involved, it is never prudent to use a safety item in a normal operating mode." Id.; see also Tr. 499 (Lewis). The Licensee's witness agreed that it is better to avoid

using the safety valves and to use only the PORV, if possible. Tr. 745 (Karrasch and Jones).

125. If solid water or a two phase mixture is being rejected through the valves rather than steam, it appears that the PORV alone does not have sufficient capacity to remove all the decay heat. Tr. 507 (Lewis). Thus, if one is rejecting liquid or two-phase coolant, the feed and bleed mode requires repeated exercising of the safety valves.

126. Although Dr. Lewis envisioned feeding and bleeding steam only in his testimony, he acknowledged that others envision the bleeding of liquid or two phase coolant. Id. at 485. Dr. Lewis also testified that one can use the feed and bleed mode he envisioned only if one has core coolant level indication. Id. at 511-12. Rancho Seco does not yet have such indication. Id. at 508. Thus, in response to a question from the Board, Dr. Lewis acknowledged that there is little guidance available to an operator regarding this type of cooling:

> (By Dr. Cole): "I have trouble visualizing how an operator would know when to bleed and when to feed and can you provide us with any guidance in that regard, sir?"
>
> A: "No, I think he would have a great deal of trouble knowing when to bleed and when to feed. I have not heard what really happened at Crystal River but it may well be that one was bleeding and feeding continuously at Crystal River. That is hearsay, I really don't have any direct information about what happened there. But, one might then err on the side of just feeding all the time, in which case, one would be pushing solid water through the PORV and the safety

> valves. The operator does need more information before he can do intermittent feeding and bleeding." Id. at 527.

Therefore, the only feed and bleed capability available at Rancho Seco requires use of the safety valves. Both the Licensee and Staff witness stated the feed and bleed method would involve rejection of liquid or two-phase coolant through the PORV and safety valves. Tr. 956-58 (Karrasch and Jones); 1332-34 (Norian).

127. PORV and safety valves have not yet been analyzed or tested to determine how they perform when passing liquid or two-phase coolant. Tr. 498-99 (Lewis); 1334 (Norian). Accordingly, there is considerable uncertainty regarding what the operators should expect in this form of the feed and bleed mode as well. Tr. 498-99 (Lewis).

128. The Licensee has admitted it is unaware of feed and bleed cooling being tested, demonstrated, or even attempted in a PWR like Rancho Seco. CEC Ex. 1, Admission Nos. 46 and 47. The NRC Staff has claimed it could neither admit nor deny these facts. CEC Ex. 2, Admission Nos. 40 through 45.

129. One of the concerns regarding the feed and bleed mode is that it could cause a safety valve to fail in an open position. Tr. 1340 (Norian). There are no block valves for the safety valves once they are opened, and there is no way to close them from outside the containment building. Tr. 745 (Karrasch and Jones); 1339 (Norian). Thus, if a safety valve were to stick open,

there is no alternative except to discharge coolant through it until the pressure in the primary system is brought down to some very low level. Id. at 1335. This may take several hours. Moreover, this process assumes that cooling through the steam generators is restored. Id.[23] If it is not restored, Mr. Norian testified that:

> . . . you are just going to stay in that mode until you get a heat sink. And if you do not get a heat sink, you are going to continue to put water out that valve. Tr. 1336

Mr. Norian and Licensee's witnesses agreed that a cold shutdown state could not be reached if a secondary heat sink could not be restored. Id.; Tr. 960-61 (Karrasch and Jones).

130. The B&W small break LOCA analysis indicates that, using standard Appendix K assumptions (i.e. only one HPI train available), adequate core cooling may not be assured in the event of a small break LOCA accompanied by a loss of all feedwater. Karrasch and Jones testimony at 59; NRC Ex. 2 at 2-4; Tr. 1017-18 (Jones). This, again, casts severe uncertainty on the feed and bleed cooling mode. Further, the B&W witnesses confirmed that where solid water natural circulation does not occur, it is unwise to attempt feed and bleed cooling.

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23. The feed and bleed mode is generally postulated for the circumstance when there is no such secondary system cooling. Thus, if one assumes that the safety valve has stuck open during this mode of cooling, it is fair to assume that cooling in OTSG has been lost. Tr. 743, 795 (Karrasch and Jones).

(Karrasch and Jones).

131. On February 26, 1980, after the admissions described in the previous finding were filed, the Crystal River Unit 3 facility in Florida (a B&W reactor like Rancho Seco) experienced a loss of non-nuclear instrumentation that resulted in improper input signals to the ICS. Tr. 365-432 (Novak); 434-60 (Karrasch). For a period of time during that event, operators maintained enough HPI flow to force steam, liquid, and two-phase coolant through the PORV and safety valves. Tr. 1334 (Norian). This action resembled feed and bleed cooling, though it appears that a secondary system heat sink was available throughout the transient. Tr. 394-5 (Novak). Significantly, a safety valve may not have properly closed for a period of time early in this event. Tr. 414-15 (Novak).

132. The theoretical and untested status of both reflux boiling and feed and bleed cooling was reflected in the depositions of the Rancho Seco operators, who were confused regarding the practical application of both techniques. The operator, for example, testified that it was always a good idea to close a stuck valve. CEC Ex. 38 at 22. When asked whether he could envision any circumstances where one would want to maintain the presence of a small break (i.e. in feed and bleed), the operator repeated that he could not. Id. When specifically asked if he would close the valve even though there was no feedwater (and

having heard the question repeated at his attorney's request and having correctly repeated the hypothetical situation under consideration himself), he still replied: "If it is closable, you close it." Id. at 22-23. This is incorrect, for without feedwater, one must reject heat through the valve. Much later in the deposition, after two opportunities to discuss his answers with Licensee's counsel and its Manager of Nuclear Operations [Id. at 34 and 64], his attorney again repeated the question and the operator changed his answer to correctly describe the feed and bleed technique. Id. at 76.

133. Similarly, the senior operator gave contradictory testimony regarding the relative merits of reflux boiling versus feed and bleed cooling. He testified first that feed and bleed is the more preferable mode of cooling. CEC Ex. 36 at 76. A few moments later, he testified that reflux boiling is more desirable. Id. at 77

134. The foregoing examples suggest that operators do not clearly understand, in practical terms, these cooling modes. This is not surprising since, as discussed in this Decision, there is considerable uncertainty regarding what operators should expect in these situations.

135. In conclusion, this Board finds that there is considerable uncertainty surrounding the adequacy of core cooling in significantly voided conditions where reactor coolant pumps are not available. While analyses and testimony have been received which indicate that core cooling can be maintained, we cannot find that a prepon-

derance of probative evidence supports a finding that adequate core cooling can reliably be maintained in conditions of significant voiding. Moreover, it appears that whatever theoretical application reflux boiling and feed and bleed cooling may have, operators as a practical matter would have significant difficulty in either mode. This conclusion leads us to state further that forced circulation cooling is highly preferable to natural circulation cooling under significantly voided conditions, except for the problems identified in I&E Bulletin 79-05C. We turn our attention to that subject in the next section.

(iii) The Pump Trip Requirement

136. The uncertainties described earlier concerning two-phase natural circulation cooling are directly related to the I&E Bulletin 79-05C pump trip requirement. Prior to that Bulletin, operators were directed to maintain forced circulation cooling when a small break LOCA was suspected. NRC Ex. 4, Appendix A. Thus, there was virtually no need to consider reliance on reflux boiling or feed and bleed cooling under the prior operating mode.

137. The pump trip requirement of 79-05C forces utilities to rely upon natural circulation cooling, including reflux boiling or feed and bleed cooling, in the event significant voiding is present. Based upon our earlier findings [Section V.F.(ii)], we are constrained to state that such reliance on untested modes of core cooling is not to be favored. We are not alone. The

NRC Staff, in NUREG-0565 entitled "Generic Evaluation of Small Break Loss of Coolant Accident Behavior in Babcock & Wilcox Designed 177-FA Operating Plants", states that the reactor coolant pump trip is not "an ideal solution". NRC Ex. 2 at 2-5; accord NRC Ex. 4 at 5-31.

138. The logic for the reactor coolant pump trip also requires us to express concern related to whether Rancho Seco (and, indeed, all other PWR's) are in compliance with 10 C.F.R. §50.46. Section 50.46 requires each PWR to be provided with an emergency core cooling system ("ECCS") that will maintain key core components (the fuel element cladding) below 2200°F during postulated loss of coolant accidents. The Licensee must have an evaluation model "sufficient to provide assurance that the entire spectrum of postulated loss-of-coolant accidents is covered." 10 C.F.R. §50.46. Historically, these analyses have not considered very small breaks, and the smallest break size analyzed has generally been significantly larger than that produced by a failed safety or relief valve. NRC Ex. 2 at 1-1; Tr. 501 (Lewis).

139. The analyses underlying the reactor coolant pump trip requirement demonstrated that previous ECCS models and analyses were not adequate. Thus, the new NRC analyses indicate that for a certain spectrum of small breaks, the ECCS is not capable of meeting Section §50.46 requirements unless the reactor coolant pumps are manually tripped within approximately 3 minutes of LOCS

initiation. Norian Testimony at 3-4; Karrasch and Jones testimony at 63.[24]

140. Although at the time Rancho Seco was licensed it was expected that the ECCS would operate in conjunction with the reactor coolant pumps, B&W did not analyze the performance of the two systems operating together. B&W did not inform its licensees that this analysis had not been performed, nor did it advise them to trip the reactor coolant pumps upon ECCS actuation. Tr. 884 (Karrasch).

141. More recently, in review of B&W's LOCA analyses subsequent to TMI, the NRC Staff concluded that the B&W analysis is satisfactory for the purpose of predicting trends in plant behavior following a small LOCA. But the Staff has several concerns regarding B&W's computer model that it believes should be evaluated before the B&W methods can be considered for NRC approval under the requirements of 10 C.F.R. §50.46. NRC Ex. 2 at 4-10.

142. Two of the Staff's eight identified concerns are that:

> (a) B&W's computer programs may not correctly predict the various modes of natural circulation and interruption of natural circulation; and
>
> (b) experimental verification of small break analysis methods is currently limited, and comparison of the total analysis method with available test data has indicated large uncertainties in the calculations. Id. at 2-1, 2-3.

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24. If the coolant pumps operate throughout a transient, no violation of §50.46 occurs. The danger, however, comes if the pumps are operated for more than 3 minutes after ECCS initiation and thereafter are tripped. See NRC Ex. 2.

143. In addition, the B&W small break LOCA analysis cannot account for the presence of non-condensible gases in the primary system. Id. at 4-5.

144. The Staff believes that B&W should revise, document, and submit its small break LOCA analysis for NRC approval and that plant-specific calculations using the NRC approved model for small breaks should be submitted by all licensees to show compliance with 10 C.F.R. §50.46. Id. at 2-3.

145. B&W witnesses testified for the Licensee that the small break LOCA analyses required by the May 7 Order were "never intended to try to meet" the requirements of 10 C.F.R. §50.46. They added that for this reason B&W has not yet agreed to perform additional analyses or to submit existing analysis for approval under that regulation. Tr. 1035-1039 (Karrasch and Jones).

146. Since the reactor coolant pumps are no longer available once RCS pressure falls to the ESFAS setpoint, Rancho Seco has significantly less defense in depth for these transient and off-normal events than would be the case if the pumps remained available. Webb testimony at 3. As we have found, forced circulation is the most reliable means of core cooling.

147. As noted earlier, the NRC Staff believes that the reactor coolant pump trip is not "an ideal solution" to the small break problem and that licensees should consider other solutions. One solution suggested is that

HPI flow rate be increased. NRC Ex. 2 at 2-5; NRC Ex. 4 at 5-3].

148. Several other witnesses also expressed dissatisfaction with the coolant pump trip requirement. Dr. Lewis testified that he personally disagreed with it and believed the NRC would someday reverse it. Tr. 486-487, 501-502. Licensee's witness Rodriguez suggested that the requirement should allow consideration of subcooling. Because the breaks which underlie the RCP trip requirement would result in saturated conditions in the primary system, Mr. Rodriguez suggested that the trip is unjustified unless subcooling is lost. Tr. 3434-35 (Rodriguez). Considering the benefits of forced circulation, the Board finds this suggestion reasonable, although it is no substitute for also investigating means to increase HPI flow rate.

149. In conclusion, the Board finds the reactor coolant pump trip requirement disturbing in the context of the uncertainties associated with reflux boiling and feed and bleed cooling. We do not question the need for the trip requirement, given existing analyses, nor do we question the theoretical validity of these core cooling methods. Nevertheless, we are concerned bacause the B&W small break analysis predicts a loss of natural circulation which is not required for certain small break LOCA's. NRC Ex. 2 at 4-26. For these events, either reflux boiling or feed and bleed cooling are relied upon for core

cooling. We stress that the analysis demonstrates that either of these methods adequately cools the core. Id. But, as we have found, there are significant uncertainties in the analysis itself as well as the practical understanding of these methods. Accordingly, the Board emphasizes the need for the following:

> (a) A demonstration that the B&W analysis meets the requirements of 10 C.F.R. §50.46;
>
> (b) More detailed analysis and verification of both reflux boiling and feed and bleed cooling, as well as upgraded operator training on these methods; and
>
> (c) Investigation of methods to limit or remove the reactor coolant pump trip requirement.

150. In answer to Board Question CEC 1-4, the Board finds that the implications for safety of operating Rancho Seco until the exact behavior of the system in a small break LOCA is well understood are:

> (a) Rancho Seco has less defense in depth for feedwater transients than was envisioned when it was licensed due to the coolant pump trip requirement;
>
> (b) Rancho Seco must rely upon natural rather than forced circulation more often than was envisioned when it was licensed, even though forced circulation is preferable; and
>
> (c) There is a greater burden placed upon the operators at Rancho Seco due to the RCP trip requirement and the increased reliance on natural circulation.

151. In answer to Board Question CEC 1-2, the Board finds that there is adequate understanding of natural convection in the Rancho Seco system, but there is not adequate understanding of reflux boiling and feed and bleed cooling.

152. In answer to Board Questions CEC 1-10 and H-C 24, the Board finds that voiding can occur in the Rancho Seco system during a LOCA or an overcooling transient. Where these conditions occur, available methods of core cooling may place undesirable demands on operators.

G. Operator and Management Competence

CEC 3-1:

Whether personnel adequately understand the mechanics of the facility, basic reactor physics, and other fundamental aspects of its operation?

CEC 3-2:

Whether personnel are properly apprised of new information pertinent to the facility's safe operation and ability to respond to transients, particularly information on operating experience of other reactors?

CEC 3-3:

Whether NRC and SMUD adequately ensure that emergency instructions are understood by and are available to plant personnel in a manner that allows quick and effective implementation during an emergency?

Board Question H-C 32:

What procedures have been used to test and evaluate the competence of Rancho Seco's operating personnel and management?

Board Question H-C 34:

What actions and/or programs are employed at Rancho Seco to assure that operating personnel, both licensed and unlicensed, adequately respond to feedwater transients?

Board Question FOE III(d):

The NRC orders in issue do not reasonably assure adequate safety because no procedures have been taken to assure facility management competence.

Board Question FOE III(e):

The NRC orders in issue do not reasonably assure adequate safety because no procedures exist or have been taken for the determination of the adequacy of operator competence.

153. CEC Issues 3-1, 3-2 and 3-3, Board Questions H-C 32, H-C 34, and FOE III(d), and III(e) raise various issues concerning the competence of Licensee's operators and management to provide reasonable assurance that Rancho Seco will respond safely to feedwater transients. These various issues address generally the competence of licensed operators, management, and unlicensed operations personnel. We therefore address these matters in that order before turning to the additional specific issues regarding emergency procedures and feedback on operating experience. It bears repeating at the start, however, that we regard these issues as extremely important in view of the basic design sensitivities of the Rancho Seco facility. We return to the statement made by the NRC Staff in April 1979: If B&W sensitivities are not reduced by design changes (and they have not been), licensees must "substantially upgrade plant operator education, training, and experience." CEC Ex. 26 at 1-8. Our findings in this section address whether such substantial upgrading has been demonstrated on this record.

(i) <u>Operator Competence</u>

154. The May Order required SMUD, both in the short and long-term, to undertake additional training of its licensed operators in light of the experience gained from the TMI accident. This is the only additional operator training experience instituted at Rancho Seco since TMI. Rodriguez Testimony at 15-18; Wilson Testimony at 4-7. Hence, we begin

our review of operator competence by considering this special training.

155. SMUD has placed considerable emphasis on this special, post-TMI training to show that operators have learned the lessons of TMI. See Licensee's Findings 125, 170-172. SMUD has stated, for example, that this training "gave a great deal of attention" to small break LOCA's. Id. No. 125. SMUD has listed no less than 13 important subject areas that were covered in the post-TMI training. Id. No. 171.[24] The implication of these proposed findings is that this special training was extensive. That is incorrect, however. All the special training given to Rancho Seco operators after the TMI accident from March 28, 1979 through facility restart after the June 21, 1979 NRC Order totalled only 27 hours, including testing and informal discussion. Rodriguez testimony at III-1. This includes the special TMI simulator training, which consisted of one day at the B&W simulator for the purpose of watching a simulation of the TMI accident without operator intervention. Operators were then permitted to view the simulation a second time and take action to arrest the

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24. The additional training covered time on the B&W simulator, the sequences and events and causes of TMI, post-TMI procedure changes, NRC I & E Bulletins, plant modification after TMI, small break LOCA's, void formation theory, saturated and subcooling operations curves, initiation and recognition of natural circulation, safety features actuation system operation, AFW operation, control of reactor trip relay, clarification of technical specifications, and requirements for notifying the NRC. Id.

accident. Tr. 3091-92 (Rodriguez).

