UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

LICENSEE'S PROPOSED FINDINGS OF FACT AND CONCLUSIONS OF LAW IN THE FORM OF AN INITIAL DECISION

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July 11, 1980

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I. INTRODUCTION AND BACKGROUND

A. History of the Case

1. The Sacramento Municipal Utility District ("Licensee," "SMUD," or "the District") is the holder of Facility Operating License No. DPR-54, issued in 1974, which authorizes the operation of the Rancho Seco Nuclear Generating Station, located at Licensee's site in Sacramento County, California. The facility, which includes a Babcock & Wilcox ("B&W") designed pressurized water reactor ("PWR"), is authorized to operate at steady state power levels not in excess of 2772 megawatts thermal (rated power). This proceeding arises

out of an Order in this docket issued by the Commission on May 7, 1979. The Order directed Licensee to take certain short-term actions and to implement certain long-term modifications at the Rancho Seco plant. Sacramento Municipal Utility District (Rancho Seco Nuclear Generating Station), Commission Order, Docket No. 50-312 (May 7, 1979), 44 Fed. Reg. 27779 (1979).[1]

2. In the course of its early evaluation of the March 28, 1979 accident at the Three Mile Island Unit No. 2 ("TMI-2") facility, which utilizes a B&W designed pressurized water reactor, the NRC Staff expressed concern that B&W designed reactors appeared to be unusually sensitive to certain off-normal transient conditions originating in the secondary system. Because of certain design features, B&W designed reactors were viewed as placing more reliance, than do other PWR designs, on the reliability and performance characteristics of the auxiliary feedwater system, the integrated control system, and emergency core cooling system to recover from frequent anticipated transients. This, in turn, was viewed as placing a large burden on plant operators in the event of off-normal system behavior during such anticipated transients. Commission Order of May 7, 1979, 44 Fed. Reg. at 27779.

3. As a result of its preliminary review of the TMI-2 accident chronology, the NRC Staff initially identified

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1 This Order will be referred to throughout this decision as "the May 7 Order" or "Commission Order of May 7, 1979."

several human errors that occurred during the accident and contributed significantly to its severity. All operating licensees subsequently were instructed to take a number of immediate actions to avoid repetition of these errors, in accordance with bulletins issued by the Commission's Office of Inspection and Enforcement. In addition, the NRC Staff began an immediate review of the design features of B&W reactors to determine whether additional safety corrections or improvements were necessary with respect to these reactors. Id.

4. The NRC Staff's evaluation identified design features which indicated that B&W designed reactors are unusually sensitive to certain off-normal transient conditions originating in the secondary system. As a result, an additional bulletin was issued by the Office of Inspection and Enforcement which instructed operating licensees with B&W designed reactors to take further actions.[2] The NRC Staff also identified certain other safety concerns that warranted additional short-term design and procedural changes at operating facilities having B&W designed reactors. Commission Order of May 7, 1979, 44 Fed. Reg. at 27779.

5. Following a series of discussions between Licensee and the NRC Staff concerning possible design modifications and changes in operating procedures, Licensee agreed, in

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2 The requirements of IE Bulletins 79-05, 79-05A and 79-05B, each of which was issued in April, 1979, are summarized in Staff Exhibit 4 at 3-1, 3-2, and A-1 to A-6.

a letter of April 27, 1979 (CEC Ex. 25), to perform promptly the following actions:

(a) Upgrade the timeliness and reliability of delivery from the Auxiliary Feedwater System by carrying out actions as identified in Enclosure 1 of the licensee's letter of April 27, 1979.

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

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

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

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

Licensee also stated that Rancho Seco would be shut down on April 28, 1979, which it was, and would remain shut down until these short-term actions were completed. Commission Order of May 7, 1979, 44 Fed. Reg. at 27779.

6. In its Order of May 7, 1979, the Commission concluded that the prompt actions set forth as (a) through (e) above were necessary to provide added reliability to the reactor system to respond safely to feedwater transients, and found that operation of Rancho Seco should not be resumed until these actions have been completed satisfactorily. Consequently, the Commission ordered Licensee to take those actions

and to maintain Rancho Seco in a shutdown condition until they were completed satisfactorily. Satisfactory completion was stated to require confirmation by the Director, Office of Nuclear Reactor Regulation, that the actions specified have been taken, the specified analyses are acceptable, and the specified implementing procedures are appropriate. The Commission also found that the public health, safety and interest required that the Order be effective immediately. Id. at 27779, 27780.

7. In addition to these modifications to be implemented promptly, Licensee proposed to carry out certain long-term modifications to further enhance the capability and reliability of the reactor to respond to various transient events. These are:

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

> 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.

In its Order of May 7, 1979, the Commission directed Licensee to accomplish these long-term modifications as promptly as practicable.[3] Id.

8. Finally, in its Order of May 7, 1979, the Commission provided as follows:

> Within twenty (20) days of the date of this Order, the licensee or any person whose interest may be affected by this Order may request a hearing with respect to this Order. Any such request shall not stay the immediate effectiveness of this Order.

Id. at 27780.

9. In response to the May 7 Order, two joint hearing requests were filed: one by Friends of the Earth, Environmental Council of Sacramento and Original SMUD Ratepayers Association (collectively "FOE"); and one by Gary Hursh and Richard D. Castro, two of the five elected members of the District's Board of Directors. In an Order issued on June 21, 1979, the Commission directed the Chairman of the Atomic

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3 Similar orders were issued to the other operating licensees with reactors designed by B&W. See, generally, Staff Ex. 4 at 3-2 to 3-4, and A-7 to A-14. As we will have occasion to observe later, however, there are significant differences between the Rancho Seco order and the order issued by the Commission for the Three Mile Island Unit 1 facility.