156. The 27 hours also included four hours of remedial training by the General Physics Corporation. Rodriguez Testimony at III-1. This training was required by the Staff after an audit of seven operators revealed that three or four did not adequately understand natural circulation and small break LOCA phenomena, notwithstanding their licensing training, requalification training, and special post-TMI training. Tr. 3791-92 (Wilson). Some of the operators were unable to identify what indications verified that natural circulation flow was adequate. Furthermore, some believed that a very high temperature difference between the hot and cold legs of the primary system indicated good natural circulation flow, when in fact it indicates the opposite. Some operators also were unable to explain why the pressurizer level at TMI was rising while RCS pressure was falling. Finally, some operators incorrectly predicted that the primary system would superheat if a saturated system was depressurized. Tr. 3799-3800 (Wilson). Although the operators were able to answer these questions correctly after the specific deficiencies were communicated to SMUD management and the additional training was given, the Board nevertheless is disturbed by the inability of a majority of the audited operators to answer such questions originally. Tr. 3803 (Wilson). We view these questions as going to basic concepts, central to the TMI accident, which effective

training (and certainly effective special post-TMI training) should have made clear. See Tr. 3807 (Wilson). Moreover, the Board notes that prior to the Staff audit, SMUD passed these operators on an exam which included a request that they "briefly discuss how the operator can ensure that natural circulation is occurring." Tr. 3801 (Wilson). According to Staff witness Wilson, a passing answer to this question need only have included that the temperature difference between the hot and cold legs should be proper, without identifying the proper difference. Tr. 3801-02 (Wilson). The Board finds this exam superficial, for certainly the knowledge of what instruments must be read is useless unless one also knows what to look for.

157. On the whole, the Board does not find 27 hours of training on a wide variety of complex subjects, given once and including several hours of testing and informal discussion, to be a substantial addition to the existing training program. This is especially true given the superficial nature of this training, as evidenced by the operators' initial performance on the NRC audit.

158. SMUD's regular operator training program has not been changed since TMI, save for the inclusion of material related to that accident. Tr. 3088 (Rodriguez). This program consists of the operator's preparation for the NRC's licensing exam and the Licensee's requalification program for its licensed operators. The preparation

for the licensing exam is of two types: "hot" and "cold". The "hot" licensing preparation has been used to prepare operator canidates since Rancho Seco began operating in 1974. The "cold" licensing program was used prior to that time to prepare the original operating crews at Rancho Seco. Overall, the "cold" licensing program was considerably more extensive than the existing "hot" program, particularly in the amount of simulator training given. The "cold" program included a 10 week simulator course, while the existing "hot" program includes only three weeks at the simulator. Rodriguez Testimony at 9 and 13.[25]

159. There are 24 licensed operating personnel at Rancho Seco, but only 16 stand regular control room watches. The others are in various supervisory and management positions. Of the 16 operators who stand shifts, 11 underwent the existing "hot" license training, two underwent the entire "cold" license training, and 3 underwent most but not all of the "cold" license training. Tr. 3047-49 (Rodriguez).

160. Pursuant to NRC regulations, 10 C.F.R. §50.54 (1-1), SMUD has a requalification program to provide post-license training and testing to its licensed operators. This includes 12 to 15 lectures per year, a few of which concern emergency procedures, and an annual one-week simulator course. CEC Ex. 36 at 115-16. The testing

25. Of the time spent at the simulator, about half consists of actual simulator experience and the remainder is devoted to classroom instruction. Id.

includes an oral exam[26] and a written exam administered and scored by the Licensee. Wilson Testimony at 4. The written requalification exam has twice been audited by the NRC Staff, most recently in 1976. Tr. 3823-24 (Wilson).

161. SMUD's overall operator training program is similar in scope, amount, and type of training to general industry practice. Tr. 3811 (Wilson). It does not substantially differ in these respects from the training given to the TMI operators. Bridenbaugh and Minor Testimony at 11; Tr. 3811-12 (Wilson).

162. As Mr. Bridenbaugh testified, however, the quantity of training is not the total picture. Tr. 3610-3611. The quality of the Licensee's training must also be considered. However, apart from speculation that it _could_ theoretically be better, the Licensee has presented no persuasive evidence to support a finding that the Rancho Seco program is qualitatively better than that of other utilities. Inasmuch as Licensee bears the burden of proof, and since the training given at Rancho Seco does not differ from industry practice in other respects, the Board must look to other evidence to determine if the quality of Licensee's training is superior.

163. The simulator is the most effective tool available for the training of operators. Tr. 3859 (Wilson).

26. This oral exam has not always been given as scheduled. _See_ Section V.G.ii, _infra_.

Rancho Seco operators receive their simulator experi nce on the B&W simulator located in Lynchburg, Virginia. This simulator is similar to the Rancho Seco control room in terms of the layout of controls and indication. Rodriguez Testimony at 9. CEC witness Lewis termed this a "modest advantage". Lewis Testimony at 13. Other witnesses in the proceeding also recognized that the congruity of the simulator to the actual Rancho Seco control room enhanced the quality of the simulator training. E.g. Tr. 3564 (Bridenbaugh). In NUREG-0667, the B&W Reactor Transient Response Task Force recognized that the conformity of the simulator to the actual control room is "a distinct advantage in the training of B&W operators". NRC Ex. 4 at 5-69.

164. There are, however, some differences between the Rancho Seco control room and the B&W simulator. For example, the auxiliary feedwater controls are not located in the same positions. The switch layout for adding boron to the coolant system also differs, as does the switch layout for the steam line break failure logic system. Additionally, some of the balance of plant systems operate differently at Rancho Seco than their counterparts at the simulator. An important example is that the B&W simulator does not represent the dual drive AFW system present at Rancho Seco. Tr. 3094-98 (Rodriguez)

165. NUREG-0667 also points out, however, that the

B&W simulator was one of the first of its kind and there is a distinct lack of fidelity in some areas compared to other simulators. NRC Ex. 4 at 5-69; see Tr. 3855 (Wilson). Modifications were necessary in order to allow the simulator to reproduce both the TMI and the Crystal River 3 events. Further, two-phase conditions in portions of the reactor coolant system other than the pressurizer and multiple failures were not part of the computation model. NRC Ex. 4 at 5-69 & 5-70; see Tr. 3094 (Rodriguez). The simulator cannot simulate the Rancho Seco light bulb incident. Tr. 3102 (Rodriguez).

166. Although much of the time that operators spend in training on the simulator involves responding to abnormal occurrences, [CEC Ex. 37 at 74-75], this does not mean that operators necessarily are familiar with a variety of degraded conditions. Mr. Rodriguez testified that the pattern of simulator training is to begin with the reactor operating normally and then present the operator with a failure. If the operator responds correctly, the simulator will display recovery from the transient. Because the training is conducted in this fashion, it appears that operators receive little or no simulator experience with severely degraded conditions like those at TMI. For example, Mr. Rodriguez testified that he was not sure whether the B&W simulator

could demonstrate the inability to condense non-condensible gases because:

> typically in operating that, we don't let it go that far so I can't recall, you know, seeing - just standing there not doing anything other than that one instance a year ago when we sat there to watch what happened at Three Mile Island. Tr. 3102.

Similarly, Mr. Rodriguez testified that operations have probably not seen simulation of the problem underlying the reactor coolant pump trip because the purpose of the training is to teach them to avoid the problem. Id. at 3105. Staff witness Wilson confirmed that this is indeed the pattern of simulator training, and added that the Staff was considering training shift technical advisors by presenting more severely degraded conditions. Tr. 3835-36.

167. Although Mr. Rodriguez testified that the simulator course provides an operator with "the opportunity to exercise his diagnostic skills and training in mitigating the consequences of those multiple failure accidents" [Rodriguez Testimony at 13-14], the testimony of the Rancho Seco operators suggests that multiple failure accidents are rarely presented in the simulator course. The senior operator testified that during his most recent week of simulator training, he was probably given only one muliple failure transient.

He could not recall whether the multiple failure transient he experienced differed substantially from the TMI incident. CEC Ex. 36 at 90-91. The operator similarly responded that he had been given only one multiple failure event. CEC Ex. 38 at 54. See also Tr. 3835 (Wilson).

168. Taken together, the evidence before the Board does not support a finding that the simulator training of Rancho Seco operators is of superior quality. While the general conformity of the simulator to Rancho Seco is advantageous, this advantage is offset by the age and fidelity of the simulator and the infrequent simulation of degraded conditions and multiple failures.

169. The overall quality of the Licensee's training program is best measured by the knowledge of its operators. Indeed, this is the issue before the Board, not the training itself. We have previously found that the deficiencies in operator understanding identified by the Staff in their audit following the special post-TMI training suggest poor quality training. Findings 156-157. To further consider operators' competence, we look to the depositions of three operators, one from each of the three classifications present on a shift, taken by the California Energy Commission.[27] As one would expect, the knowledge of the operators varied with their seniority.

27. The Licensee made seven of the 16 operators available for this examination, from which the Energy Commission selected at random a shift supervisor, a senior operator, and an operator.

In our judgement, the shift supervisor displayed a thorough understanding of the plant and its operating procedures [CEC Ex. 37, passim], the senior operator a somewhat less complete understanding [CEC Ex. 36, passim], and the operator an inadequate understanding. CEC Ex. 38, passim.

170. As described in Finding 133, the senior operator displayed confusion regarding alternative methods of core cooling, and the operator gave incorrect responses regarding feed and bleed cooling. The operator also was apparently unaware of the basic concept that hot, pressurized water usually cools as it depressurizes. CEC Ex. 38 at 18-19. As one Board member pointed out, this basic concept is the way a household refrigerator functions. Tr. 3259. More disturbing, despite the training given the operators (including at least 240 hours of physics; Rodriquez Testimony at II-1), Mr. Rodriguez testified that he did not expect his operators to be aware of this phenomena, let alone the magnitude of the temperature drop. Tr. 3238-39. The Staff's expert on operator training disagreed, stating he would expect an operator to be aware of this principle of basic reactor physics. Tr. 3808 (Wilson).

171. CEC Issue 3-1 also addresses the operators' understanding of the fundamental aspects of the operation of the Rancho Seco facility. One such aspect which received considerable attention during the course of

of this hearing is the relatively brief amount of time in which the OTSG will boil dry following a loss of feedwater. See Section V.A. In this context, it is notewor.hy that when asked if he knew how quickly the OTSG could boil dry in such an event, the senior operator stated he did not know. CEC Ex. 36 at 16. We consider this to be astonishing in view of the emphasis which has been placed on B&W sensitivities since TMI.

172. Mr. Rodriguez described the purpose of the academic phase of the "hot license" program as "assuring that the candidate has basic skills in mathematics, an understanding of classical physics, atomic physics, and physics directly related to the reactor core." Rodriquez Testimony at 8. With regard to mathematics, Energy Commission counsel asked the operators to describe the mathematics they must perform. In responding, the senior operator indicated he could not recall what his mathematics of dynamic systems course was about. CEC Ex. 36 at 99. Similarly, the reactor operator, responding to a question regarding his trigonometry class, replied: "Trig, what in the heck is trig, anyway?" CEC Ex. 38 at 44. The senior operator also could not recall the substance of his hot license training on brittle fracture of the reactor vessel. CEC Ex. 36 at 89.

173. The incorrect statements and the inability to recall the subjects of various training classes are not

necessarily indicative of operator incompetence, for no individual can be expected to always recall training, especially during a deposition. On the other hand, some of the concepts mentioned above are relatively basic and simple, such as the OTSG boil dry time or the relationship of pressure and temperature of liquids. The Board also notes that these deficiencies were revealed in relatively limited substantive questioning and not a comprehensive inquiry into the operators' knowledge of the concepts they should understand. A considerable portion of each deposition was devoted to matters such as description of the facility, operator experiences with various transients, equipment availability, descriptions of the SMUD organization, and other matters not central to the operators' training. Thus, while the depositions do not lead us to conclude that operators at Rancho Seco are less competent than at other facilities, the depositions likewise do not presuade us that their training is superior or that they are more competent than operators at other facilities.

174. The issue of operator understanding of emergency procedures raised by CEC 1-1 is subsumed in the overall issue of operator training and competence. On this particular aspect of operator training, the evidence suggests that while the requalification training includes some lectures on these procedures [Rodriquez testimony at 11], they are not well informed of the

basis for changes in these procedures. Thus, emergency procedure changes are routinely communicated to operators through the standing order program, discussed <u>infra</u>, which does not in our view, ensure that operators understand the reasons for or substance of emergency procedure changes. Tr. 3459 (Rodriguez). This is of special concern to the Board because these procedures have undergone substantial revision in recent months. Tr. 3847 (Wilson); Wilson Testimony at 15.

175. NRC Staff witness Wilson testified that the NRC determines whether licensed personnel adequately understand emergency procedures through the licensing examination process. Wilson Operator Training Testimony at 13. He stated that on the basis of the examinations conducted to date at Rancho Seco, the NRC is satisfied that licensed personnel understand emergency procedures. <u>Id</u>. However, this examination qualifies operator canidates; it does not assure a continuing understanding or an understanding of subsequent changes. Mr. Wilson testified that licensed personnel demonstrate a continuing understanding of emergency procedures through an oral examination that is part of Licensee's requalification program. <u>Id</u>. at 14. He cited section 3.2.1 of the requalification program which requires an annual oral examination which includes a discussion and simulation of required actions during abnormal or emergency conditions. <u>Id</u>. The Board notes, however, that

the NRC's Performance Appraisal Branch has found that several licensed operators have not been given this oral examination in a timely manner. See Finding 187.

176. The depositions of the Rancho Seco operators reveal the way that changes to emergency procedures are made at Rancho Seco. The senior operator stated that operators are informed of procedure changes by a copy of the procedure being brought into the control room. CEC Ex. 36 at 94. The procedure is given to an operator (though not any particular operator) who places it in the appropriate manual. He then signs a cover sheet in the binder indicating that a procedure change has been made. Id. Other operators are informed of procedure changes through the special order program which is a written order or memo kept in a special order book in the control room. Id. at 95. The senior operator testified that operators are not obligated to read the special orders before each shift [id.], but operators are required to review such new procedures and document completion of that review. Rodriguez Testimony at 32. Usually a copy of the special order is given to each licensed operator, but the senior operator stated that he had never been tested either in writing or orally on his knowledge of the contents of such an order. CEC Ex. 36 at 95-96. The operator described the training he receives when the operating procedures are changed:

> The actual memo that comes out the SO [Standing Order] lists a bunch of things that they want to keep us aware of. Then it lists in there which procedures have been revised. So then I read the SO's and I can go back and look in the procedures to see what the changes are. A few times where there was a significant change that they wanted us to know right away, there is also in the shift supervisors office they have a little blackboard and they'll make sure and note it that these changes have been made or I have had operation supervisor come in and, you know, give us a brief rundown on why the change was made. Because it's kind of nice to know sometime why the changes were made. CEC Ex. 38 at 55-56 (emphasis supplied).

177. The same operator also suggested that operators have difficulty coping with the numerous procedural changes that have been implemented since the TMI accident:

> A lot of this stuff, man, you just kind of, read it and there's so many damn changes going on you don't want to memorize all these things. When you get setting down hard, you want to remember those things. CEC Ex. 38 at 66-67.

178. The foregoing operator comment raises the question whether operators have sufficient time for necessary training. In our view, a training and requalification program should be administered so that a person's training is an integral part of regular duties, not an added burden. At Rancho Seco, however, the evidence suggests that training is limited by operators' other duties. Thus, the Board notes that operators must come in early periodically for requalification training.

Tr. 3081 (Rodriguez). Further, Mr. Rodriguez expressed concern about overloading operators with too much material. Id. at 3305.[28] Finally, there was testimony that when operators leave for simulator training, this puts a considerable strain on remaining personnel. Tr. 3232 (Rodriguez). We do not consider the foregoing evidence to be conclusive, particularly in the post-TMI environment when many new requirements necessarily put new burdens on operators. However, we feel this is a matter that deserves close attention by SMUD management.

179. In summary of the evidence on operator competence, the Board concludes that the record does not support a finding that Rancho Seco operators are substantially more competent than operators of other facilities or that their level of competence has been substantially improved since TMI. Indeed, the evidence before us raises concerns regarding operators' understanding of basic reactor physics and certain fundamental aspects of the operation of Rancho Seco. Both in this proceeding and in the NRC audit in June 1979, Rancho Seco operators have exhibited misunderstanding of important concepts. While we do not conclude that these operators are incompetent, the Board finds that Rancho Seco operator training needs improvement, especially in light of the greater demands placed on these operators

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28. We are certainly sympathetic with this concern but many relevant materials apparently are not routinely made available to operators. See Finding 218.

due to the B&W design and reactor coolant pump trip requirement. Moreover, with respect to CEC 1-1, the Board finds that SMUD should improve its training regarding changes of emergency procedures and institute management controls to ensure that operators understand new procedures and the bases for changes.

180. The Board does not believe it appropriate to determine for SMUD how to improve its training program. However, the Board has several observations in this regard, each of which it expects SMUD to consider seriously. First, a serious apparent shortcoming of the SMUD program is operators' lack of time to be trained. SMUD should analyse carefully whether there are means to better manage operators' time to allow greater training, including the possibility of employing additional operators so that each operator has sufficient time for training and review of relevant data. Second, and related to both operators' training and management competence, we believe SMUD seriously should consider hiring outside consulting expertise to analyse carefully and suggest improvements in SMUD's operator training program. This would be similar from a personnel point of view to the control room human factors study which SMUD has recently begun. Finding 224. Third, the record suggests that SMUD should consider substantially more simulator training, including more simulation of degraded conditions and multiple failures. Another idea that merits consideration was the suggestion

of Dr. Lewis that SMUD develop procedures and simulator training for the most likely accident sequences identified in the Reactor Safety Study. Tr. 525 (Lewis). A related improvement which was discussed in the hearings was the construction of a simulator at Rancho Seco. The Licensee suggested such a facility would cost $15 to $20 million to construct.[29] Staff witness Wilson gave lower estimates, however. He suggested the capital cost of a simulator would be $8 million and that the total cost, including the building, trainers, etc. would be $20 million. Tr. 3857-58 (Wilson). The record also shows, however, that SMUD spends some $300,000 annually for the current one week of training at the B&W simulator, not including overtime costs to replace absent operators. Tr. 3233-36 (Rodriguez). Over the remaining 34 years of the operation of Rancho Seco, this will total some $10 million and likely more if SMUD should decide that additional training were warranted. Thus, the costs of constructing and operating a simulator at Rancho Seco will be offset somewhat. The Board believes, however, that the benefits of a simulator are substantial and thus this proposal deserves careful consideration even if costs are high. A new simulator would precisely mirror the Rancho Seco

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29. This assumed $5 million inflation as a result of the competition of other utilities with an interest in purchasing simulators. It also assumed the cost of a building to house the simulator, although Mr. Rodriguez acknowledged that SMUD plans to construct a new building in any event. Tr. 3233-36 (Rodriguez).

control room and have a greater capacity that the existing one. Operators would have ready access to the training device that clearly seems most beneficial. See Tr. 3233 (Rodriguez). And management and supervisory personnel would have ready ability to test new procedures before they were put into effect.

(ii) Management Competence

181. The competence of SMUD's management is, of course, directly related to the competence of its operators, since management is responsible for the training and evaluation of operators. For this reason, many of our findings on operator training also bear upon the issue of management competence.

182. Licensee witness Rodriguez testified that Rancho Seco management is sufficiently competent to provide reasonable assurance that the facility will respond safely to feedwater transients. Rodriguez Testimony at 21. The bases for this conclusion are that: 1) four key management personnel maintain senior reactor operator licenses and have participated in requalification and post-TMI training; 2) two of these four "have been active in industrial organizations dealing with plant activities at facilities across the country"; and 3) "management and supervisory personnel have begun participation in a command and control training program being

presented by a consultant to the District." Id. at 19-20.

183. The NRC Staff presented several witnesses on this issue. Staff witness Allenspach described the NRC's criteria and procedures for evaluating management competence, as well as proposed changes in these criteria. Allenspach Testimony, passim, following Tr. 3920. In general, the NRC criteria pertain to the structure of the Licensee's organization and the qualifications of its personnel. Id. Mr. Allenspach stated that no significant deficiencies in SMUD's capability to operate the facility have been noted under existing criteria, but that the criteria are being upgraded as a result of the TMI accident. Id. at 6. He concluded by stating that new procedures which will be required of SMUD by the upgraded criteria "will provide the management and technical capability needed to assure adequate safety of the Rancho Seco facility". Id. at 9.

184. Staff witnesses Allen D. Johnson, Gerald B. Zwetzig, and Harvey L. Canter, inspectors for NRC's Inspection and Enforcement Office, also addressed the management competence issue. Mr. Johnson concluded that "the SMUD organization and personnel are competent to safely operate" Rancho Seco, and Mr. Zwetzig and Mr. Cantor testified that they had no reason to disagree. Johnson Testimony at 11, following Tr. 3920; Zwetzig Testimony at 6, following Tr. 3920; Cantor Testimony at

8, following Tr. 3920. These witnesses relied in large measure upon the number of items of non-compliance and reportable occurances at Rancho Seco since it commenced commercial operation, which are discussed in Findings 191-195.

185. The staff also submitted testimony from two members of its Performance Appraisal Branch (PAB), Darrell G. Hinckley and James E. Gagliardo. Hinckley and Gagliardo Testimony following Tr. 4232. A PAB team completed an inspection of the Rancho Seco management control systems on May 8, 1980. As described in this testimony, the PAB inspection was:

> to determine _how_ the Licensee manages licensed activities to assure continued compliance with regulatory requirements and guidance. This differs from the regional based inspections which are oriented toward verification that the Licensee is in compliance with the regulatory requirements and guidance. Hinckley and Gagliardo Testimony at 2.

186. The PAB witnesses testified that their inspection focused on eleven functional areas of management and that they identified weaknesses in seven of these areas at Rancho Seco. The seven areas included the following:

(a) _Fire Protection_ - lack of drills and inadequate procedures;

(b) Training - failure to implement some procedures for training licensed and unlicensed personnel, poor training records;

(c) Corrective Action System - failure to routinely enter quality assurance audit findings "into a corrective action system for resolution", certain non-supervising personnel were unaware of the "Reportable Occurance Report", and some items in the Nonconformance Report had remained open for as much as five years;

(d) Design Change and Modifications - failure to give proper safety evaluation reviews for changes to Class I systems;

(e) Maintenance - insufficient maintenance procedures, lack of an adequate system to ensure that technical manuals are up to date;

(f) Quality Assurance Audits - failure to adequately audit operations personnel, the preventive maintenance system, surveilance activities, or major maintenance activities, poor quality audits ("the adequacy of several audits in their scope and depth and the procedures by which they were conducted raised questions as to the ability of the audit program to serve as an effective, independent review function"); and

(g) Committee Activities - failure of both the offsite and onsite review committees to perform required audits and reviews. Id. at 2-5.

187. On cross-examination, the PAB witnesses gave more specific examples of their concerns. They stated that operators were not properly made aware of modifications to their own facility. Tr. 4240. Further, in some cases non-licensed personnel were not being given retraining

in maintenance or instruction how to report non-conformances, work requests, and similar things as specified in their training procedures. Id. at 4250-51. They also testified that several licensed operators were not given their oral requalification exam within the time allotted in the requalification program. Id. at 4255. The PAB team allowed a 25 per cent margin of error in meeting such deadlines, and for an annual requirement would allow a couple of months as permissible margin. Id. at 4256-57. This testimony therefore suggests that for several operators, the oral requalification exam requirement was missed for a significant period of time. Another important observation made by the PAB witnesses concerned audits of unlicensed personnel training. Remarkably, for two consecutive years beginning in 1978, this audit was not carried out because the unlicensed personnel training procedure had not been implemented. No corrective action was instituted to determine why the procedure had not been implemented. Id. at 4262.

188. On the whole, the PAB witnesses felt that Rancho Seco management controls were poor in comparison to the other facilities they had inspected.[30] Mr. Gagliardo described his overall opinion of the Rancho

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30. At the time of this testimony, Rancho Seco was the seventh licensee to receive the full management appraisal inspection. Id. at 4241 (Gagliardo).

Seco management control system this way:

> My opinion from what I have seen, and I have been involved in most of these inspections myself, and those that I was not involved in, I was the branch chief, I would classify Rancho Seco as in one of the lower groupings. We do not try and rank the licensees, and we do not intend to do that unless forced to do so. What we look at is, we classify licensees as those who have good management control systems average or a poor system, and I would say that Rancho Seco on the preliminary look puts them in that lower category. Id. at 4249.

189. In weighing the evidence before us on this issue, the Board has distinguished testimony which only described the procedures and organization which are appliable to Rancho Seco from testimony which described the way in which these procedures are in fact implemented. We recognize that a Licensee's plans for ensuring safe operation of its facility must be carefully examined, especially in initial licensing proceedings where there is no implementation record to examine. But in this unique proceeding, where we must determine whether an operating facility will safely respond to future feed-water transients, we have the benefit of examining the implementation of procedures and not just the procedures themselves. Obviously, procedures that exist only on paper are of no practical use. For example, the required annual operator requalification oral exam cannot be

considered a particularly useful training exercise because it is not implemented properly.

190. For this reason, the Board has weighed heavily the testimony of the PAB witnesses. It is in our experience unique to have the testmony of management control experts who have just completed an in depth, on site review of this nature. Their testimony is based upon a thorough, expert review of the Licensee's management as it is actually functioning at Rancho Seco today.[31] For this reason, while there is contradictory evidence on this issue, the Board is persuaded to accept the PAB witnesses' conclusion that SMUD's management controls are poor.

191. Although the Board has relied heavily upon the testimony of the PAB witnesses, other evidence before the Board confirms their conclusions. As noted in Finding No. 184, Staff witnesses Zwetzig, Johnson and Cantor relied primarily upon Licensee's record of non-compliance items and reportable occurrences in concluding that Rancho Seco is operated competently. These items, while clearly relevant, may not be a clear indication of management or operator competence. Such factors as the age of the facility and the interpretation of the criteria for

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31. The PAB have devoted approximately 500 hours to its review of Rancho Seco. Tr. 4234 (Hinckley).

reporting such items play a significant role in determining the number of these events at any given facility. Tr. 3480-81 (Rodriguez); Tr. 4071-80 (Cantor, et. al.). Thus, the Board considers these occurrences and reports only one of many factors to be considered in evaluating the competence of SMUD's operation.

192. Moreover, the Licensee's record of non-compliance items and reportable occurrences presents a mixed picture. In a report prepared by the NRC for the Three Mile Island Unit 1 restart hearing, the Staff tabulated and statistically evaluated the numbers of licensee event reports for each of 70 operating nuclear power plants for the period January 1, 1969 to December 31, 1979. Tr. 3444-45 (Rodriguez). Rancho Seco ranked 16th out of the 70 plants surveyed, where the first unit would have the lowest number of reports and the last unit the greatest. As a Board member pointed out and Mr. Rodriguez agreed, however, there is some doubt as to the statistical significance of the fact that Rancho Seco ranks 16th. Tr. 3489. Inasmuch as the technical specifications impact the number of reports at a particular unit, it is noteworthy that Rancho Seco had the best record among the operating B&W facilities, which have similar technical specifications. Tr. 3445 (Rodriguez).

193. However, older units generally generate fewer such reports than newer ones, and this effect is not

reflected in the rankings in the NRC Staff's report. Tr. 3480-82 (Rodriguez). Furthermore, when only the reportable occurrences caused by personnel or operations-related errors are considered, Rancho Seco ranks 41st out of the 70 plants despite its relatively long period of operation. Id. at 3462-63. Of the 70 plants surveyed, Rancho Seco had a higher proportion of its total reportable occurrences causes by personnel error than any other facility. Tr. 3790-94 (Wilson). Thus, this survey suggests that while Rancho Seco does not have an extraordinary number of overall reportable occurrences, it does have a disproportionate number caused by personnnel errors.

194. In examining these items and occurrences, it is important also to consider the seriousness of the violations. On this point, the Board notes that earlier this year SMUD was fined the maximum possible civil penalty as a result of three clearly serious reportable occurrences that compromised the performance of the high pressure injection system. Tr. 3141-49 (Rodriguez). It should be pointed out that each of these violations was discovered by SMUD and promptly reported to the NRC. Tr. 3146 (Rodriguez). However, one of these instances remained undiscovered for almost a month. Id. at 3147. Many of the other LER's also involved important safety systems, particularly the emergency diesel generators. CEC Ex. 40, passim; Tr. 4001-02 (Johnson). Another consideration in evaluating reportable occurrences is

whether they seem to be increasing or decreasing. At Rancho Seco, the frequency of these reports recently has increased rather dramatically, beginning around November of 1979. Tr. 4070 (Cantor). Staff witness Mr. Cantor testified that this increase appears to be "relatively significant". Id. at 4071. The record indicates that from October 1, 1979 until May, 1980, SMUD filed 37 licensee event reports. Id. at 4063-70. Thus, the frequency of reportable occurrences at Rancho Seco has tripled in recent months. The Staff witnesses suggested that the cause of the increase may be changing interpretations of existing regulations and an increased sensitivity to violations on the part of NRC inspectors. Id. at 4071-4080.

195. The Board accepts that an increased frequency and seriousness of reported occurrences may to some extent be caused by more stringent inspection and enforcement practices. But it cannot rely upon such explanations without further support to account for a three-fold increase in these events. Inasmuch as these occurrences are apparently increasing in number and seriousness, together with the high proportion of these occurrences that have been personnel related, the Board finds that there have not been substantial improvements in the operation of the facility since TMI. Moreover, the Board is not persuaded by the remaining evidence that Rancho Seco is operating substantially better than other

nuclear power plants.

196. Additional evidence regarding the competence of SMUD management can be found in its response to a human factors engineering study of the Rancho Seco control room conducted by the Electric Power Research Institute (EPRI). Licensee witness Rodriguez testified that to his knowledge this was the only human factors study of the Rancho Seco control room. Tr. 2965-66. This study examined a number of nuclear power control rooms, including Rancho Seco's. Id. Mr. Rodriguez was shown a copy of a report of this study (CEC Ex. 33) and although he testified that he had probably seen it, he stated that he was not familiar with its contents. Tr. 2968. Although the study identified a number of concerns regarding the Rancho Seco control room (Tr. 2971 et seq.), Mr. Rodriguez stated that he was unaware of any formal response or analysis by SMUD of the EPRI report. He also stated that he was unaware of anyone being assigned within the SMUD operation to review this study. Id. at 3031. The Board considers this lack of interest by SMUD in the only professional, expert review of their control room to be further evidence of poor management.

197. During a discussion of reportable occurrences at Rancho Seco, Mr. Rodriguez testified that he would not expect all of his employees to notify SMUD management when they noticed an irregularity in the installation of control room indication. Tr. 3221-22 (Rodriguez) Mr.

Rodriguez agreed that such irregularities should be routinely reported. Id. at 3224. These statements suggest that the conclusions of the PAB team regarding SMUD's management controls are correct.

198. Finally, the Board notes that Mr. Rodriguez' management responsibilities include reviewing the requalification exam scores of the Rancho Secc operators. Yet, in this hearing he had no idea how operators were performing on this exam. Tr. 3084-86 (Rodriguez).

199. Overall, the Board finds that SMUD's management controls are poor in comparison to other utilities. The Performance Appraisal Branch will recommend in its final report improvements and investigations that should be made regarding SMUD's management controls. Gagliardo and Hinckley Testimony at 5. Licensee should promptly carry out those recommendations.

(iii) Unlicensed Operator Training

200. Board Question H-C 34 raises the issue of the ability of unlicensed operators to respond to feedwater transients. The Rancho Seco technical specifications require that unlicensed persons be present on shift to assist the licensed operators. NRC Staff Testimony of Philip J. Morrill on Training of Unlicensed Plant Operators, at 3, "Morrill Testimony", following Tr. 4141. These unlicensed personnel assist the licensed operators by starting and stopping motorized equipment, opening and shutting valves, conducting periodic maintenance on or checking of equipment, and maintaining plant records. Id. In response to a feedwater transient, unlicensed personnel may be called upon to take actions that are necessary to ensure the safety of the facility. Tr. 3111 (Rodriguez). These actions may include operation of the integrated control system auxiliary feedwater valves, changing the valve lineup on those valves, changing the valve lineup on the makeup pump to assure that it has emergency power, and changing breaker positions to make sure that power is supplied to the "swing" high pressure injection pump in the case of a failed diesel generator. Tr. 3111, 3113-14 (Rodriguez).

201. There are three classifications of unlicensed operators at Rancho Seco. The least experienced personnel are power plant helpers. CEC Ex. 38 at 3; Tr. 3109 (Rodriguez). The next most experienced are equipment

attendants. The most experienced unlicensed personnel are auxiliary operators. Id. A power plant helper may become an equipment attendant after one year, and an auxiliary operator after two years. CEC Ex. 38 at 3-4.

202. The power plant helper is trained on the job and receives little training prior to being assigned to a crew, with the exception of some training in health physics, the emergency plan, and security. Tr. 3116 (Rodriguez); CEC Exh. 36 at 112-13. Sometime after a power plant helper is assigned to a crew, he is given a three to four week classroom systems training course, and then reassigned to the crew. Tr. 3116 (Rodriguez). Unlicensed personnel may also participate in many of the lectures that are given to the licensed operating personnel.[32] Id. On the whole, however, it is fair to say that unlicensed personnel are primarily trained on the job and that these personnel receive their primary instruction the first time they are given a task. Tr. 3116 (Rodriguez); CEC Ex. 36 at 113-114. Unlicensed personnel are usually instructed by other unlicensed personnel. Tr. 3117 (Rodriguez); CEC Ex. 36 at 110-114.

203. Unlicensed operators are given a set of power station manuals to familiarize them with the operation of various plant systems. Licensee witness Rodriguez

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32. However, emergency procedures are not a major portion of the NRC examination or the lectures given in requalification training. CEC Ex. 36 at 115.

testified that unlicensed operators take oral and written quizzes at various times on their familiarization with these manuals. Tr. 3117 (Rodriguez). Staff witness Morrill, however, testified that he understood that the written tests were self-administered. Tr. 4166 (Morrill).

204. Licensee is instituting a formal system-by-system checkoff program for unlicensed operators, beginning May, 1980. Tr. 3117 (Rodriguez). Mr. Rodriguez testified that an unlicensed operator would be required to study a system and be checked off by a licensed operator to ensure that his level of knowledge is satisfactory. Id. However, Mr. Rodriguez stated that even under the new program, unlicensed operators need not be checked off on an important system before assuming responsibility for it. Tr. 3117-18 (Rodriguez). He also testified that unlicensed operators will still primarily be trained on the job by fellow unlicensed operators. Id. at 3126-27.