Safety and Licensing Board Panel to select a board to determine whether the requesters meet the requisite personal interest test and to conduct any hearing which may be required. The Commission specified the subjects to be considered at the hearing and confirmed that resumed operation of the Rancho Seco facility on terms consistent with the Order of May 7, 1979, was not stayed by the pendency of these proceedings. Sacramento Municipal Utility District (Rancho Seco Nuclear Generating Station), CLI-79-7, 9 N.R.C. 680 (1979), motion to stay denied, Friends of the Earth, Inc. v. United States, 600 F.2d 753 (9th Cir. 1979), pet. review pending.

10. On June 22, 1979, the Chairman of the Atomic Safety and Licensing Board Panel established this Atomic Safety and Licensing Board ("the Board") to rule on petitions for leave to intervene and/or requests for hearing and to preside over the proceeding in the event that a hearing is ordered. 44 Fed. Reg. 37702 (1979). On November 20, 1979, the Board was reconstituted to substitute the current Chairman in place of the original Chairman, who was unable to continue his service on the Board. 44 Fed. Reg. 69063 (1979).

11. On June 27, 1979, the Commission's Office of Nuclear Reactor Regulation issued its Evaluation of Licensee's Compliance with the NRC Order dated May 7, 1979 (following Tr. 362; hereafter "Staff Evaluation"), in which it concluded that Licensee had satisfactorily completed the actions prescribed in items (a) through (e) of paragraph (1) of Section IV of the May

7 Order, the specified analyses are acceptable, and the specified implementing procedures are appropriate. On the same day, Licensee was authorized to resume operation of Rancho Seco. Authorization to Resume Operation, 44 Fed. Reg. 40459 (1979).

12. By its Order for Filing of Amended and Supplemental Requests for [Hearing] and Notice of Prehearing Conference, dated July 3, 1979, the Board directed the parties requesting a hearing to amend their petitions, pursuant to 10 C.F.R. § 2.714, to set forth their interests specifically, and invited the parties to supplement their petitions with specific contentions and the bases therefore.

13. On July 17, 1979, the California State Energy Resources Conservation and Development Commission ("California Energy Commission" or "CEC") filed notice of its intent (treated by the Board as a request) to participate on behalf of the State of California as an interested state pursuant to 10 C.F.R. § 2.715(c).

14. A prehearing conference was held, pursuant to 10 C.F.R. § 2.751a, on August 1, 1979, to consider the requests for hearing and amended petitions. In its subsequent Prehearing Conference Order of August 3, 1979, the Board admitted FOE and Messrs. Hursh and Castro as intervening parties pursuant to 10 C.F.R. § 2.714, and granted interested state status under 10 C.F.R. § 2.715(c) to the California Energy Commission.

15. In its Order Ruling on Scope and Contentions, dated October 5, 1979, the Board ruled on the admissibility for adjudication of the specific issues and contentions put forward by CEC and the Intervenors. On the same day, the Board issued its Notice of Hearing.

16. A second prehearing conference was held on February 6, 1980, to consider outstanding motions and a schedule for the evidentiary hearings. In their opening statements at the conference, intervenors Hursh and Castro announced their withdrawal from the proceeding. In its Order Subsequent to the Prehearing Conference of February 6, 1980, dated February 14, 1980, the Board viewed the statement of Messrs. Hursh and Castro as notification of their voluntary default under 10 C.F.R. § 2.707, and dismissed them from the proceeding. Subsequent to the prehearing conference, but prior to the commencement of the evidentiary hearing, counsel for Friends of the Earth, the Environmental Council of Sacramento and Original SMUD Ratepayers Association notified the Board by letter of February 19, 1980, of the withdrawal of those intervenors from the proceeding.[4]

17. Pursuant to the Board's Order Scheduling an Evidentiary Hearing, dated January 24, 1980, 45 Fed. Reg. 7356

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4 While the Board had not received the letter prior to the start of the hearing, a representative of FOE announced the withdrawal as a part of his limited appearance statement. Tr. 170-171.

(1980), the evidentiary hearing commenced on February 26, 1980, in Sacramento, California, with Licensee and the NRC Staff appearing as parties, and the California Energy Commission as a representative of an interested state. The hearing sessions on February 26 (including an evening session) and 27 were devoted to limited appearance statements presented by interested members of the public pursuant to 10 C.F.R. § 2.715(a). Sessions of the hearing to receive sworn testimony were held in Sacramento on February 28, March 3 through 6, April 8 through 11 and 14 through 17, May 6 through 10 and 12 through 14, 1980.

B. Scope of the Proceeding, Allocation of Burdens and Summary of Issues Presented

18. In the experience of the members of this Board, this is a unique administrative proceeding. Licensee has not applied for an amendment to its operating license. Instead, the Commission here has ordered that certain actions be undertaken by Licensee under the license. Neither does this proceeding have the requisite indicia of a show cause enforcement proceeding under Subpart B of 10 C.F.R. Part 2. The May 7 Order is not an order to show cause or a notice of violation. An Atomic Safety and Licensing Board -- which on applications for construction permits, operating licenses and amendments thereto typically is the first decision-maker in the formal adjudicatory structure of this agency -- has been called upon to compile an evidentiary record essentially to review the adequacy of an immediately effective order

issued by the Commission, the ultimate adjudicating body in the agency. The Commission's Order of June 21, 1979 (and inferentially the Order of May 7, 1979) is under review in court,[5] and we assume that this decision will in turn be the subject of the usual appellate review within the agency and subsequent court review if it is sought. In the meantime, Licensee has proceeded to implement the Commission's Order of May 7, 1979, as directed.