205. Mr. Rodriguez testified that because of the on-the-job training program for unlicensed operators at Rancho Seco, it would be possible for these personnel to be called upon to perform an operation for which they had never been trained and which they had never before performed. Id. at 3118. Presumably this includes emergency operations, since there is no evidence that these operations are treated specially.

206. In the 12 months prior to October, 1979, there was a high turnover of power plant helpers at Rancho Seco. Tr. 3120-22 (Rodriguez). During this period, 10-12 such

individuals left Rancho Seco. Mr. Rodriguez testified that the reasons for leaving were primarily economic and related to promotional opportunities and that SMUD has made some adjustments in its unlicensed operator program in response to this turnover. In the ensuing six months, two additional unlicensed personnel left Rancho Seco. Id. Thus, it is fair to assume that at least 12 to 14 unlicensed personnel at Rancho Seco have not been working there long and, since they are trained on-the-job, that these personnel are relatively inexperienced.

207. In response to anonymous telephone allegations, the Nuclear Regulatory Commission conducted an investigation at Rancho Seco between June 19 and July 6, 1979. CEC Ex. 39. Among the anonymous allegations (which were later determined to have come from an operations employee at Rancho Seco [Tr. 4167-68 (Morrill)]) was that the turnover of unlicensed station operators and other personnel is excessive and the training of new people is minimal. Tr. 3125 (Rodriguez); CEC Ex. 39. Mr. Rodriguez testified that he was familiar with this and other personnel complaints about minimal training. Tr. 3125-26 (Rodriguez). During this investigation, licensee management personnel stated to the NRC that the individual has the responsibility for keeping himself or herself informed of plant activities. CEC Ex. 39. Mr. Rodriguez testified that under the new formalized training program for unlicensed operators, this would still be true. Tr. 3128 (Rodriguez).

208. The Board's conclusion with respect to Board

Question H-C 34 is that unlicensed personnel training at Rancho Seco is minimal. The Licensee has suggested that the role of unlicensed personnel in responding to a feedwater transient is also minimal. Licensee's Finding 192. The evidence shows, however, that during a feedwater transient compounded by additional failures or errors, unlicensed personnel may be required to perform safety related operations on critical systems such as the high pressure injection system. Tr. 3111 (Rodriguez). We do not believe that unlicensed personnel should be required to undergo an excessively formal training program for their routine activities. But the Licensee has admitted that these operators may be called upon to perform operations for which they are not trained or experienced, and we do not find this acceptable in the case of safety related activities on emergency equipment during off-normal events. Accordingly, the Board finds that unlicensed personnel should be instructed and "checked off" on these operations before standing shifts.

(iv) Emergency Procedures

209. We already have made findings regarding operators' understanding of emergency procedures. Findings 174-177. However, we believe certain further findings relating to emergency procedures are warranted. In the course of the hearing, we have had an opportunity to review certain of

SMUD's emergency procedures, namely procedure D.5 entitled "Loss of Reactor Coolant/Reactor Coolant System Pressure." CEC Ex. 43 (revision of 9/5/79); CEC Ex. 46 (revision of 3/4/80). There was examination relating to these documents, which gave the Board an opportunity to review their contents. E.g., Tr. 3281, et seq; 3459, et seq.

210. Our examination of SMUD's LOCA procedures, particularly those for small breaks, leaves us with the conviction that SMUD's procedures are no models of clarity. Indeed, even after the extensive emphasis on small break LOCA's since TMI, we find SMUD's procedures to be plainly inadequate in certain respects. We provide the following examples of our concerns:[34]

(a) Procedure D.5 contains directions for operators to follow in the event of a LOCA. The procedure is divided into directions for three "cases": Case 1, a small leak within makeup pump capacity; Case 2, a medium leak (like PORV break) within HPI capacity; and Case 3, a large rupture in excess of HPI capacity. CEC Ex. 46, §2.0. Prior to setting forth instructions for each case, D.5 sets forth seven LOCA "symptoms" to guide operators in determining whether to apply D.5.[35] CEC Ex. 46, §3.0.; Tr. 3281-82

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34. The examples are from CEC Exhibit 46, the latest SMUD revision of Emergency Procedure D.5

35. In fact, there are more than seven symptoms since one, "possible annunciations," includes 11 sub-symptoms. Id.

(Rodriguez). However, the procedure does not indicate whether only one symptom or more than one symptom must be present, provides no priority among symptoms, and provides no indication whether particular symptoms are more applicable to one case or another. CEC Ex. 46; Tr. 3282 (Rodriguez). On examinations, Mr. Rodriguez testified that there is no priority but that some symptoms are more important than others. Id. Further, it appears that more than one symptom should be present before D.5 is utilized. Id. at 3283. We recognize that operators must exercise judgment on diagnostic matter [Id. at 3283], but conclude that procedure D.5 unnecessarily omits guidance that would be helpful and useful.

(b) A second shortcoming was aluded to above. There are three sets of LOCA procedures, depending on the size of the break. However, there are no instructions which guide an operator to a particular procedure to start with, nor any indication whether an operator is required to take action in any particular order. See CEC Ex. 46. On examination, it became clear that operators are not rigidly bound to begin with a particular procedure or even at the start of a procedure if they have reason to believe that other actions are required. Tr. 3317 (Rodriguez). Again, while operators may understand this,[36] there appears to be no sound reason for the procedure not to provide guidance.

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36. It is not possible from the record to reach any conclusions regarding operators' understanding of these particular procedures.

(c) At numerous places in D.5 operators are directed to verify that certain events have occurred or that particular conditions exist. However, the procedure provides no guidance, either as to individual provisions or generally, regarding operator actions if verification cannot be made.

211. This Board does not have the desire nor the expertise to determine the extent to which procedure D.5 is deficient. We do believe, however, that the foregoing examples and others that could be cited illustrate that there is room for improvement. We find that procedure D.5 and the rest of SMUD's emergency procedures should promptly be revised to make them more understandable and logical and to provide necessary guidance.[37]

(v) Feedback on Operating Experience

212. CEC Issue 3-2 expresses a concern regarding whether personnel are properly apprised of new information pertinent to the facility's safe operation, particularly information on operating experience at other reactors. The importance of adequate feedback is clear. There had been PORV failure prior to TMI where operators had been mislead by pressurizer level indication. NRC Ex. 2 at 1-1. Obviously, the lessons of those earlier events were not

37. We suggest but do not require that SMUD retain expert outside assistance to evaluate their procedures and to make necessary changes.

conveyed to and learned by TMI operators. The Staff's witness on this issue had no personal knowledge of whether important information is provided to Rancho Seco operators, and relied totally on the Licensee's interrogatory response "that through the Requalification lecture series significant operating events at Rancho Seco and other facilities _may be_ discussed." Wilson testimony at 8 (emphasis supplied); tr. 3827-28 (Wilson).

213. Mr. Rodriguez testified that operating personnel are made aware of significant events at Rancho Seco and other reactors by a variety of means. Rodriguez Testimony at p. 34. However, with regard to modification at Rancho Seco, Mr. Rodriguez's testimony is contradicted by the Staff's witnesses, who testified that Rancho Seco operators are not adequately provided this information. Finding 187. With regard to information from other facilities, the Board notes that Mr. Rodriguez's testimony does not address what information _is_ provided, but rather what information _could be_ provided:

> Events which occur at other units _and_ come to the attention of facility management, _and_ which are _deemed_ to be _directly_ pertinent to Rancho Seco operation _can_ also be communicated to operating crews through the Special Order program. Rodriguez testimony at 34-35 (emphasis supplied).

214. The only information which apparently is, in fact, regularly provided to operating crews is the weekly B&W newsletter summarizing events at its reactors.

Rodriguez Testimony at p. 35.

215. Mr. Rodriguez's carefully worded testimony on this issue implies that little or no information on events at other facilities is regularly provided to Rancho Seco operators. This implication was confirmed by the operators in their depositions. Although the shift supervisor testified that it was his responsibility to inform operators of these events [CEC Ex. 37 at 75], the senior operator ould not recall any such discussions with his shift supervisor aside from TMI. CEC Ex. 36 at 96-97. When directly asked how they learn of events at other units, both the senior operator and the operator indicated that the B&W newsletter was virtually their only regular source of such information:

> ...we get very few changes or transient conditions from other vendors. However, B&W sends out a weekly newsletter, and usually the transients are listed in there. CEC Ex. 36 at 96.

> Q: (by Mr. Ellison): Can you recall information being given to you about transients at other reactors through the Standing Order program?
>
> A: I don't know if it was through the Standing Order, but we had some writeups before on some different transients at other plants, but I don't know if it was an SO. I don't know if it was something that I read from the B&W handout or not.
>
> Q: Can you briefly identify which transients your speaking of?
>
> A: It was just a little newsletter, little flyer some of the supervisors were getting.

> Q: Were you required to reach that?
>
> A: It wasn't required. That wasn't required reading. It just that the supervisor used to bring it in because they're addressed to his home. So he used to bring it in and I'd just read through it. CEC Ex. 38 at 73-74.

216. Mr. Rodriguez testified that the annual one week requalification training "provides the opportunity" for instruction regarding events at other r-actors. Rodriguez Testimony at 35. The operator's testimony suggests this is true. CEC Ex. 38 at 74. But this means of communicating important information is a poor one because it occurs only one week per year.

217. Staff witness Wilson testified that the staff "believes that substantial improvement can be made in the process of dissemination of operating experience." Wilson Testimony at 11. He added on cross-examination that while organizations have been formed to distribute this information to Licensees, "we still need the mechanisms to get the proper information to the operator." Tr. 3837 (Wilson).

218. We share this concern that proper mechanisms need to be established for dissemination of information to operators. SMUD has no written criteria to determine whether particular information should be communicated to licensed operators. Tr. 3289-90 (Rodriguez). Rather, it is a "judgment evaluation" whether particular documents or data are passed on. Id. at 3290. While we agree that overly rigid criteria cannot be set, the record on feedback to operators indicates only limited dissemination:

-- Only operating type licensee event reports go to operators and even those are not required to be distributed within any particular time period. Tr. 3295 (Rodriguez). 38

--There does not appear to be regular provision for distribution of NUREG documents. Indeed, Mr. Rodriguez doubted whether the NRC's NUREG-0623 regarding the reactor coolant pump trip rationale had been communicated. Id. at 3310. Similarly, NUREG-0667, the latest statement on B&W feedwater transient response, probably was not communicated either. Id. at 3315.

--Mr. Rodriguez did not know whether the ICS FMEA or the AFW reliability analysis had been communicated. If so, they probably would not have been communicated in full. Id. at 3311-12.

219. Licensee has suggested that it does not want to overload its operators with unnecessary information. Tr. 3305 (Rodriguez). We share this view but believe that much data - such as that identified in the preceeding finding - is wholly or partially relevant to operators' training. If there is not sufficient time for operators to review and study such data, then that provides added support for our concern that operators have too little time set aside for training. Finding 178.

220. Based on the foregoing findings, the Board finds that the Licensee has not met its burden of proof with respect to this issue, and, indeed, that there is substantial evidence which demonstrates that Rancho Seco operators are not provided in a prompt and consistent manner with important data regarding the Nuclear industry. Licensee is directed promptly to adopt criteria to guide distribution of relevant data, including the type identified in Finding 218.

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38. There is no exam or testing of operators' understanding of the LER's which are distributed. Id. at 3296.

H. Control Room Configuration

Board Question H-C 31:

> Are there features of Rancho Seco's control room design and configuration which make it difficult for operators to avoid a loss-of-feedwater transient?

221. This issue, although ill-worded,[39] concerns whether there are defects in the design of the Rancho Seco control room which could affect operators' ability to diagnose and respond to a loss of feedwater transient. Wilson Testimony at 3.

222. Upon review of the evidence, we conclude that there are no serious design shortcomings in the Rancho Seco control room. The Rancho Seco control room has compact control consoles which allow operating personnel quick access to controllers for a wide variety of equipment. The overall control room layout minimizes the amount of movement the operator must make in taking actions involving multiple pumps and valves. Rodriquez Testimony at 40. The compact character of the Rancho Seco control room and the relatively small number of displays makes it substantially different from the TMI control room. On the whole, it is better than TMI's, especially during normal operation. Minor and Bridenbaugh Testimony at 17-18.

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39. The configuration of the control room has very little, if any, effect on wheter or not a loss of feedwater transient will occur. Wilson Testimony at 2; Rodriquez Testimony at No. 39.

Indeed, NRC Staff witness Wilson concludes that the Rancho Seco control room is one of the best. Wilson Testimony at 4-5.

223. The foregoing findings do not mean that the Rancho Seco control room could not be improved. In 1976, the Electric Power Research Institute (EPRI) published a study comparing the Rancho Seco control room with four others from a human engineering standpoint. "Human Factors Review of Nuclear Power Plant Control Room Design," identified as CEC Ex. 33. The EPRI study identified several human engineering weaknesses with the Rancho Seco control room design. The more substantial problems were:

a. Functionally related control connsoles are separated. Tr. 2971-73 (Rodriquez).

b. Control board is unwieldy. Id. at 2974.

c. Controls and instrumentation are located in areas outside the primary control room area or outside the operators' line of vision. Id. at 2976.

d. Rod monitor display is poorly placed with respect to the reactor control panel. Id. at 2978.

e. Auxiliary feedwater controls are not grouped with main feedwater controls. Id. at 2980.

f. A "B" switch on the functionally grouped (A and B panels) safeguards panel is located on the "A" panel. Id. at 2981.

g. There is no differentiation in apperance between some switch lights and indicator lights on the safeguards panel. Id. at 2998-99.

h. Printed numbers on chart recorders are difficult to read and have been assigned more parameters than they are designed for. Id. at 3019-20.

224. SMUD has not previously given serious consideration to improving the Rancho Seco control room, even since the EPRI study was completed. Id. at 2975. The Board believes that in light of the design sensitivities of the B&W system, SMUD should consider eliminating these identified weaknesses, even though the Rancho Seco control room may have fewer defects than most. The Board is confident this will occur, since SMUD has recently contracted for a human factors study of its control room to be undertaken this year. Id. at 2974. Such a study comes at a particularly opportune time, as the study will evaluate the new instrumentation incorporated at Rancho Seco since the TMI accident.

I. Instrumentation

Board Question CEC 5-3a:

> Are the special features and instruments installed at Rancho Seco adequate to aid in diagnosis and control after on off-normal condition engendered by a loss-of-feedwater transient?

Board Question H-C 22:

> What instrumentation is available to give positive indication as to whether or not the coolant is subcooled throughout the core at all times? How does that instrumentation work? In the event that a non-subcooled condition is indicated, what instrumentation would then give reliable information on the water level in the core?

225. The concerns relating to Rancho Seco instrumentation expressed in Board Question CEC 5-3a and H-C 22 are significant in view of the sensitivities of the B&W reactor system. As stated earlier, if those design sensitivities are not substantially reduced, and they have not been, then there is a need to upgrade substantially the instrumentation available at B&W facilities such as Rancho Seco. See Section V.A.

226. It is undisputed that Rancho Seco has considerable instrumentation for diagnosis of and following the course of feedwater transients. Rodriguez Testimony at 41-45. In addition, SMUD has installed more instrumentation to upgrade auxiliary feedwater indication [CEC Ex. 25] and also has installed two subcooling meters. Tr. 3405 (Rodriguez); Rodriguez Testimony at 41-42. In general, we find this instrumentation to be adequate for diagnosis and control of feedwater transients.

227. We do have concern, however, that certain additional instrumentation has not been provided at Rancho Seco. First, there is no core level indication available, despite the fact that keeping adequate core cooling is a primary concern in operating any reactor. Tr. 484, 508 (Lewis). Direct core coolant level indication has been proposed to guide operators during saturated conditions. Id. at 484 (Lewis). While core level indicators are indisputabl hard to design, Dr. Lewis testified that void detectors are a possibly preferable alternative because they are not single point measurements. Id. at 484, 490. He described conceptually how this indication could be devised:

> A void indicator, or a quality indicator would be some measurement of the -- of essentially the effect of density of the fluid. I will describe a conceptual one in a moment, and it would have an advantage over the level indicator in that it is a single point measurement. It does not require the matching at two different points as a level indicator, as a rear level indicator would. An example of one, for example, might be a capaciter in which one has two parallel places in which the fluid you are interested in, between which the fluid you are interested in flows, and you measure the capacitance of the thing which is the measure of the mean dialectric constant of this stuff in between, and water has a different dialectric constant from steam, so we learn directly, and there are lots of ways of implementing this, what the mean density is of the fluid. If you have bubbles going through, you have a time fluctuating density, but it is a fairly simple measurement, and that is just one example. You might be able to do it non-invasively by measuring the speed of sound across a pipe, which is again different between steam and water, so the problem is not whether there is a possible way, but the problem is choosing the optimal design of such a thing, and my personal view,

> and this is all I am offering, is that would be beneficial to the utilities, to the NRC, to the operators, and all of us. Id. at 490-91.

228. Licensee has argued that no core level indications are needed because operators have other instruments, such as the subcooling meter, which indicate whether adequate core inventory is present and that even if operators had core indicators, it would lead to no different operator actions. SMUD Findings 209-12. We disagree. Most of the additional instrumentation implemented since TMI was not "needed" in the sense that operators already could gain the data from other means. See discussion of auxiliary feedwater changes, Section V.C. However, by giving greater redundancy or more direct indication, the system's reliability is enhanced at least somewhat. The same, in our view, applies to core level indicators. Such indicators would give operators more direct indication of the condition of the primary system and hence would enhance operators' confidence that the actions taken were correct and necessary.[40]

229. A second instrumentation change that should be implemented relates to pressurizer level. It is undisputed that if Rancho Seco experienced at TMI-type accident, pressurizer level would read off scale. Tr. 3032-33 (Rodriguez). A wider range pressurizer level

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40. Such core level indicators would be all the more useful in view of the fact that pressurizer level is not an accurate indication of core level when subcooling is not maintained. Norian Testimony at 3-4; Tr. 933 (Jones); Tr. 1369 (Norian).

indication has been endorsed by the NRC Staff and would enhance operators' ability to understand the condition of the primary system. NRC Ex. 4 at 5-64.[41]

230. Finally, CEC witness Minor suggested that natural circulation flow indication should be provided, particularly in view of the increased emphasis on natural circulation cooling. Bridenbaugh and Minor Testimony at 16, 19. As of this time, operators verify natural circulation by indirect means using temperature indications. Tr. 3444 (Rodriguez); Tr. 3894, 3895 (Wilson). However, as demonstrated by the fact that Rancho Seco operators initially had difficulty verifying natural circulation after post-TMI training, we think additional, more direct indication of successful natural circulation initiation would be exceedingly helpful. This particularly is appropriate since issuance of I&E Bull. 79-05C, which places new emphasis on natural circulation. See Section V.F. supra. We recognize that a natural circulation meter may be difficult to design, given the low flow rates involved. Tr. 3619 (Minor). However, there is no probative evidence to indicate that such instrumentation would be impossible to implement.

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41. NRC Exhibit 4 recommends a number of other instrumentation changes for B&W plants. Id. at 5-64. While the record is not clear which of the parameters are presently covered by Rancho Seco equipment, we believe SMUD should consider implementation of those NRC recommendations as promptly as possible.

231. In conclusion, we find that the overall instrumentation of Rancho Seco has been improved since TMI but that further actions can and should be taken. SMUD should immediately commence with high priority to investigate and install core level indication, wide range pressurizer indication, and natural circulation flow meters.

J. Pressurizer and Quench Tank Sizing

Board Question H-C 21:

> Do the fundamental transient assumptions utilized in sizing Rancho Seco's pressurizer and quench tank truly represent extrema or are there other expected transients (or even transients already experienced elsewhere) which call for greater capacity in these pieces of equipment?

232-234. As findings 232-234, which basically describe the pressurizer, the Energy Commission suggests that the Board adopt SMUD's Proposed Findings 88-90.

235. Although the Rancho Seco pressurizer meets current NRC design criteria, those criteria address only over-pressure events. The criteria do not address under-pressure events and there are a number of these transients or accident conditions that can result in emptying the pressurizer or causing it to go solid. Tr. 1465 (Matthews). For instance, the pressurizer may empty during a depressurization or overcooling transient. This also can occur in an overheating transient in which the "feed and bleed" mode of core cooling has been exercised for an extended period of time and then there is a start of AFW delivery. Tr. 1128 (Jones). Also, an anticipated transient without reactor trip could cause pressures beyond the pressurizer's design criteria. Tr. 1680 (Matthews). Finally, the design basis for sizing the pressurizer does not seek to accommodate continuous fluid inventory losses that may occur due to a break in the system. Tr. 1127, 1128 (Karrasch); Tr. 1681 (Matthews). For these reasons, the NRC is re-evaluating its criteria for pressurizer sizing.

236. Licensee's witnesses testified that a B&W analysis of operating data from all B&W PWR's shows that in every instance of a reactor trip, including those involving overcooling events, the pressurizer liquid volume was maintained; that is, the pressurizer did not empty. E.g., Tr. 771-73, 775-77 (Karrasch and Jones). However, they admitted that on a number of occasions, pressurizer level indication in the control room was lost. Tr. 774 (Karrasch).

237. During a normal reactor trip, there is a tendency for B&W-designed plants to lose pressurizer level indication and depressurize to near or below the safety features actuation setpoint. Operators at B&W plants historically have attempted to dampen the system response by securing letdown flow and starting a second makeup pump. There are some indications that occasionally operators have also throttled back feedwater and/or actuated HPI (high-pressure injection) to reduce the amount of primary system depressurization. NRC Ex. 4 at 5-13. The B&W Transient Response Task Force concluded that the loss of pressurizer level, along with the need for operator actions such as anticipatory use of HPI, places the plant in an undesirable condition and should be remedied. Based on this conclusion, the Task Force made these recommendations:

1. Following a reactor trip, pressurizer level should remain on scale, and system pressure should remain above the HPI actuation set point;

2. The system response (e.g., secondary pressure) should be appropriately modified in order to meet the above two objectives; and

3. Meeting these objectives should be independent of all manual operator actions (e.g., control of feedwater, letdown isolation, and startup of a makeup pump). NRC Ex. 4 at 5-13 and 5-14.

238. Staff witness Matthews testified that there are several potential approaches to meeting these recommendations. These include increasing the size of the pressurizer, increasing the secondary side volume of the OTSG, changing the set point on the turbine bypass valves, and expanding the range of the pressurizer level indication. Tr. 1462-63. However, a Staff analysis of increasing pressurizer size suggests that even with a 50 percent increase, pressurizer level and pressure loss could not be entirely avoided. Id. at 1461-62. Mr. Matthews also pointed out that increasing the secondary side volume of the OTSG is "pretty extreme" (id. at 1463), and that expanding level indication would not address all the concerns. Id. at 1464.

239-244. As findings 239-244 relating to the pressurizer relief tank, the Energy Commission suggests that the Board adopt SMUD's Proposed Findings 93-98.

245. In conclusion, we find:

(a) The fundamental transient assumptions utilized in sizing Rancho Seco's pressurizer and quench tank do not represent extrema. There are some expected transients which will cause the pressurizer to empty and there are also other expected transients that will discharge fluid to the PRT in excess of its capacity.

(b) To date, it appears that the pressurizer has not been emptied during any transient. The PRT, on the other hand, has been overfilled on three occasions. This overfilling should not, however, have any adverse safety consequences.

(c) During normal reactor trips, there is a tendency for B&W designed plants to lose pressurizer level indication and to depressurize to near or below the ESFAS set point. Though there are several potential solutions to this undesirable conditions, none appear sufficiently feasible or effective so as to warrant implementation or site specific study.

K. Hydrogen Concentration

Board Question H-C 20:

Does Rancho Seco's present system for coping with hydrogen release in containment provide for:

a. recombiner availability early enough to respond to a situation like that at TIM-2?

b. proper radiological protection of the surroundings if purging is depended upon?

246. This question related to concern about the adequacy of means available at Rancho Seco to control hydrogen concentrations below the flammable level, a concern emanating from the hydrogen spike which occurred at TMI.

247. There are two methods available for removal of hydrogen from a containment building: a purge system and a recombiner. Rancho Seco has a purge system but does not have a recombiner. Dieterich Testimony at 21; Greene Hydrogen Testimony at 2. After a severe accident, the purge system could not be used for approximately 13 or 14 days because earlier use would lead to large radioactive releases to the environment. Dieterich Testimony at 20; Tr. 2843 (Greene).

248. A recombiner may be used earlier in an accident sequence than a purge system because the recombiner vents back into the containment building rather than releasing radioactivity to the environment. Tr. 2842-44 (Greene). The NRC has recognized the advantage of hydrogen recombiners by requiring them, rather than purge systems, on newer facilities. 10 C.F.R. §5044(g).

249. While there is no hydrogen recombiner presently available which would cope with the rapid hydrogen buildup experienced at TMI, a hydrogen recombiner, always available on site, would allow earlier coping with hydrogen generated in an accident than does a hydrogen purge system and would possibly permit early enough utilization to keep hydrogen concentrations below the 4 percent flammable level. Dieterich Testimony at 22; Tr. 2176 (Dieterich). This would particularly be true for an accident more severe than present design basis but less severe than the TMI accident. Id. at 2363.

250. SMUD witness Dieterich testified that Rancho Seco has arranged for use of a hydrogen recombiner on a 24-hour notice basis. Dieterich Testimony at 21. However, that recombiner is not and has never been on site, there are no procedures for installation or use of that recombiner, Rancho Seco personnel have never been trained in the use of such a recombiner, and there is no evidence that a dedicated containment building penetration has been made available for the recombiner. Tr. 2053-55 (Dieterich); 3266 (Rodriguez).

251. NRC regulations do not require that Rancho Seco have a hydrogen recombiner on site and always available. However, those regulations do not prohibit use of such a system by SMUD. 10 C.F.R. §50.44(g).

252. It is desirable to control hydrogen concentrations below the flammable level since hydrogen combustion,

either by slow burn or by detonation, could damage the containment or equipment in containment, making it more difficult to cope with the consequences of a severe accident. Tr. 2176 (Dieterich).

253. In conclusion, the Board finds that: (a) Rancho Seco does not have a readily available hydrogen recombiner, nor procedures for use of one on loan. Under these circumstances, we find that Rancho Seco has no hydrogen recombiner that effectively might be used shortly after a severe accident. (b) The present method for coping with hydrogen buildup, the purge system, cannot be utilized early in an accident sequence without having severe environmental consequences. (c) A hydrogen recombiner could potentially have significant benefits in terms of keeping hydrogen concentrations below the flammable level. We, therefore, conclude that SMUD should immediately install one or more hydrogen recombiner systems; or implement procedures and training which ensure that the borrowed recombiner will, in fact, be available for use promptly after an accident.

L. Venting Back Into Containment

CEC Issue 5-1:

> Whether those systems identified as contributing to the release of radioactivity during the TMI accident, which are outside containment, should be changed to vent into the containment building?

254. This issue reflects concern that during the TMI accident, there were diverse pathways for escape of radioactive materials from the TMI containment. Mann Testimony at 1-11, inserted following Tr. 2926; Wing Testimony at 2, following Tr. 2740; Dieterich Testimony at 17. Thus, the issue raises questions whether similar release paths may exist at Rancho Seco and, if so, whether additional measures need to be implemented to ensure that such releases do not occur at Rancho Seco. We find, however, that the evidence does not support imposition of this, or any similar, requirement.[42]

255. One contributor to release of radioactivity during the TMI accident was the fact that the TMI containment isolated only on high reactor building pressure. This delayed isolation until several hours after the accident began, thus permitting radioactive releases. Dieterich Testimony at 18. The Rancho Seco containment isolates on low primary system pressure (1600 psig), as

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42. Licensee has argued that the Energy Commission failed to sustain its burden of going forward with evidence. SMUD Finding 235. We believe, however, that the Energy Commission presented sufficient evidence by documenting the TMI release paths and raising the question whether similar paths exist at Rancho Seco.

well as on high reactor building pressure and thus would come very early in a TMI-type accident. Id. at 18-19.

256. However, even after the TMI containment had isolated, significant releases from containment to the auxiliary building were experienced due to the necessity to operate certain systems, including the letdown system. Mann Testimony at 14. Indeed, the letdown system was probably the most significant pathway for radioactive releases at TMI. Tr. 3172 (Donohew). Accordingly, early containment isolation, while helpful, does not ensure that there will be no releases of radioactivity from the reactor building.

257. However, SMUD has instituted two programs to attempt to ensure that radioactive releases to the environment will not occur. First, SMUD has identified essential and nonessential systems within containment and has taken steps to ensure that all nonessential systems will be isolated immediately upon either high reactor building pressure or low primary system pressure. Wing Testimony at 3-5. The Board finds that this program should reduce releases due to unnecessary operation of systems after an accident. In addition, SMUD has instituted a leak reduction program concerning its radwaste system, designed to ensure that leakage in that system will be kept to a minimum amount. Id. at 3-5.

258. There is no capability at Rancho Seco to vent back from the radwaste system into containment. Tr. 3189

(Donohew). Such a vent back capability, if available, would allow use of the containment as an additional storage facility for radioactive waste if the radwaste system were not sufficiently sized for all the waste present in the auxiliary building or if the radwaste system should have leaks. Id. at 3188-89. While this capacity merits study and is part of the NRC's post-TMI action plan (Wing Testimony at 8), it is also a proposal which has possible drawbacks. Tr. 2129, 2134-36 (Dieterich); Tr. 3175-76 (Donohew).

259. In conclusion, the record does not support imposition of a vent-back requirement.

M. Timetable for Long-Term Modifications

Board Question FOE III(c):

> The NRC Orders in issue do not reasonably assure adequate safety because there is no reasonable time for implementation of the long-term modifications established in the Commission orders.

260. The May 7 Order provides that SMUD shall "as promptly as practicable" accomplish the long-term modifications set forth in that Order. May 7 Order at 8.

261. The first long-term modification is:

> The licensee will provide to the NRC Staff a proposed schedule for implementation of identified design modifications which specifically relate to items 1 through 9 of enclosure 1 to the licensee's letter of April 27, 1979 and would significantly improve safety. May 7 Order at 5.

262. The record in this proceeding discloses that SMUD has agreed to implement several further AFW modifications, including safety grade initiation and control of AFW, automatic loading onto the diesels upon a loss of offsite power, and upgrading AFW flow meters to safety grade. Tr. 2099, 2116 (Dieterich); CEC Ex. 21, Encl. 1 at 4-5. However, SMUD has not completed its AFW reliability study in a manner satisfactory to the NRC Staff or to this Board, which makes it impossible for this Board to assess the adequacy of SMUD's actions. See Section V.C. Accordingly, this Board cannot find that SMUD has met its obligation to accomplish the first long-term modification as promptly as practicable.

263. The second long-term modification called for performance of an ICS FMEA as soon as practicable. May 7

Order at 5 & 8. The ICS FMEA is neither complete nor adequate. <u>See</u> Section V.B. Accordingly, this second long-term modification also has not been performed.

264. The third long-term modification pertains to the upgrade of the anticipatory reactor trip to safety grade. May 7 Order at 5. That upgrade is currently being implemented by SMUD [Dieterich Testimony at 26], and, accordingly, we find that SMUD has met its obligations in this regard.

265. The fourth long-term modification pertains to simulator training for Rancho Seco operators. May 7 Order at 5. This simulator training is specifically designed to acquaint operators with the TMI sequence of events. SMUD has fully complied with this requirement. Dieterich Testimony at 27.

N. Controlled Filtered Venting

CEC Issue 5-2

> Whether the containment building should be modified to provide overpressurization protection with a controlled filtered venting system to mitigate unavoidable release of radionuclides?

266. CEC Issue 5-2 poses the question whether the Rancho Seco containment building should be modified to include a controlled filtered venting system ("CFVS"). Such systems are designed to prevent the uncontrolled release of radioactive materials that would result if a serious overpressurization accident resulted in failure of the containment. As will be seen, there is considerable debate regarding the precise numbers of deaths and illnesses and the economic costs that such an uncontrolled release would cause at Rancho Seco. However, there can be little debate about two facts: an uncontrolled release from the containment of the kind guarded against by a CFVS would have tremendous human and economic costs; and if a CFVS were installed and performed as intended, that system would vastly reduce human and economic costs. Because of these perceived benefits, widespread interest has developed in methods, like CFVS, which might substantially reduce the impact of uncontrolled radioactive releases, especially since the TMI accident. Indeed, when such a release was feared at TMI, the NRC began design work to enable a CFVS to be installed there. Nix Testimony at 16, following Tr. 2403. It is appropriate, therefore, that

this issue be considered in this hearing, which arises from that event.

267. As a preliminary matter, it is appropriate to describe conceptually a CFVS and explain its purposes. A CFVS is designed to prevent an uncontrolled radioactive release from the containment due to overpressurization. The CFVS accomplishes this by venting accident-generated gases in a controlled manner through a filtration system. Nix Testimony at 8. It has two major components: (1) an interface with the containment atmosphere, and (2) a filtration system for scrubbing and slowing the escaping radioactive gases. Id.

The containment interface consists of large penetrating pipes, sealed off by temperature and pressure-sensitive discs. Nix Testimony at 10. There are different designs for this configuration. Energy Commission's Underground Siting Study designed a passive system, by which the discs would burst without external power or operator action. Sandia Laboratory has designed a conceptual system requiring power and operator activation. Id. The disc is designed to burst or activate to relieve containment pressure so that the radioactive gases are removed from the containment in a controlled manner rather than being released to the environment in an uncontrolled way.

Once the discs rupture, the gases would expand and cool in a pressure relief volume. Id. at 10. Again, designs of the CFVS differ. The Energy Commission Study provided for a pressure relief volume filled with natural stone. Id. However, CFVS filtration systems could use various types and combinations of rock, sand, natural soil, gravel, and charcoal in different systems above ground, on the surface, or underground and can include a scrubber for an added barrier to releases to the environment. Id. at 10, 12.