19. Consequently, from the outset of this proceeding and at several subsequent junctures the Board has been called upon to consider the scope of its jurisdiction and of this proceeding. Licensing boards have limited jurisdiction. As delegates of the Commission, their authority extends only to matters which the Commission places before them, and they may exercise only those powers which the Commission has given them. Portland General Electric Company, et al. (Trojan Nuclear Plant), ALAB-534, 9 N.R.C. 287, 289 n.6 (1979); Union Electric Company (Callaway Plant, Units 1 and 2), ALAB-527, 9 N.R.C. 126, 144 (1979); Public Service Company of Indiana, Inc. (Marble Hill Nuclear Generating Station, Units 1 and 2), ALAB-316, 5 N.R.C. 167, 170 (1976). The Board believes these principles of general applicability warrant particularly careful adherence in a proceeding, such as this one, where a hearing is not mandatory.[6] Consequently, we looked first to the Commission orders which gave rise to this proceeding.

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5 Friends of the Earth v. United States, supra.

6 No hearings have been or will be conducted with respect to (footnote continued next page)

20. The subjects to be considered at the hearing were described by the Commission in its Order of June 21, 1979, supra, 9 N.R.C. at 681, to include:

> 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.[7]

(continued)
the orders issued to the other B&W operating licensees, except in the case of TMI-1, where the Commission determined that a hearing and further order of the Commission itself were mandatory prior to the restart of the facility. See, Metropolitan Edison Company, et al. (Three Mile Island Nuclear Station, Unit No. 1), Commission Order, Docket 50-289 (July 2, 1979), 44 Fed. Reg. 40461 (1979).

7 This order is in sharp contrast to the one issued by the Commission in the TMI-1 restart proceeding, where a much more expansive set of issues were specified for adjudication by that licensing board. See, Metropolitan Edison Company, et al. (Three Mile Island Nuclear Station, Unit No. 1), CLI-79-8, 10 N.R.C. 141 (1979). In the TMI-1 proceeding, the Commission concluded that "[i]n addition to the items identified for the B&W reactors, the unique circumstances at TMI require that additional safety concerns identified by the NRC staff be resolved prior to restart." Id., 10 N.R.C. at 143. Thus, it
(footnote continued next page)

The Board first expressed its view of the scope of the proceeding in footnote 3 to its Order for Filing of Amended and Supplemental Requests for [Hearing] and Notice of Prehearing Conference, July 2, 1979:

> The Commission has already identified in its June 21, 1979, Order the broad issues to be considered. Each requester for hearing can, of course, assert in its July 16 filing that further issues should be specified as long as they are related to the action taken by the Commission in its May 7, 1979, Order.

In addition, in an open public meeting on July 11, 1979, to consider whether or not to amend its Order of June 21, 1979, in this docket, the Commission 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.

21. Following a discussion of the scope of the proceeding and the Board's jurisdiction at the first prehearing conference on August 1, 1979, the Board directed the parties to file briefs on the subject. Prehearing Conference Order, August 3, 1979. The Board subsequently held, as a general matter, that the proceeding ". . . includes all matters and issues which hinge upon response to feedwater transients."

(continued)
is clear from the Commission's actions in the TMI-1 proceeding that the licensing board's jurisdiction in that case is substantially broader than this Board's, in a proceeding where only an interim shut down was ordered and restart explicitly was removed as an issue for hearing.

Order Ruling on Scope and Contentions, October 5, 1979, at 3. More specifically, the Board ruled that:

> In this proceeding, it will be appropriate to investigate questions concerning the propagation of a response throughout the Rancho Seco system, where "system" includes the physical facilities as well as the organization and personnel which operate them.

Id. The Board further stated:

> As to "various transient events" as the phrase is used at page four of the Commission's May 7 Order, we believe that, taken in the context of page five of that same Order, the scope of this proceeding can be expanded no further than ". . . feedwater and/or trip of the turbine . . ." We will, therefore, not allow matters such as loss of off-site power to be raised and considered among the contentions here.

Id. at 4. The Board also ruled that the subject of emergency planning was beyond the scope of the proceeding because it was about to become the subject of generic Commission rulemaking. Id. at 3, 4. In response to a motion by the California Energy Commission, the Board referred this ruling to the Atomic Safety and Licensing Appeal Board. LBP-79-33, 10 N.R.C. 821 (1979). The Appeal Board accepted the referral, but CEC subsequently moved to terminate the Appeal Board's consideration of the referred question. The Appeal Board granted the motion. ALAB-576, 11 N.R.C. 16 (1980).

22. Within the frame of these ground rules, the Board proceeded to rule upon the specific issues and contentions of CEC and the Intervenors. The Board's rulings, in its order of October 5, 1979, were supplemented in Order Relative

to Proposed New Schedule, December 4, 1979, and in Memorandum of Clarification, December 27, 1979. In Additional Board Questions, dated January 7, 1980, the Board posed written questions which it expected Licensee and the Staff to address at the hearing. and which it encouraged the other parties to address.

23. In addition, Licensee moved for summary disposition of all of the contentions of intervenors Hursh and Castro and one issue raised by CEC. While intervenors Hursh and Castro entered no opposition to Licensee's motion and, in fact, withdrew as parties to the proceeding prior to the Board's ruling on the motion, the Board denied summary disposition as to 9 of the 21 Hursh-Castro contentions and adopted them as Board questions.[8] In its Order Subsequent to the Prehearing Conference of February 6, 1980, dated February 14, 1980, the Board rephrased these 9 Hursh-Castro contentions in order to reflect more exactly the aspect of each contention which the Board felt presents a litigable question.[9] While FOE withdrew from the proceeding prior to the commencement of evidentiary hearings, Licensee and the NRC Staff presented testimony in

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8 Referred to hereafter as "Board Question H-C [number]".