268. Before this hearing began, SMUD moved for summary disposition contending that a CFVS would mitigate accidents beyond the design basis accident and, therefore, this issue improperly challenged the Commission's General Design Criteria for Nuclear Power Plants, set forth in 10 C.F.R. Part 50, Appendix A, criteria 16 and 50. After carefully considering oral and written arguments from all parties,[43] the Board determined that this issue did not challenge Commission regulations and was a proper subject of inquiry. Prehearing Conference, Feb. 6, 1980. The Commission's criteria require a containment design that will remain leak-tight throughout a design basis accident. Id. However, these are minimum standards which do not limit the Commission's, and by delegation this Board's, power to require additional protection pursuant to 10 C.F.R. §50.109, provided the additional protection will not compromise the ability of the containment building to remain

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43. The Staff neither supported nor opposed SMUD's motion. See "NRC Staff's Response to Licensee's Motion for Summary Disposition," dated February 4, 1980, at 9.

leak-tight throughout design basis accidents. The Energy Commission contended that a CFVS would not compromise containment integrity for design basis accidents. Accordingly, the Board denied SMUD's motion and, later, its request for reconsideration, and heard evidence on this issue. In considering the evidence, the Board viewed the impact of a CFVS on containment integrity for design basis accidents as an important jurisdictional question. As we have found below however, a CFVS would not diminish existing containment integrity for design basis accidents. Therefore, the Board sees no conflict between a CFVS and any Commission regulation.

269. In opposing SMUD's summary disposition motion, the Energy Commission suggested that this issue encompasses the question of whether SMUD should conduct a specific feasibility study of a CFVS for Rancho Seco. California Energy Commission's Response to Licensee's Motion for Summary Disposition of Contention 5-2 of the California Energy Commission, dated February 4, 1980, at 8. As will be seen, there are site and facility specific design issues which must be resolved before the costs and difficulty of implementing such a system at Rancho Seco can be accurately determined. For this reason, the Board considers a site specific feasibility and design study, such as that being prepared for the Indian Point and Mount Zion reactors (see Findings 324), to be prerequisite to consideration of the merit of implementing such a system. Since this

study has not yet been undertaken, the Board sees the question of whether such a study should be required to be the central issue presented by CEC Issue 5-2.

270. The Energy Commission overwhelmingly has met its burden of going forward.[44] Accordingly, SMUD has the burden of proving that CFVS does not merit study for Rancho Seco. In opposition to the need for a CFVS study, SMUD has raised a number of issues which the Board summarizes as follows:

(a) That the risk of containment failure at Rancho Seco is so low that there is no reason to be concerned about mitigating such an accident;

(b) That a CFVS cannot be designed to effectively mitigate such accidents and, impliedly, that a feasibility study by SMUD cannot reasonably be expected to resolve these design issues; and

(c) That the CFVS concept is under study by others and it would be premature or duplicative for SMUD to examine it now.

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44. The CEC has met its burden of going forward with evidence. Briefly, the Energy Commission presented the detailed testimony of Daniel Nix, project manager for the $1.3 million Underground Siting Study, one of the most extensive studies of CFVS ever undertaken. Mr. Nix presented testimony that CFVS could dramatically reduce the largest remaining risks in the operation of nuclear power plants. The testimony showed that the potential public health and economic consequences of an over-pressurization accident are very large and that a CFVS described by Mr. Nix could eliminate those consequences reliably, economically, and compatibly with other reactor safety system.

Given Licensee's assigned burden of proof, the Board has organized its decision according to these assertions. However, we first examine the threshold question of whether a CFVS would compromise containment integrity for design basis accidents in order to satisfy ourselves that there is no jurisdictional bar to consideration of this issue.

(i) Effect of CFVS on Containment Integrity for Design Basis Accidents.

271. The Licensee supported its claim that a CFVS would violate the Commission's General Design Criteria through the testimony of Mr. Dieterich. Dieterich Testimony on California Energy Commission Issue 5-2 at 5, following Tr. 1988 ("Dieterich CFVS Testimony"). Before considering the substance of Mr. Dieterich's testimony on this issue, however, it is appropriate to consider his expertise on CFVS. Mr. Dieterich is a senior engineer in SMUD's Generation Engineering Department. Id. at 3. SMUD has never peformed any independent analysis of the benefits and detriments of a CFVS at Rancho Seco. Tr. 2210 (Dieterich). The Board's examination of Mr. Dieterich's statement of professional qualifications does not disclose any evidence of his having any familiarty with the CFVS concept prior to his employment with SMUD. Dieterich Testimony at 3-4. Thus, it appears that Mr. Dieterich has no personal knowledge with regard to a CFVS and that his acquaintance with these systems is based entirely on a review of studies done by others,

particularly the Energy Commission's Underground Siting Study. Tr. 2210 (Dieterich). (This study, which investigated the CFVS concept at considerable length, was introduced into evidence as SMUD Ex. 14.) Mr. Dieterich acknowledged that he has never personally participated in any studies of the CFVS concept and that his testimony is indeed based only upon the work of others. Tr. 2210 (Dieterich). In weighing Mr. Dieterich's testimony, therefore, the Board is constrained to consider his lack of experience in this field, especially in comparison to Energy Commission witness Mr. Nix and Staff witness Mr. Meyer, both of whom have professional experience in this area. Nix Testimony at 1-2; Meyer Testimony on Professional Qualifications, following Tr. 2786.[45]

272. Mr. Dieterich was the only witness who testified that a CFVS would violate the Commission's General Design Criteria. On cross-examination, however, he explained that he based his views on the presumption that a CFVS would activate at the temperatures and pressures associated with design basis accidents. Tr. 2229-30 (Dieterich).

273. Mr. Dieterich was the only witness who presumed that a CFVS would be designed to activate at the pressures

45. Mr. Nix has had extremely extensive experience relating to CFVS, having been project manager for the Energy Commission's Underground Siting Study. Nix Testimony at 1. Mr. Meyer has been involved with the subject matter since the summer of 1979 in conjunction with analysis of care mitigation devices for Zion and Indian Point. Meyer CFV Testimony, Professional Qualifications. Mr. Greene, the other staff witness, has experience with containment design and hydrogen control but no experience with CFVS. Greene CFV Testimony, Professional Qualifications.

and temperatures associated with a design basis accident. Both Energy Commission witness Nix and Staff witness Meyer testified that a CFVS could be designed to remain as leak-proof as any other containment penetration up to and beyond these pressures and temperatures and still be effective. The CFVS they envisioned would only activate at temperatures or pressures greater than those of a design basis accident. Meyer Testimony at 2; Nix Testimony at 10. Such a CFVS would be consistent with NRC General Design Criteria. In fact, a CFVS would be complimentary, since it allows every safety system to function as intended; only if every defense failed would this last defense, the CFVS, be actuated. Tr. 2728 (Nix).

274. Mr. Dieterich attempted to explain this presumption in two ways. He suggested first that a CFVS could not be designed to actuate reliably at a prescribed set point yet be leak tight below this point. Tr. 2284-86 (Dieterich). Energy Commission witness Nix, however, noted that any desired degree of reliability could be achieved:

> It is possible to carefully control the pressure at which the [rupture] disks will fail by machining portions of the disk surface to a required thickness. Temperature sensitivity was achieved by using alloys at prescribed temperatures. The system is amenable to testing and any desired degree of design reliability may be achieved by placing disks in series. Thus, the access points to the CFV system can be make as, or more, reliable than any other containment penetration. Nix Testimony at 11.

NRC witness Meyer also noted that the passive nature of the Energy Commission proposed system would make its reliability "quite high". Tr. 2836 (Meyer). On cross-examination, SMUD witness Dieterich revealed that his unshared skepticism was rooted in the error bands around each disc. Tr. 2285-86 (Dieterich). He conceded, however, "[b]y putting more and more discs in the series, I would relieve the possibility or probability of a failure at a low set point, however, I'd increase the possibility of failure at a high set point." Tr. 2285 (Dieterich). Mr. Dieterich further admitted that only premature failure could make a system worse off than it would be without a CFVS. Tr. 2369-70 (Dieterich). If a CFVS were designed with set point conservatisms on the "high side", the only "danger" would be that containment w[illegible]d rupture before the CFVS activates -- a danger present with or without a CFVS. Tr. 2370 (Dieterich); Tr. 2[illegible]3 (Nix).

275. Mr. Dieterich's second rationale for presuming that a CFVS would actuate at design basis temperatures and pressures was that it must be set there if it is to actuate before containment failure. Tr. 2232-36 (Dieterich). Witness Dieterich seemingly assumed at this point that if the set point were above design basis temperatures and pressures, the containment building, which is designed to those parameters, would fail before the set point was reached. Id.

276. This idea was not only contradicted by every other witness in the proceeding, but by Mr. Dieterich himself. No witness could guarantee containment integrity

beyond its designed-for pressures and temperatures. Tr. 2358-59 (Dieterich); Tr. 2691, 2707 (Nix); Tr. 2811 (Greene). But the witnesses, including Mr. Dieterich, agreed that it was reasonable to expect the containment building to withstand conditions well beyond its design basis. Greene Testimony at 7; Dieterich CFVS Testimony at 3; Meyer Testimony at 4; Tr. 2215 (Dieterich); Tr. 2830-32 (Meyer).

277. Realistically, then, one can assume that containment failure becomes possible as conditions exceed design basis assumptions; but the probability of containment failure is small and increases as conditions deteriorate further. Nix Testimony at 8; Tr. 2369 (Dieterich); Tr. 2810-11 (Greeme). The Rancho Seco containment design pressure is 59 psig. Greene Testimony at 3; Dieterich CFVS Testimony at 3; Tr. 2214 (Dieterich); Tr. 2806 (Greene). Witnesses for all parties agreed that containment failure at Rancho Seco was very unlikely at pressures less than 70 psig. Tr. 2688 (Nix); Tr. 2830 (Meyer); Tr. 2215 (Dieterich). It is evident, therefore, that there is a comparatively wide range within which a CFVS can be set to actuate that will be above design basis pressures yet below the probable failure pressure of the containment.

278. Thus, the Board finds that a CFVS can be reliably set to actuate at conditions beyond design basis assumptions but below the point of probable

containment failure. The Board accordingly rejects witness Dieterich's presumption that a CFVS must actuate at or below design basis pressures and temperatures and, with it, his statement that a CFVS would therefore violate the Commission's General Design Criteria.

(ii) Risk of Containment Overpressurization

279. The need for a CFVS at Rancho Seco depends upon the risk of accidents at Rancho Seco for which the CFVS would provide additional protection. Therefore, we consider first whether there are potential accidents that can exceed the capability of existing safety systems and, then, the risk of these accidents.

280. There is no single reactor accident. Indeed, the complexity of a modern light water power reactor gives rise to large numbers of possible accidents of different likelihood and severity. Nix Testimony at 2. The safety systems at reactors are designed to control a selected spectrum of accidents, termed design basis accidents, which are only a limited sample of potentially hazardous events. Nix Testimony at 3.

281. The design basis accident for the containment building at Rancho Seco is a double-ended rupture of the largest pipe in the primary system. Dieterich CFVS Testimony at 3. Analysis of the effects of such an accident resulted in the Rancho Seco containment building being designed to withstand pressures of 59 psig. Greene Testimony at 3; Dieterich CFVS Testimony at 3. While

this design pressure calculation incorporates several conservatisms, these are necessary to guarantee that the containment will, in fact, tolerate the design basis pressures. Tr. 2832-34 (Meyer and Greene). The possibility of containment failure at Rancho Seco is, therefore, integrally dependent on the choice of this design basis accident. Nix Testimony at 3. It is also important to recognize that the design basis accident does not bound all accidents. There are accidents which could occur at Rancho Seco which would result in pressures and temperatures beyond those of the design basis accident and hence exceed the design of the containment building. Dieterich CFVS Testimony at 3; Tr. 2220 (Dieterich). Accordingly, in considering the adequacy of existing protections at Rancho Seco, it is important to remember that Rancho Seco has not been designed to contain all potential accidents or even a single "worst case" accident. Rather, Rancho Seco's containment building is designed to withstand a single design basis accident which is less severe than many postulated accident sequences. The need to upgrade Rancho Seco with systems like a CFVS depends upon the <u>risk</u> of accidents more severe than this design basis accident.

282. The best source of data regarding the risk from nuclear reactor accidents is the Reactor Safety Study (WASH-1400). Although this study screened about 130,000 accident sequences, it did not examine all

accidents. Nix Testimony at 4-5; Tr. 2476 (Nix). Indeed, two specific findings of the Risk Assessment Review Group were that there is an inherent logical inability to consider all accident sequences [Tr. 2458, 2521 (Nix)], and there is especially great uncertainty in predicting operator and other human responses under accident sequences. Tr. 2508 (Nix).

283. The TMI accident illustrates the uncertainty of the Reactor Safety Study probabilities, especially under accident conditions which, of course, would exist if a CFVS were close to actuation. First, TMI was not one of the 130,000 accident sequences considered in the study. Nix Testimony at 5; Tr. 2458, 2476, 2507-09, 2519 (Nix). The fact that the first core damage accident experienced in a U.S. commercial reactor was not considered in the Reactor Safety Study demonstrates that the study was far from exhaustive. However, the study did consider an accident involving all the major elements of TMI except for the improper valve position on the AFW system and the reactor coolant pump trip. Tr. 2513, 2517-18, 2521-22 (Nix). This sequence, which involves fewer coincident errors or failures than TMI, should be substantially more probable than TMI. Yet the Reactor Safety Study gave this sequence a probability of 2×10^{-8} (or one occurrence every 200,000,000 reactor years. Tr. 2518 (Nix). The occurrence of the even less probable TMI sequence in only a few hundred reactor

years demonstrates the vast uncertainty of the Reactor Safety Study predictions.

284. A principle cause of this uncertainty is that the probabilities associated with component failures are very uncertain. Tr. 2476, 2502-03 (Nix). The Reactor Safety Study, however, relied upon these numbers as accurate "frequency of occurrence" estimates. Tr. 2475 (Nix). In this sense, the study misused these estimates to make dubious absolute risk assessments. Tr. 2465, 2477 (Nix).

285. Thus, there is substantial uncertainty in the absolute probability calculations of the Reactor Safety Study. The Board therefore finds it improper to rely heavily on the absolute probabilities in that study to determine risk. Tr. 2478 (Nix). While the Board finds these probabilities unreliable, it is noteworthy that they suggest containment overpressurization accidents are far more likely than the accident sequence resembling TMI. PWR-2 and PWR-3 accidents (discussed below), which are containment overpressurization accidents with catastrophic releases, were predicted to occur once in 125,000 reactor years and once in 250,000 reactor years, respectively. SMUD Ex. 11 at 3-4; Tr. 2479-81 (Nix); _see also_ Licensee's Findings at 194-95, para. 262. If one believes these estimates, then one must believe that a containment failure from overpressure is several orders of magnitude more likely than TMI.

286. While there is great uncertainty in the absolute probabilities given in the Reactor Safety Study, one might assume that the same or similar errors are present in the absolute probabilities for each accident sequence. If so, the relative probability of one sequence versus another may be accurate, despite the uncertainties in the absolute probabilities of each. Tr. 2478 (Nix).

287. The Reactor Safety Study classified potential accidents resulting in release of radioactivity into nine categories, designated PWR-1 through PWR-9. Nix Testimony, Table 7, at 15. Of these categories, the PWR-2 and PWR-3 sequences are releases resulting from containment overpressurization which a CFVS would be intended to mitigate. Nix Testimony at 14. Using relative risk assessment figures based on the Reactor Safety Study, these two categories comprise a large majority of the total public risk from PWR accidents. Id. The risks of PWR-2 and PWR-3 accidents comprise 65 percent of the early fatalities, 75 percent of the late fatalities, and 74 percent of the property damage associated with all postulated PWR accidents. Nix Testimony at 13-14. This suggests strongly that a CFVS could substantially reduce public health and safety risks from the operation of Rancho Seco.

288. Licensee has suggested that the errors and uncertainties may not be evenly distributed among the accident sequences, and that, therefore, the absolute probabilities may be as reliable as the relative ones.

Licensee's Findings at 192-93, para. 260. However, the Board first notes that the Licensee has the burden of showing that the errors are unevenly distributed if it wishes the Board to make this finding, and that it has presented no evidence whatsoever on the distribution of errors. More importantly, even if it were shown that the relative probabilities are no more reliable than the absolute, this hardly suggests that the absolute probabilities are reliable. The presumption that errors are sufficiently uniform among accident sequences to allow meaningful comparison is reasonable absent evidence to the contrary. But if this presumption is untrue, then, in the Board's opinion, no meaningful risk analysis exists.

289. SMUD also argues that because the likelihood of PWR-1 accidents (where containment failure results from an explosion generated missile and a CFVS could be ineffective) is highly uncertain, the relative risk of the various accident categories is unreliable. Licensee's Findings at 193, para. 261. Again, this assertion merely underscores the uncertainty in the Reactor Safety Study probabilities. Further, the uncertain likelihood of PWR-1 events does not affect the likelihood of those events under consideration here -- the PWR-2 and PWR-3 events -- for which a CFVS would provide substantial additional protection. That PWR-1 events may be more probable than predicted in the Reactor Safety Study does not suggest that efforts to protect the public from PWR-2 and PWR-3 events are unsound. As CEC witness

Nix observed: "[T]hat's a little like saying 'Don't put safety belts in the Pinto because you think there's problems with the gas tank.'" Tr. 2488 (Nix).

290. The foregoing findings compel the conclusion that not enough is known about reactor accident sequences to assess their probability of occurrence with reasonable certainty. Indeed, no witness denied these substantial uncertainties. NRC witness Meyer concurred with the testimony of CEC witness Nix, discussed above, regarding the existing poor understanding of reactor accidents:

> We certainly do not know enough detail on any reactor to accurately present all the accident sequences, their probabilities, and the consequences. . . . Tr. 2827 (Meyer).

Even SMUD witness Dieterich admitted that great uncertainty surrounds mechanisms of reactor containment failure. Tr. 2373-74 (Dieterich). The Board, therefore, finds that the probability of an accident resulting in containment failure due to overpressurization at Rancho Seco, while undoubtedly small, is largely unknown. Before taking up the consequence half of this risk equation, however, the Board further observes that it finds the debate between absolute and relative risk immaterial to the CFVS issue before us. First, as we have found, both are plagued with the same uncertainties and errors (although their distribution is such that the relative risk figures are more valid). Second, the CFVS concept is favored by both risk estimates. As discussed above, the evidence of absolute probability suggests that containment overpressurization is much more likely than the TMI accident, which would prove that such accidents are not incredible. The relative risk evidence similarly supports the CFVS concept by showing that it would mitigate nearly all of the total public risk from nuclear accidents. Thus, in the Board's opinion, the CFVS concept has merit if either or both absolute probabilities or relative risks are valid.

291. The evidence before the Board on the consequences of a containment overpressurization accident is based upon the California Energy Commission's Underground Siting Study. SMUD Ex. 11; Nix Testimony, Tables 2, 3 and 5, at 6-7, 13. This study indicates that the potential consequences of a severe reactor accident are enormous. Weighted average of the population distributions and meteorological patterns of four California reactor sites led to consequence calculations of 17-450 early deaths, 3,900-6,300 latent cancer deaths, 160-7,700 early illnesses, and 3,300-17,000 thyroid cancers. Economic consequences from evacuation, relocation, farmland interdiction, and medical treatment would total \$0.34 to \$8.60 billion. If winds were blowing towards the maximum population densities at these same four sites, the consequences would increase to 210 to 1,900 early deaths, 4,300 to 7,200 latent cancer deaths, 2,000 to 62,000 early illnesses, and 6,300 to 130,000 thyroid cancers. Economic consequences would be \$0.64 to \$36 billion. Id.

292. Moreover, the Energy Commission study reveals that the consequences of a severe reactor accident at

Rancho Seco are enormous. One of the four sites analyzed contained the characteristics of the area surrounding Rancho Seco. The weighted health effects of a severe accident at this site would be 32 early deaths, 630 early illnesses, 920 to 3,900 cancer deaths, 9,000 to 11,000 thyroid cancers, 10,000 to 13,000 thyroid nodules, 8 prenatal deaths, 620 to 2,600 genetic disorders, 200 to 840 spontaneous abortions, and 7,500 cases of temporary sterility. The associated economic consequences would be $1.3 billion. If winds were blowing towards the most densely populated sections of Sacramento, the health effects of a severe reactor accident would be 240 early deaths, 6,400 early illnesses, 3,300 to 6,100 cancer deaths, 130,000 thyroid nodules, 130,000 cancers, 130 prenatal deaths, 2,300 to 4,200 genetic disorders, 710 to 1,300 spontaneous abortions, and 100,000 cases of temporary sterility. The associated economic consequences would be $13 billion. SMUD Ex. 18 at V-29 to V-33.

293. Licensee's witness acknowledged that the Licensee has never conducted any studies comparable to the Energy Commission study. Tr. 2211-13 (Dieterich). Mr. Dieterich's testimony did not address the consequences of a containment failure accident at Rancho Seco. Dieterich CFVS Testimony, passim. Despite SMUD's lack of study of this subject, however, it nevertheless asserts that the Energy Commission's consequence figures are "unreasonable" but "not out of line with other risks, both man-made and natural, deemed

acceptable by society although not necessarily by all individuals." Licensee's Findings at 201, para. 269.

294. SMUD's assertion that the consequence figures are unreasonable rests upon its claim that the Energy Commission made improper assumptions for variables such as evacuation times, threshold doses, and dose effectiveness factors. Licensee's Findings at 201, para. 269.[45] Licensee claims that the Energy Commission's consequence estimates are "much higher" than would be the case if alternative assumptions had been made. Id. The Board disagrees that the Energy Commission's assumptions for these variables were unreasonable. But more fundamentally, the Board's examination of the Energy Commission study, reveals that, rather than assuming specific variables, it analyzed the range of consequences resulting from a broad spectrum of

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45. SMUD does not question the levels of radioactive core contaminants released. See Licensee's Findings 265-269. It merely notes that the study considered that only insoluble molecules and the core melt itslef would not be available for release. Licensee's Finding 265; SMUD Ex. 18 at V-5. SMUD also does not challenge the study's assumption that radioactive effluent is distributed evenly across a 22.5 degree sector downwind from the release point (SMUD Ex. 18 at V-11) nor its weighted averaging of six wind speed classes and seven stability categories. Id.at V-35; SMUD Ex. 11 at 7-5. It summarily dismisses, however, any worst case analysis where winds would blow towards the most dense population sector, "[b]ecause of the independent nature of the two events, the probability of their simultaneous occurrence is much lower than the already low probability of the accident sequence." Licensee's Findings 269, n. 168. However, SMUD presented no reason to doubt that this "simultaneous occurrence" was indeed the maximum credible accident it was clearly labeled to be in the Underground Siting Study. Since the Board does not know the absolute

(footnote cont. next page)

critical assumptions.[46] The important conclusion, in the Board's opinion, to be drawn from the Energy Commission's work is that the consequences of a containment failure accident are severe using <u>any</u> reasonable assumptions.

295. The most significant variables affecting accident consequences are the site of the reactor and the weather (especially wind velocity and direction). As noted previously, the Energy Commission assumed four very different sites representing reactor locations in California and one of the reactor sites considered was Rancho Seco. But to consider the sensitivity of accident consequences to critical variables, it is instructive to compare all four sites. The study presents consequences under both average and worst case weather assumptions for each site. Thus, the consequence figures range from those of an accident at a site far removed from major populations

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(footnote 45 cont.)
probability of such an accident, it cannot conclude that the coincidence with a worst case wind direction is beyond the preview of policy consideration. The uncertainty of reactor accident probabilities mandates a prudent policy that considers the full range of conceivable consequences.

46. SMUD criticizes the Underground Siting Study evacuation assumptions for calculating health effects on the basis of the "improbable extreme" of a 24-hour evacuation scenario. Licensee's Findings 265, n. 160. In fact, however, the study assumed a much more rapid evacuation:

> Four different evacuation cases were considered. Time for evacuation ranged from 1.5 to 2.4 hours and varied with distance from the reactor site. <u>The base case</u>, the results of which are reported subsequently, assumed <u>four hours</u> for evacuation from the time of containment failure. SMUD Ex. 11 at 7-5 (emphases added).

under average weather conditions to those of an accident with the wind blowing directly toward a nearby major city. While the difference between the accidents is large, the important point is that even the most optimistic assumptions result in severe consequences: 17 early deaths, 3,900 latent deaths, 160 early illnesses, and 3,300 long-term thyroid cancers. The Board finds even these most optimistic consequences to be very substantial.

296. In summary, then, the probability of an over-pressurization of containment leading to an offsite release of radionuclides at nuclear reactors is small, but how small is not at all certain. The consequences of such an event are, however, certain to be severe under the most optimistic assumptions, and can potentially be catastrophic. The Licensee suggests that the overall public risk from these events is comparable to that of a dam failure or a major earthquake -- risks that society accepts. Licensee's Findings at 201, para. 269, citing CEC Ex. 11 at 7-10, 7-11,[47] and Tr. 2539-43 (Nix). The Board does not share Licensee's sanguine acceptance of this risk. First, the Board notes that the more severe consequence estimates predict impacts greater than those of dam failures or earthquakes. Nix Testimony at 6-7. Given the uncertainties in weather, evacuation, and health effects of radiation it is

47. This exhibit was not received into evidence. The Board believes SMUD's reference is to its own Exhibit 11.

prudent to consider the full range of possible consequences rather than the most optimistic. Nix Testimony at 6. More importantly, the Board rejects the implication that society accepts such risks when, as here, there are potential alternatives or practical methods of avoiding the risk. Simply put, there is no analog for a CFVS applicable to earthquakes or dam failures. Therefore, the critical issue in our view is not whether the risk is comparable to others which society has little choice but to accept. The issue here, instead, is whether the risk is sufficiently great to warrant careful study of a system that promises to reduce or eliminate it. The Board finds that it is.

297. Additionally, the Board wishes to point out that throughout the preceding findings on probabilities we have considered the probability of containment failure accidents in pressurized water reactors generally. This is because the evidence presented to us on this issue was largely based on the Reactor Safety Study. But having found that Rancho Seco is especially sensitive to feedwater transients compared to other pressurized water reactors because of its B&W design, we are constrained to add that the risk of containment overpressurization there is, logically, somewhat higher than at other facilities. Furthermore, the Board also has found that the OTSG sensitivity places a greater burden on operators who, at Rancho Seco, have not been given substantially better instruments, controls, and training to cope with the sensitivity. This logically increases the probability

of operator error at Rancho Seco. Most important in the Board's mind, however, is that these factors illustrate the inherent uncertainty and illogic of predicting absolute accident probabilities at a given reactor. It is this large uncertainty which most strongly supports our conclusion on this issue.

(iii) Effectiveness of CFVS

298. The idea of a CFVS is not new; the concept has received considerable attention since 1975. Meyer Testimony at 5; Nix Testimony at 15. The California Energy Commission Underground Siting Study is, as the Board has already noted, one of the most comprehensive studies of these systems. That study included conceptual designs of a CFVS developed by several well known engineering firms. The system effectiveness was evaluated by the Aerospace Corporation, Advanced Research and Applications Corporation, and Intermountain Technology, Inc. Nix Testimony at 16. As will be seen, their conclusion, as summarized in the Energy Commission study, was that a CFVS can be made to be extremely effective.

299. But the Energy Commission study is not the only technical review of the CFVS concept. Sandia Laboratories also developed a conceptual design of a CFVS for use at TMI-2 in the event of continued core melting. Nix Testimony at 16. Norwegian and Swedish studies on underground siting have considered the use of soil and rock as a filtering medium, and a UCLA s udy group has developed a conceptual

design of a more sophisticated filter involving sand, gravel, and charcoal. Meyer Testimony at 16.

300. The concept of CFVS is not untried, either. Various types of such systems have been or are being implemented in Fast Breeder Reactor facilities here and abroad. Meyer Testimony at 4. The Zero-Power Plutonium Reactor Test Facility, the Fast Flux Test Facility, and the German SNR-300 prototype LMFBR all have or are installing CFV's. Meyer Testimony at 4-5. A somewhat similar concept has also been employed at some of the Canadian multi-unit CANDU reactors. Id. In short, there is considerable information available on the CFVS concept and the technical capability exists to design and implement it. Nix Testimony at 16; Meyer Testimony at 6.

301. As this on-going activity suggests, a CFVS can effectively mitigate the release of radionuclides from an overpressurized containment. The consensus of CEC witness Nix and NRC witness Meyer, the two most expert witnesses testifying on this issue, was that a CFVS can do more than reduce exposures from such an event. Both witnesses agreed that a CFVS can be designed to effectively eliminate exposures for all practical purposes. Dr. Meyer testified that "systems can be designed and implemented which can vent large volumes of gases and vapors in a controlled manner and which can attenuate (absorb) virtually any radioactive isotope known to be harmful." Meyer Testimony at 5-6. Mr. Nix noted that charcoal was added to the

Energy Commission filter design "when analysis found that the only significant atmospheric release product was radioactive organic iodine." Charcoal can effectively remove iodine. Nix Testimony at 11; Meyer Testimony at 2. Dr. Meyer summarized the effectiveness of various filters by saying: "For all designs, the attenuation factors for particulates and molecular iodine are better than 98%." Meyer Testimony at 2.

302. The most revealing testimony on the effectiveness of these systems, however, was a table presented by Mr. Nix comparing the consequences to the public of uncontrolled releases versus controlled, filtered releases. Using the same assumptions that the Licensee has argued result in unreasonably high consequences (<u>see</u> Findings 293-294, <u>infra</u>); the consequences of controlled, filtered releases from containment were shown to be virtually neglible. Under average weather conditions, deaths were prevented completely and thyroid cancers were reduced from many thousand to only a very few. Nix Testimony, Table 5, at 13.

303. The Board notes this data corresponds well to the ability of the filter to all but eliminate radioactive releases set forth in the testimony of Mr. Nix and Dr. Meyer, as well as the considerable increase in evacuation time resulting from the controlled release. Tr. 2812-13 (Meyer). These dramatically smaller impacts also confirmed the testimony of Dr. Meyer that:

> [W]hatever the final choice, the Filtered Vent Containment System (FVCS) will result in a considerable reduction in societal risk relative to an uncontrolled, unfiltered containment failure. Meyer Testimony at 2.

Mr. Nix concurred, stating:

> CFV is extremely effective in reducing public risks from containment overpressurization failure accidents. Nix Testimony at 14.

304. This impressive effectiveness of CFVS was virtually uncontested by the witnesses in the proceeding.[48] Although Mr. Dieterich raised certain design concerns regarding a CFVS (which we discuss below), he did not dispute the ability of the various filter materials to absorb radionuclides. Indeed, Mr. Dieterich conceded he would prefer controlled filtered releases to uncontrolled ones and agreed that a CFVS would result in a substantial difference in health impacts if the system operated properly. Tr. 2260, 2265-2266, 2299-2300 (Dieterich).

305. The Licensee suggests in its proposed findings, however, that a CFVS would be ineffective because it would not absorb noble gases such as xenon or krypton. Licensee's Findings at 209-210, para. 278. The Board finds Licensee's assertion that this is a "very significant

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48. Dr. Meyer mentioned that two days before his cross-examination he had heard that a filter composed of one-inch pieces of gravel had been tested in Sweden and had shown rather discouraging decontamination results. Tr. 2881 (Meyer). This news did not change his opinion that a CFVS could be designed that would be extremely effective in attenuating the release of radioactivity from containment. Tr. 2908 (Meyer).

concern" unsupported by the evidence. As CEC witness Nix explained, these gases, while not absorbed, are dispersed by the filter so slowly that the release approximates background levels. Tr. 2659-2664 (Nix). Moreover, Mr. Nix also pointed out that kryogenic filters might be installed which would remove even the noble gases, though he felt the need to do so was questionable. Id.

306. The Licensee in its proposed findings also suggested that the high attenuation factors predicted in the Energy Commission study could not be achieved at Rancho Seco. The basis of this assertion was that the Energy Commission study assumed an underground release while Rancho Seco might employ an above-ground filter. Licensee's Findings at 211-212, para. 281. Since filter size contributes to the attenuation of xenon and krypton, a surface filter would have to approximate the size of the underground filter to achieve the same attentuation of these noble gases. Tr. 2718-19 (Nix). CEC witness Nix therefore believed it would be very important to go through an engineering design and balance the filter size and the release one might tolerate from the system. Tr. 2736 (Nix). The Board notes that this type of balance would logically consider such site specific parameters as the available room for a large filter and the need to protect local populations from dispersed noble gases. Thus, it appears to the Board that the need for this balancing supports a site specific design study at Rancho Seco.

307. In sum, while there are certainly design choices such as filter location, composition, and size to be made, the Board concludes that it is reasonably certain that a CFVS can be designed for Rancho Seco that will very effectively control and filter releases of radionuclides.[49]

308. Although Mr. Dieterich's testimony did not contest the effectivness of CFVS in mitigating radionuclide releases, he did express concern that a CFVS would interfere with other plant safety systems, increase the likelihood of hydrogen ignition, reduce plume rise, and potentially actuate when containment would not have failed. Dieterich CFVS Testimony at 5-7.

309. The Board views the issue of unnecessary actuation of a CFVS, that is, actuation where containment would have otherwise been maintained, to be more apparent than real. We have already examined the issue of rupture disk reliability and found it quite high. <u>See</u> Finding 274, <u>infra</u>. Thus, the potential for premature actuation is limited and this issue rather quickly distills to judgments regarding the best set point for a CFVS. <u>E.g.</u> Tr. 2825-30 (Meyer). The Board recognizes that a balance must be struck between preventing containment failure and allowing unnecessary releases through the CFVS. <u>Id</u>. But we are convinced, such a judgment

49. The Board notes in this regard that a filtering system proposed by Sandia in a recent study produced attentuation factors for particulates and elemental iodine greater than 98%. Tr. 2905-2906 (Meyer); Meyer Testimony at 2.

can be made (indeed, it has been made at various test reactors; see Meyer Testimony at 4-5). The Board believes this balance further suggests need for a site specific design study. The Board also notes, however, that the very high effectiveness of the filter would largely mitigate any unnecessary releases, suggesting that the balance should favor actuation of a CFVS over increased risk of containment failure.

310. Mr. Dieterich testified that a CFV system could worsen a primary system break by reducing containment pressure and increasing leakage of coolant. Dieterich CFVS Testimony at 5; Tr. 2821-22 (Meyer). On cross-examination, however, he admitted that the significance of this effect had not been studied. Tr. 2255 (Dieterich). Moreover, he stated that loss of coolant calculations that did incorporate building backpressure assumed pressurized containment, not over-pressurized containment. Tr. 2255 (Dieterich).

311. NRC witness Meyer similarly testified that, in the event of a double ended pipe rupture accident, where the ECCS would be reflooding the core, the backpressure in containment would increase the rate of heat transfer to the reflood coolant. Tr. 2825 (Meyer). He also noted that steam binding in the remaining portions of the primary loop would be less severe with high containment backpressure. Id. He suggested that a CFVS, by reducing backpressure, would undermine the effectiveness of reflooding and increase steam

binding problems. Id. However, he conceded that these problems would only occur if the CFV system setpoint were below 59 psig. Tr. 2824 (Meyer). As we have found already, however, the setpoint would not be at or below design basis pressures. See Finding 277, infra.