9 Unfortunately, the schedule previously imposed called for the filing of written direct testimony prior to the reformulation of these contentions by the Board. While some testimony was supplemented in writing, most of the written testimony refers to the Hursh-Castro contentions as worded by those intervenors. The Board, however, examined each witness who addressed those subjects and asked the witness to respond to the reformulated questions advanced by the Board.

response to, and the Board herein will rule upon, each of the FOE contentions admitted on October 5, 1979.

24. The unique nature of this proceeding also raised questions about the allocation of the burdens of proof and of going forward with evidence. The Commission's Rules of Practice, at 10 C.F.R. § 2.732, provide simply that "[u]nless otherwise ordered by the presiding officer, the applicant or the proponent of an order has the burden of proof." It is not clear in this situation who is the proponent of the already issued, immediately effective Commission Order of May 7, 1979. See, e.g., comments of the NRC Staff at Tr. 10-12. In its Prehearing Conference Order of August 3, 1979, the Board ruled that the burden of proof on all contentions would be placed upon Licensee and the burden of going forward on contentions would be placed upon the party making the contention.

25. The participation of the California Energy Commission under 10 C.F.R. § 2.715(c) raised additional questions about the allocation of burdens. That rule allows a representative of an interested state a reasonable opportunity to participate and advise the Commission without requiring the representative to take a position with respect to the issues. It also provides that the Board may require the representative to indicate with reasonable specificity, in advance of the hearing, the subject matters on which he desires to participate. While CEC, throughout the course of the hearing and prior thereto, did not take a position on any issues before the

Board, CEC submitted a statement of issues of concern which it desired to have addressed in the hearing. In its brief, dated August 27, 1979, CEC took issue with the Board's ruling, stated above, on the allocation of burdens. Specifically, CEC argued that it did not have the burden of going forward with respect to its issues. In its Order Ruling on Scope and Contentions, October 5, 1979, at 6, the Board announced that some of CEC's "issues" would be framed as Board questions,[10] whereas others were referred back to CEC for the submission of succinct contentions. In a motion for reconsideration, dated October 24, 1979, CEC once more asked the Board to rule that Licensee and the NRC Staff should carry the burden of going forward on CEC's issues. In its Order Ruling on CEC's Motion of October 24, 1979 Relative to Burden and Going Forward, dated December 17, 1979, the Board ruled that CEC's issues are in the Board's opinion "contentions" and CEC has the burden of going forward with these issues. The Board also ruled that after it reframed certain CEC issues as Board questions, CEC was relieved of any unique burden of going forward with those issues. The Board did not rule on the burden of proof or the burden of going forward with respect to Board questions, other than to note that it intended Licensee, the NRC Staff and any other party to file testimony on them.

26. The withdrawal, prior to the evidentiary hearing, of each of the intervenors who requested this hearing

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10 Referred to hereafter as "Board Question CEC <u>[number]</u>".

and qualified for party status under 10 C.F.R. § 2.714 has affected the posture of the matters to be adjudicated in this proceeding. The viable Hursh-Castro contentions became Board questions and the FOE contentions, in the absence of the proponents to go forward with evidence, essentially must be treated as Board questions as well. CEC, participating as an interested state, took no position on the issues even though it presented testimony. In his opening statement, CEC counsel stated that his witnesses "represent a variety of viewpoints on nuclear power and none of their views necessarily reflect the views of the California Energy Commission Staff or the California Energy Commission itself." Tr. 349. Consequently, no participant in this hearing contended that the Commission Order of May 7, 1979, is inadequate. Given this essentially uncontested posture of the hearing, Licensee's assigned burden of proof has been altered. The matters ultimately heard were: (a) 18 Licensing Board questions; (b) 4 contentions of a withdrawn intervenor (which, as we have stated, must be treated essentially as Board questions); and (c) 7 "issues" raised by CEC, but on which CEC took no position.

27. In spite of the absence of formal controversy, the Board conducted a thorough inquiry into the matters before it, which raise serious questions as to the adequacy of the Commission's Order of May 7, 1979. These matters were all addressed with written testimony by Licensee and the NRC Staff, written testimony or cross-examination by the California Energy

Commission, and examination by the Board. The issues heard, which are fully described below in our findings of fact, address fundamental aspects of the B&W nuclear steam supply system, related balance-of-plant design features at Rancho Seco, Licensee's operating procedures, the competence of Licensee's management and plant operators, control room configuration and diagnostic instrumentation as they relate to feedwater transients, and several plant modifications suggested by the issues and testimony presented.

28. Finally, in order to understand the Board's view of the scope and nature of this proceeding, it may be helpful to identify some of the subjects which are not a part of this proceeding. The accident at Three Mile Island has been the subject of investigation and study by the President's Commission on the Accident at Three Mile Island, the NRC's Special Inquiry Group, committees of the Congress, the Commonwealth of Pennsylvania, NRC's Office of Inspection and Enforcement, the Advisory Committee on Reactor Safeguards, and NRC's TMI-2 Lessons Learned Task Force. Staff Ex. 4 at 3-6 to 3-8. In addition, the NRC established a Bulletins and Orders Task Force in 1979 to review and direct TMI-2-related Staff activities, and in 1980 established a B&W Reactor Tr[illegible]nt Response Task Force to assess the generic aspects of [illegible]ng experiences of the B&W plants. Staff Ex. 4 at 1-1, 3-5 and 3-6. Further, the NRC appointed a TMI-2 Action Plan Steering Group to organize, define and assess the recommendations of

various groups by developing a TMI-2 Action Plan which would provide a comprehensive and integrated plan for all actions judged necessary by the NRC to correct or improve the regulation and operation of nuclear facilities based on the experience from the accide at TMI-2 and the official studies and investigations of the accident. Staff Ex. 4 at 3-9. This Board does not view its function to be to conduct an additional investigation into the accident at Three Mile Island, or to assess the necessity or adequacy of the many requirements, other than those of the Commission's Order of May 7, 1979, which have been imposed upon Licensee, other B&W plants, other PWRs and other operating plants generally. Neither has the Commission directed this Board to determine Licensee's compliance with the May 7 Order (in the absence of a contention challenging the Staff's conclusions)[11] or any of these other post-TMI-2 requirements.[12] On the other hand, the Board has not addressed the May 7 Order in a vacuum or isolated in time. Where it is relevant to our assessment of the adequacy of that

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11 The Commission specifically delegated to the Director, Office of Nuclear Reactor Regulation, the responsibility of determining Licensee's compliance with the short-term actions required by the May 7 Order. 44 Fed. Reg. 27779, 27780 (1979).

12 The Commission separately has issued orders, with the opportunity to request a hearing, with respect to some of these requirements. <u>See</u>, <u>e.g.</u>, in this docket, Order to Show Cause, January 2, 1980, 45 Fed. Reg. 2447 (1980) (Category A requirements of NUREG-0578, TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations); Confirmatory Order, April 14, 1980, 45 Fed. Reg. 26856 (1980) (Short-term actions in response to the event which occurred at Crystal River, Unit 3 on February 26, 1980).

order, we have received evidence on related post-TMI-2 requirements imposed or under consideration by the NRC. In the final analysis, however, the Board views its charge from the Commission to be to determine whether the actions and modifications required by the May 7 Order provide reasonable assurance that the Rancho Seco facility will respond safely to feedwater transients.

C. Description of the Record

29. The record of the hearing includes the written and oral testimony of four witnesses presented by Licensee, six witnesses presented by the California Energy Commission, and eighteen witnesses presented by the NRC Staff. In the findings of fact below, the location of the direct written testimony of each witness will be identified fully the first time it is cited. For convenience, we have also compiled an alphabetical listing by witness, Appendix A to this decision, which identifies the location in the transcript of all of the written testimony.

30. The record also includes exhibits which were offered and received into evidence. Appendix B to this decision is a list of the exhibits which were marked for identification. It also identifies those exhibits which were received into evidence.[13]

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13 Appendix B does not provide for an identification of exhibits offered but not admitted into evidence by the Board because all offered exhibits were received.

D. Organization of the Decision

31. The Board's findings of fact have been organized by subject matter. Each Board question, issue and contention which is addressed under a given subject is quoted in full at the outset of our findings on that subject. Some subject matter sections address only one specific question or contention, while others address a number of them. While there is considerable overlap among the various specific questions, issues and contentions, the Board has reached its concluding findings of fact on any given question, issue or contention only in the section in which it is identified by quotation.

32. The arrangement of our findings of fact is best viewed from an examination of the Table of Contents to this Initial Decision. The order in which subjects are addressed reflects what the Board believes to be a logical sequence. The Board first addresses matters related directly to the B&W designed nuclear steam supply system, plus the Rancho Seco auxiliary feedwater system. Sections II.A through II.H, infra. Next, operator and management competence are addressed (section II.I), followed by the Board's findings on instrumentation and control room configuration (sections II.J and II.K). The Board then reviews the long-term modifications as a whole. Section II.L, infra. Finally, we examine those Board questions and issues raised by CEC which address potential modifications to the facility. Sections II.M through II.O, infra. The Board's Findings of Fact are concluded in section II.P, which is followed by our Conclusions of Law and Order.

II. FINDINGS OF FACT

A. 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?

33. One of the long-term actions directed by the Commission in its Order of May 7, 1979, was that "[t]he licensee will submit a failure mode and effects analysis of the Integrated Control System to the NRC staff as soon as practicable." 44 Fed. Reg. at 27779 (1979). Such an analysis was performed by B&W for Licensee as part of B&W's study of the reliability of the integrated control system ("ICS"). The results of B&W's reliability study are contained in B&W Report BAW 1564, "Integrated Control System Reliability Analysis" (CEC Ex. 3).

34. In order to assess the completeness and adequacy of B&W's analysis, it is important first to understand the Rancho Seco ICS and the Staff's concerns regarding it. The ICS is an automatic control system whose basic function is to match continuously a unit's power generation to its load demand. The ICS does this by coordinating the rate of steam generation and the steam flow to the turbine. NRC Staff Testimony of Dale F. Thatcher Relative to the Integrated Control System (Board Question 16), following Tr. 163 ("Thatcher ICS Testimony"), at 2.

35. During normal operations, the ICS provides proper coordination of the reactor, steam generator, feedwater control, and turbine. Proper coordination consists of producing the best load response to unit load demand within the limitations and capabilities of the plant equipment. Id. at 3.

36. Almost all PWRs, and indeed many fossil-fired units,[14] have automatic control systems which perform the functions of the ICS. The basic functional design of the ICS is not unusual or particularly complex. The unique feature of the ICS is that through feed-forward control the ICS produces signals for parallel control of the turbine, reactor and steam generator feedwater, integrating these three signals to achieve an optimum response of the actuated components. Tr. 622, 1104-1105 (Karrasch); Thatcher ICS Testimony at 2.

37. The ICS includes four subsystems: unit load demand control, integrated master control, steam generator control, and reactor control. Thatcher ICS Testimony at 2. Each of these subsystems (except for the unit load demand control) regulates and interacts with a number of other plant control systems, such as the control rod drive system and the feedwater pump and valve controls. Id. at 3.