312. Moreover, the CFVS would not open until an accident exceeding the design basis accident occurred. Tr. 2720 (Nix). At that point other safety systems would have failed, the reactor would be essentially out of control, and the existence of backpressure would be irrelevant to bringing the reactor under control. Tr. 2721 (Nix). Finally, the Board notes that depressurization could result from either a CFVS activation or a breach of containment. Tr. 2256 (Dieterich); Tr. 2835 (Meyer). Thus, the CFVS would not create a risk not already present. Given a system failure creating an overpressurization and a risk of breach of containment, even SMUD witness Dieterich conceded that a controlled release would be preferable to an uncontrolled breach of containment. Tr. 2260 (Dieterich).

313. Mr. Dieterich also testified that a CFV system depressurization could flash containment sump water and result in cavitation of reactor building spray pumps and low pressure injection pumps. Dieterich CFVS Testimony at 5; Tr. 2824-2825 (Meyer). However, depressurization and flash damages could result from either CFV actuation or breach of containment. Tr. 2260 (Dieterich); Tr. 2825 (Meyer). Again, the CFV system would not create a risk not already

present. Moreover, once containment pressurization exceeded 59 psig, the reactor building spray pumps and the low pressure injection pumps would have failed their major purposes -- to prevent containment overpressurization and radioactive releases into the environment. The primary objective should be to avoid overpressurization of containment and not merely maintain the pumps. Tr. 2721 (Nix). Given a system failure and a risk of breach of containment, Mr. Dieterich again conceded the desirability of controlling a radioactive release and rendering pumps inoperable over having an uncontrolled release with operable pumps. Tr. 2265, 2266 (Dieterich).

314. Mr. Dieterich further testified that delayed spray systems could cause a temporary overpressurization of containment, leading to "unnecessary releases of radioactivity through the vent system." Dieterich CFVS Testimony at 6. However, these releases would be filtered before being released, minimizing public hazards. Tr. 2722-2723 (Nix). Furthermore, Mr. Dieterich could not guarantee that the delayed pumps would be restored or that containment would remain unbreached. Instead, he could no more than "hope that the operator could get a spray system initiated to turn that transient around, or at least stop it." Tr. 2268 (Dieterich). Again, once overpressurization above the design basic accident results, whether temporary or permanent, the primary safety concern ought to be to prevent an unfiltered breach of containment. Tr. 2722 (Nix).

315. Mr. Dieterich testified that a CFV system might cause an increase in hydrogen concentration relative to air and an explosive ignition. Dieterich CFVS Testimony at 6. The basis of his assertion was Section 5.3 of the Sandia study (CEC Ex. 32), which assumed that the core was melted and generating hydrogen by decomposing the containment building floor. Tr. 2273, 2274 (Dieterich). Once an accident were to go this far, however, a hydrogen ignition would be of much less concern than an imminent breach of containment through the building floor. On cross-examination, SMUD witness Dieterich admitted the following basis for his special concern regarding hydrogen ignition: "I don't think destruction of that concrete in the base mat is of any significant concern to the health and safety of the public." Tr. 2275 (Dieterich). The Board does not share this judgment that a molten core decomposing the containment floor is of no public concern. Indeed, this sort of accident is precisely that for which a CFVS might prevent large public exposures.

316. SMUD witness Dieterich further testified that an "area of concern" regarding CFV was the possibility of hydrogen explosions in the vent line itself. Dieterich CFVS Testimony at 6. On cross-examination, however, he admitted "a very strong probability" that ignition sources within the containment structure would lead to a burning of hydrogen before an explosive concentration could occur in the vent line. Tr. 2278 (Dieterich). Additionally, he admitted that he was unaware of any ignition sources in the vent line. Tr. 2277 (Dieterich).

317. Lastly, Mr. Dieterich testified that a CFVS might worsen an accident's effects by reducing the temperature and buoyance of a plume, causing greater radiation contaminations over smaller areas. Dieterich CFVS Testimony at 6. On cross-examination, however, he was unable to find this conclusion in his reference (Sandia Study, section 5.4). Tr. 2278 (Dieterich). Moreover, he had "no strong feeling" for the significance of this effect. Tr. 2280 (Dieterich). Mr. Dieterich could not even vouch that the effect would be "adverse"; he merely asserted that it needed further study. Id.

318. In summary, then, the Board finds that the CFVS concept has been studied and even applied for several years. The result of these studies and applications has been the development of extremely effective filtering media and design techniques which, taken together, provide reasonable assurance that a CFVS can be designed to very effectively prevent radionuclides from being released to the environment from containment overpressurization accidents. The Board recognizes that a number of design issues would have to be resolved before a CFVS could be implemented at Rancho Seco, but we find no design issues that suggest a site specific design study is not appropriate.

(iv) <u>The Need for Study in Addition to Existing Commission Actions.</u>

319. The Licensee's last argument in opposition to a CFVS design study at Rancho Seco is that such an effort would be inappropriate because the Commission is studying this concept generically. SMUD suggests that a site specific study at Rancho Seco is unnecessary in light of these other efforts. <u>See</u> Licensee's Findings at 217-220, para. 288-290.

320. It is true that the Commission has, since the accident at TMI, accelerated its examination of methods of mitigating the consequences of core melt accidents, including CFVS. Among these actions is a Staff recommended rulemaking on this subject. Meyer Testimony at 7; Dieterich CFVS Testimony at 7. This recommendation envisions a broad inquiry into core melt and core degradation accidents applicable to all operating Licensee's, including SMUD. Tr. 2841, 2893-2896 (Meyer). The Commission has not acted on the Staff recommendation. Meyer Testimony at 8. Moreover, even if the Commission does act to adopt it, the rulemaking will not conclude for several years. Tr. 2840-2841 (Meyer).

321. As we have already seen, there are several site and facility specific issues that must be resolved before the merit of implementing a CFVS at Rancho Seco can be determined. Tr. 2838 (Meyer). The proposed rulemaking, if it occurs, will consider industry wide guidelines, but it will not be a specific plant by plant analysis. Tr. 2892-2896 (Meyer).

322. The NRC Staff is conducting an Interim Reliability Evaluation Program ("IREP") which includes a probabilistic analysis of individual reactors similar to the Reactor Safety Study. Tr. 2840 (Meyer); Staff Ex. 4 at 6-1 to 6-4. Dr. Meyer testified that the site specific factors necessary to implement whatever results from the rulemaking will be factored in through the IREP program. At the present time, however, it is unknown when the IREP program will examine Rancho Seco. Tr. 2899 (Meyer). For this reason, as well as the length of the proposed rulemaking discussed above, it is apparent to the Board that it will be several years before a decision on a CFVS for Rancho Seco can be made unless additional efforts are undertaken by the Licensee.

323. There is no reason why SMUD could not go forward and study the design, costs, and general feasibility of a CFVS rather than awaiting the recommended rulemaking. Dr. Meyer testified, for example, that the data necessary to do an IREP-type analysis exists at every reactor, and there is no physical or technological reason why SMUD could not perform such a study. Tr. 2899 (Meyer).

324. The NRC Staff is, in fact, studying the expedited implementation of CFV systems at two operating reactors, Indian Point 3 and Zion. Tr. 2888, 2897 (Meyer). Because these reactors are very near New York City and Chicago, the Staff believes that they pose a higher public risk than other facilities, and therefore a CFVS merits special consideration at these facilities. Tr. 2897-2899 (Meyer).

Accordingly, the Staff is not awaiting the proposed rule-making to determine whether to implement a CFVS at these two reactors. Id. In fact, the Staff expects to decide on whether such a system should be installed there by the end of this year. Id.

325. The Board, having carefully considered not only the CFVS concept but the overall ability of the Rancho Seco system to safely respond to feedwater transients, strongly believes that consideration of a CFVS for Rancho Seco should also be expedited. Rancho Seco is located near two population centers, albeit not of the size of New York City or Chicago, and is also in a turbulent weather zone within a large, highly productive agricultural area. Tr. 2693 (Nix). This alone, in our view would not place Rancho Seco in the class of Indian Point 3 or Zion with regard to public risk. But these facts, together with the safety concerns arising from the sensitivity of the OTSG, which exist at Rancho Seco but not at Indian Point 3 or Zion, causes us to conclude that Rancho Seco joins these reactors in presenting an unusually high public risk. As Dr. Meyer testified, it is consistant with the Staff's rationale for studying CFVS at Indian Point 3 and Zion to also give expeditious treatment to a facility that, although it may have a somewhat lower surrounding population, has a somewhat higher probability of an accident. Tr. 2899 (Meyer). As we have found throughout this decision, Rancho Seco is such a facility.

326. In answer to CEC Issue 5-2, there, the Board concludes that the Licensee should, in cooperation with the Commission Staff, perform a site and facility specific study intended to develop a proposed design for a CFVS at Rancho Seco, as well as the projected costs and performance characteristics of the proposed system. This study shall be completed no later than one year from the effective date of this decision.

O. Concluding Findings of Fact

327. The foregoing findings represent the vast majority of the necessary conclusions on the issues presented in this proceeding. We deem it appropriate, however, to address the question of the adequacy of the May 7 Order. While no issue expressly challenges that Order's adequacy, one evident purpose of this proceeding has been to determine whether the short and long-term modifications were sufficient to ensure safe operation of the facility. There was extensive examination relating to the adequacy of that Order which permits us to enter these findings.

328. On the whole, we find that the May 7 Order was not adequate to ensure safe Rancho Seco operation. First, with respect to both the short and long-term items, there was no in depth study of possible actions which were necessary to ensure safe operation of Rancho Seco. Rather, in an effort to avoid shutdown without restart criteria, the items selected for the May 7 Order were devised almost overnight. See Section II.

329. In particular, the May 7 Order was most deficient in failing to require in depth ICS and AFW analyses prior to facility restart. We are persuaded that when particular systems are identified as potentially contributing to dangerous conditions as were the ICS and AFW [CEC Ex. 26], it is inadequate to provide for restart without

first carefully analyzing those systems. Yet, that is what occurred with respect to Rancho Seco and this renders the May 7 Order inadequate, at least with respect to the short-term requirements.

330. The inadequacy of the May 7 Order is underscored by the small break LOCA analyses which were a precondition to facility restart. Although the NRC Staff certified these analyses complete in late June, 1979, only a month later the Staff issued I&E Bulletin 79-05C, which found at least a portion of the B&W analyses totally invalid. Clearly, the short-term actions, as implemented, were not adequate.

331. In the longer term, the same criticisms can be made. The long-term items contained in the May 7 Order were not carefully studied prior to selection. However, with respect to the long-term items, the NRC provided a "moving target" by indicating that parties would be free to allege that more long-tem modifications were necessary. See NRC June 21, 1979 Order. As is clear from our findings herein, we have ruled that additional items should be accomplished as part of Licensee's, post-TMI efforts. many of these items would likely have been included in the May 7 Order if sufficient analyses had been accomplished, prior to formulating the May 7 Order. However, we deem the NRC's intent satisified by ordering, as specified hereafter, that the additional items be accomplished according to the time table set forth in the Conclusions of Law.

CONCLUSIONS OF LAW

1. Based upon the documentary evidence presented in this proceeding and in accordance with the foregoing findings of fact, this Board enters the following conclusions of law.

2. The Licensee has satisfactorily completed the short-term actions required by subparagraphs (a) through (e) of Section IV of the May 7 Order. These actions, however, were not sufficient to provide reasonable assurance that the facility would respond safely to feedwater transients, pending completion of the long-term modifications set forth in Section II of the May 7 Order.

3. The long-term modifications set forth in Section II of the NRC's May 7 Order pertaining to the auxiliary feed-water system and to the integrated control system have not been completed satisfactorily. They should promptly be completed in accordance with paragraph 5 of these conclusions. The remaining long-term requirements set forth in the May 7 Order have been or are being satisfactorily completed. Additionally, the Licensee has performed or committed to perform certain other actions ordered by the NRC.

4. The actions which the Licensee has performed or committed to perform in the near future provide a reasonable assurance that the facility will safely respond to feedwater transients while the measures set forth in the subsequent conclusion are completed.

5. The following actions will provide substantial additional protection necessary to assure that Rancho Seco will continue to safely respond to feedwater transients over the remaining life of the facility. Therefore, in accordance with the schedule and other requirements specified in conclusion 6, the Licensee shall accomplish the following:[50]

(a) The Licensee shall investigate or actively participate with others to investigate additional methods to reduce or eliminate the close coupling of the primary and secondary systems induced by the OTSG. Licensee shall determine the scope and direction of its investigation within 60 days of this initial decisions and shall complete its investigation and determine the actions, if any, it plans to take as a result of this investigation no later than one year after the decision.

(b) The Licensee shall upgrade its failure modes and effects analysis of the ICS to include related systems described in Finding 62 and determine what actions, if any, it proposes to take based upon that study.

(c) The Licensee shall revise and upgrade its AFW reliability study in accordance with the comments of the NRC Staff set forth in CEC Exhibit 21 no later than

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50. Where no time is set forth below for accomplishment of a task, the Licensee shall accomplish the same no later than 6 months after this initial decision.

six months from the date of this initial decision. By the same date, SMUD shall also determine what actions, if any, it will take based upon the results of that study to improve AFW reliability. In the event the revised study shows Rancho Seco's AFW system to be substantially less reliable than the comparable Westinghouse system for any time sequence, SMUD shall institute remedial actions to make the Rancho Seco AFW system more reliable than the Westinghouse system.

(d) The Licensee shall develop, for the NRC Staff's approval, operator procedures for a loss of all AC power event that include procedures to ensure timely AFW operation as described in CEC Ex. 21.

(e) The Licensee shall demonstrate that the B&W small break LOCA analysis described in NRC Ex. 2 fulfills the requirements of 10 C.F.R. 50.46.

(f) The Licensee shall review and revise, as necessary, its procedures for reflux boiling and feed and bleed cooling as well as provide every operator with specific instruction, including simulator training, on both cooling methods.

(g) Licensee shall develop procedures and supporting analysis which allow operators to consider subcooling in determining whether to trip reactor coolant pumps on low RCS pressure. The procedures and analysis shall be submitted to the NRC Staff and, if approved, become effective.

(h) Licensee shall, within one year of this decision, investigate or participate with others to investigate means of improving HPI performance to avoid the necessity of tripping the coolant pumps as required by I&E Bull. 79-05C.

(i) Licensee shall revise its operating crew assignments and operator hours as necessary to ensure that each operator is given regular time during working hours, but while not standing watch, to participate in training. Licensee shall submit a schedule for implementing this requirement no later than 4 months from this decision.

(j) Licensee shall investigate the costs and relative merit of installing a simulator at Rancho Seco or providing operators with four days simulator training every four months at the B&W simulator. Licensee shall elect and implement one of these options unless it submits an alternative plan for substantially upgrading operator training that is approved by this Board or its delegate.

(k) Licensee shall implement the recommendations of the Commission's Performance Appraisal Branch.

(m) Licensee shall revise procedures for unlicensed operator training to ensure that there is instruction of these personnel before they are requested to perform emergency actions relating to safety systems.

(n) Licensee shall provide all information received from the Institute for Nuclear Power Operations, reactor vendors, and the Commission or its Staff regarding the operating experience of other reactors to all shift supervisors on a regular basis. Licensee shall develop written criteria to guide distribution of materials relevant to the operation of Rancho Seco.

(o) Licensee shall revise its emergency procedures in accordance with findings 209 through 211 of this decision.

(p) Licensee shall within one year investigate and submit to this Board or its delegate a proposal for installing core level indication, wide range pressurizer indication, and natural circulation flow meters.

(q) Licensee shall ensure that a reliable hydrogen recombiner is available at Rancho Seco, as provided in finding 253.

(r) Licensee shall investigate and propose to the NRC design changes for making the PORV safety grade, as described in finding 97.

(s) The Licensee shall perform and submit to the Board or its delegate a design specific feasibility study of a controlled, filtered vent system for the containment, including cost estimates and proposed implementation schedules within one year of the effective date of this decision.

6. With respect to each action required under the preceeding paragraph, Licensee shall by the date for performance of each action, advise the Board, the NRC Staff and the California Energy Commission of the status of actions required to be taken. Within 60 days of such notification by Licensee, any participant or the Board on

its own motion may request (or, in the case of the Board, Order) that this proceeding be reconvened to determine the adequacy of SMUD's compliance.

7. The Board may, at its discretion and to the extent permitted by law, delegate review of Licensee's compliance with this decision to an arbitration panel comprised of one member of the Commission Staff, selected by the Board, member selected by the California Energy Commission, and one member selected by the Licensee. Orders of such a panel shall be deemed orders of the Board.

ORDER

WHEREFORE, IT IS ORDERED, in accordance with 10 C.F.R. §§ 2.760(a) and 2.762, that this Initial Decision shall constitute the final action of the Commission thirty (30) days after the date of issuance hereof, unless exceptions are taken in accordance with section 2.762 or the Commission directs that the record be certified to it for final decision. Any exceptions to this Initial Decision or designated portions thereof must be filed within ten (10) days after service of the decision. A brief in support of the exceptions must be filed within thirty (30) days thereafter (forty (40) days in the case of the NRC Staff). Within thirty (30) days of the filing and service of the brief of the appellant (forty (40) days in the case of the NRC Staff), any other party may file a brief in support of, or in opposition to, the exception.

IT IS SO ORDERED.

Respectfully submitted,

Original signed by

CHRISTOPHER ELLISON

LAWRENCE COE LANPHER

Attorneys for the California Energy Commission

DATED: August 4, 1980