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14 The B&W ICS is an adaptation of a control system successfully employed by B&W in its fossil-fired boilers. Licensee's Testimony of Bruce A. Karrasch and Robert C. Jones, Etc., dated February 11, 1980, following Tr. 535 ("Karrasch-Jones Testimony"), at 7. All B&W PWRs have an ICS similar to that existing at Rancho Seco. Tr. 741 (Karrasch); Tr. 3748 (Capra).

38. During normal power operation (15 to 100% rated power) the ICS maintains constant average reactor coolant temperature, as well as constant turbine inlet steam pressure at all loads. It also performs a variety of limiting actions to optimize performance and to maintain a proper relationship between generated electrical load, steam pressure, feedwater flow, and reactor power. Id. The ICS can provide automatic response to and control of step load changes up to 10% rated power per minute and ramp load changes up to 5% rated power per minute. Tr. 618, 619 (Karrasch). The ICS has been proven to provide stable plant control over the complete load range from 15 to 100% percent power, both during steady state and transient conditions. Karrasch-Jones Testimony at 9, 10.

39. The ICS is also designed to reduce power automatically to a lower value for certain anticipated transients (i.e., to "runback"), while maintaining plant parameters within the limits of the reactor protection system ("RPS"). The ICS is thus designed to keep the reactor on line during transients, reducing the potential for reactor trips and enhancing plant availability. Karrasch-Jones Testimony at 7; Tr. 1076 (Karrasch).[15]

40. If the RPS limits are exceeded and a reactor trip takes place,[16] the ICS ceases its overall control function

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15 Because of the addition of anticipatory reactor trips on loss of feedwater and turbine trip, some of the transient-mitigating functions of the ICS are no longer in effect. Karrasch-Jones Testimony at 10.

16 A reactor trip occurs when the RPS causes the control (footnote continued next page)

and limits itself to controlling steam pressure by means of the turbine bypass valves,[17] and steam generator level through the feedwater startup valve.[18] Tr. 1117-1119 (Karrasch).

41. The ICS has no role in starting the pumps that deliver auxiliary feedwater ("AFW"). Those pumps are started automatically upon the loss of reactor coolant pumps or main feedwater pumps or upon a safety features actuation signal ("SFAS"). Tr. 1384 (Thatcher); Tr. 2047 (Dieterich). However, if main feedwater is not available, the ICS controls the delivery of AFW to the steam generator by actuating the AFW flow control valves so that the pre-set steam generator level is maintained. Tr. 624 (Jones).

42. It was this role of the ICS in controlling the delivery of AFW to the steam generator that led to a renewed Staff intere[illegible] in the reliability of the ICS shortly after the TMI-2 accident of March 28, 1979.[19] Two sorts of concerns

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rods to unlatch and drop into the reactor. While the ICS normally governs the position of the control rods, the rate of control rod insertion provided by the ICS is too slow to accommodate a transient that exceeds the RPS limits. Once the control rods have been unlatched they are no longer under ICS control. Tr. 611-612, 1117 (Karrasch).

17 Following a reactor trip, the turbine trips immediately upon a signal from the RPS. The ICS then closes the main feedwater valves and opens the turbine bypass valves to provide a heat sink. Tr. 1118, 1119 (Karrasch).

18 The steam generator level is maintained at a pre-established setpoint by the ICS if main or auxiliary feedwater is available. Tr. 1105 (Karrasch).

19 There was a concern within the Staff at that time that the
(footnote continued next page)

appear to have been formulated: (1) that a failure or malfunction of the ICS could prevent AFW from being supplied to the steam generator during a loss of feedwater transient; and (2) that a failure or malfunction of the ICS could be the cause of such a transient. Tr. 1270-1272 (Thatcher).

43. The first concern was addressed on a short-term basis in the Commission Order of May 7, 1979, by requiring Licensee to "[d]evelop and implement operating procedures for initiating and controlling auxiliary feedwater independent of Integrated Control System control." 44 Fed. Reg. at 27779 (1979). As will be further discussed below, this short-term modification was accomplished by Licensee prior to the restart of Rancho Seco. The adequacy of Licensee's compliance with this aspect of the May 7, 1979 Order was established by the Staff by visiting the site and conducting examinations of the operators to verify the adequacy of their training. This evaluation included a walk-through of some of the procedural aspects of manually controlling AFW independently of the ICS and a review of plant diagrams to verify that the valves that would be utilized for AFW flow control were indeed independent

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ICS may have contributed to the TMI-2 accident. Tr. 1270 (Thatcher). This concern, later proven to be erroneous, was based largely on what Staff witness Capra characterized as "myth and folklore" -- that is, incomplete knowledge on the part of the Staff as to the reliability and operating history of the ICS. Thatcher ICS Testimony at 4; Tr. 3713 (Capra). The ICS is not a safety system and therefore it had not received, prior to the TMI-2 accident, the detailed level of review given by the Staff to safety-related systems. Tr. 937, 938 (Karrasch).

of the ICS. Thatcher ICS Testimony at 4, 5; Tr. 1386 (Novak); Tr. 3730, 3731 (Capra); Staff Evaluation at 13.

44. A permanent solution to the first concern has been provided by Licensee's commitment to install during the 1981 refueling outage a safety-grade AFW control system independent of the ICS. This modification will remove completely the operation of the AFW system from the ICS. Thatcher ICS Testimony at 5; Tr. 1273 (Thatcher).

45. The second concern relating to the ICS led the Staff to ask that a failure mode and effects analysis ("FMEA") of the ICS be performed. Since the Staff was interested in the potential role of the ICS as the instigator of a transient, it sought to have an analysis made of the reliability of the ICS and the effects of failures of that system on the plant's operation. Tr. 648, 937-939 (Karrasch); Tr. 1270-1273 (Thatcher).

46. The reliability analysis of the ICS was performed by B&W on behalf of all licensees owning a B&W nuclear steam supply system ("NSSS"). However, the unit chosen to model the plant-specific elements of the ICS was Rancho Seco; so there is no question as to the applicability of the results of the analysis to the Rancho Seco unit. Tr. 694 (Karrasch); CEC Ex. 3 at 4-19. The methodology selected for the reliability evaluation of the ICS consisted of a three-part analysis: the FMEA, a systems simulation, and operating data collection and analysis. The FMEA was used to identify the

failures within and without the ICS which could lead to plant transients. The simulation model was used to study in more detail the failures identified by the FMEA. Finally, collection and analysis of operating data were used to compare the operating history to the analytical results. Oak Ridge National Laboratory ("ORNL") Review of Integrated Control System Reliability Analysis, Bd. Ex. 1 ("ORNL report") at 5.

47. A FMEA is a systematic procedure for identifying the modes of failure of a system and for evaluating the consequences of such failures. Thatcher ICS Testimony at 6. By definition, a FMEA seeks to determine if any single failure of a system can prevent the system's function. If the system is one impacting a safety function, it is a requirement of plant operation that no single failure of the system shall prevent the safety function from being accomplished. Id. at 6, 7.

48. To perform the ICS FMEA, a functional block diagram[20] of the ICS was developed to permit the major functional points of each input to and output from the ICS to be analyzed. CEC Ex. 3 at 4-4. Once this was done, each input to

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20 A functional block diagram attempts to describe a system in terms of blocks that represent the various functions performed by the elements in the system. By contrast, a component block diagram will show the actual mechanical or electronic modules that exist in the system. Tr. 646 (Karrasch). While a functional block diagram can be drawn to different degrees of detail, a component block diagram will usually be more detailed than a functional block diagram. Tr. 647 (Karrasch).

and output from the ICS, and each functional block, was failed "high" and "low" to determine the effect of the failure on the NSSS and the plant at large.[21] The FMEA concentrated on worst cases, i.e., failures that caused the most drastic transient. CEC Ex. 3 at 4-20. Since the analysis was conducted so that each piece of equipment actuated by the ICS ended up in the position that produced the most severe consequences, the FMEA analysis bounds all intermediate failure modes. Tr. 1086 (Karrasch).

49. For each failure considered in the FMEA, the analysis was carried out to determine whether there would be a reactor trip as a result of the failure and, if so, whether there was a potential for requiring AFW or high pressure injection ("HPI") after the trip. Tr. 645 (Karrasch). The failures then were divided into three categories: category one, those that did not have a significant effect on the operation of the plant and were unlikely to cause a reactor trip; category two, those that could cause a reactor trip but were unlikely to have an adverse impact on the plant beyond that; and category three, those failures for which an impact beyond the reactor trip was likely, requiring actuation of systems such as HPI and AFW. Tr. 639, 640 (Karrasch); CEC Ex. 3 at 4-22.

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21 For ICS inputs, "high" was selected as the maximum output of the transmitter and "low" as the minimum output. For ICS outputs, "high" was chosen as the output signal that fully opened valves, caused pumps to attain maximum speed, pulled control rods, etc., while a "low" output caused the inverse of these actions. CEC Ex. 3 at 4-20.

50. The FMEA identified a number of ICS input, output and functional block failures that could lead to reactor trip,[22] although only a small proportion of these fell into "category 3", i.e., would require AFW or safety systems actuation.[23] The operating history of the ICS at B&W plants reveals, however, that only a few of these potential failures have actually been experienced. Thus, only 6 out of 162 instances of ICS hardware malfunctions in 35 years of operating experience at B&W plants have resulted in a reactor trip. CEC Ex. 3 at 5-8; Tr. 1774 (Thatcher). This data demonstrates that the system is failure tolerant to a significant degree. Thatcher ICS Testimony at 8; Bd. Ex. 1 at 14. The tolerance to failure is due in part to a number of cross-checking features built into the system. Bd. Ex. 1 at 14, 15.

51. Indeed, the reliability record of the ICS, as outlined in the operating history section of CEC Exhibit 3, is impressive. Only 6 of 310 reactor trips experienced from all causes in 35 years of B&W reactor operating history were caused by internal ICS failures. Tr. 710 (Karrasch); CEC Ex. 3 at 5-4 and 5-5.[24] By contrast, in a 5-year period for just one plant

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22 The FMEA identified approximately 40 types of category 1 failures, about 60 types of category 2 failures, and 15 types of category 3 failures. CEC Ex. 3 at 4-61 to 4-64.

23 In every hypothetical ICS failure studied in the FMEA the reactor core remains covered throughout the transient. CEC Ex. 3 at 2-1; Tr. 1090 (Karrasch).

24 There are two "generations" of B&W integrated control systems in the field: the "721" system on earlier plants and (footnote continued next page)

(apparently Rancho Seco), the ICS carried out 47 successful runback operations during transients that would otherwise have led (at least in most cases) to a reactor trip. Tr. 701-702, 704, 706 (Karrasch). Thus, in addition to its important role during normal plant operations, the ICS is a positive contributor to plant safety during transients, for it prevents more trips than it causes and reduces the net number of challenges to the reactor protection system.[25] Tr. 702-703, 713 (Karrasch); CEC Ex. 3 at 2-2; Bd. Ex. 1 at 14, 15.

52. In its review of the B&W reliability analysis of the ICS, the ORNL report agreed that the ICS has a low failure rate and is not a significant contributor to plant transients.[26]

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the "820" system on later plants. Rancho Seco has an 820 system. Bd. Ex. 1 at 23. The 820 system has major hardware changes that account for improved reliability. The mean time between failures of the 820 system is on the order of 33,000 to 49,000 hours, over ten times longer than for the 721 system. CEC Ex. 3 at 5-8 and 5-9. All six ICS failures that resulted in a reactor trip occurred in plants that used the 721 system. No trips due to ICS failure have been experienced so far in plants having the 820 ICS. CEC Ex. 3 at 5-14 and 5-15.

25 The number of ICS runbacks will probably decrease somewhat in the future because of the anticipatory trips on loss of main feedwater and on turbine trip implemented in response to the Commission's Order of May 7, 1979. Tr. 1088, 1089 (Karrasch). Such a reduction, however, would not negate the validity of the comparison between trips caused and averted by the ICS: even with a decreased number of successful runbacks, the ICS will have averted more trips than it caused. Tr. 1089 (Karrasch). This is particularly true at Rancho Seco, which has a more reliable "820" ICS, a system whose malfunction has never led to a reactor trip.

26 Both the B&W reliability analysis and ORNL's evaluation were examined in a Staff document dated May 9, 1980, entitled "Assessment of B&W Report BAW-1564, 'Integrated Control System
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Bd. Ex. 1 at 13, 14; Tr. 1278 (Thatcher). While agreeing with B&W's findings and conclusions and with the recommendations made by B&W for further improvements in areas relating to the ICS, the ORNL report pointed out a number of perceived deficiencies in B&W's approach to the FMEA portion of the reliability analysis. Tr. 1706-1707, 1774 (Thatcher).

53. The main criticism leveled at the FMEA by ORNL was that the scope of the FMEA was too limited, leading to results having only limited value.[27] Bd. Ex. 1 at 4. The scope limitations identified by ORNL were: (1) not considering the interactions between plant safety and non-safety systems such as the ICS; (2) not including analysis of failures of plant systems external to the ICS; (3) not considering multiple system failures; and (4) utilization of functional as opposed to component diagrams as the building blocks in the analysis. Bd. Ex. 1 at 3-4, 6-8.

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Reliability Analysis.'" This Staff document agrees with the ORNL report that the ICS does not initiate a significant number of challenges to the plant's reactor protection system. Staff Ex. 5 at 6.

27 This criticism is apparently based on a misunderstanding by ORNL of the anticipated scope of the FMEA and a confusion between the requirements of the Commission's Order of May 7, 1979, and the suggestions for further study made by the Staff in its document "Staff Report on the Generic Assessment of Feedwater Transients in Pressurized Water Reactors Designed by the Babcock & Wilcox Company", NUREG-0560 (May 1979) (this document is incorrectly cited in Bd. Ex. 1 as "the NRC shutdown orders"; see Bd. Ex. 1 at 2 & n.2). NUREG-0560, which was issued after the May 7, 1979 Order, went beyond the Order in its recommendations for further action. Tr. 1277, 1278 (Thatcher). It is evident that the ICS FMEA was not intended to respond to the recommendations in NUREG-0560.

54. The first of these alleged limitations is somewhat of an unfair criticism. A study of the "plant interactions resulting from failure in non-safety sytems, safety systems and operator actions" (Bd. Ex. 1 at 2) was already contained in the Rancho Seco licensing safety analysis. Tr. 685, 686 (Karrasch). It was not required to be made anew in the ICS FMEA, but was merely recommended as a future action in NUREG-0560.[28]

55. The second criticism by ORNL was that the FMEA did not deal with failures of plant systems that the ICS controls and with which it interacts. Bd. Ex. 1 at 3. However, the FMEA did analyze failures of every input to the ICS and the effect of such failures, and likewise considered the failure of each piece of equipment actuated by the ICS.[29] Therefore, the ORNL statement to the contrary is at least partially incorrect. Tr. 681-683 (Karrasch). However, to the extent that the ORNL criticism goes to B&W's failure to perform a FMEA of the systems providing inputs to the ICS or actuated by it, the criticism appears accurate, but again beyond the requirements of the May 7 Order.[30] For example, power supply

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28 Indeed, the Staff has a study under way known as the "Integrated Reliability Evaluation Program (IREP)" which seeks, among other things, to identify the risk significance, if any, of the systems interactions originating in the ICS at B&W plants. Thatcher ICS Testimony at 8.

29 The ICS actually controls only a few pieces of equipment: the feedwater control and startup valves, the turbine bypass valves, the main feedwater pumps, and the control rods. Tr. 1085, 1086 (Karrasch).

30 In the short time frame (two months) desired by the Staff (footnote continued next page)

failures were studied in the FMEA only insofar as they constituted ICS input failures; the effects of power supply failures on the rest of the plant were not analyzed.[31] Bd. Ex. 1 at 11; Tr. 699, 700 (Karrasch). With respect to the importance of this possible shortcoming of the FMEA, ORNL concluded that the FMEA would have been of greater significance if it had been expanded to include other systems with which the ICS interacts, but also concluded that redoing the FMEA analysis to incorporate these areas would be of doubtful usefulness.[32] Bd. Ex. 1 at 16.

56. The third criticism of the FMEA's scope by ORNL is that it did not consider the effects of multiple system failures. Again, this is not a valid criticism, because the overall purpose of a failure mode and effects analysis is to

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for the performance of the FMEA, it would appear logical, if not necessary, that B&W would concentrate its study on the reliability of the ICS itself, i.e., the contents of the ICS cabinets. Tr. 1274, 1275 (Thatcher); Tr. 1277 (Capra).

31 One of the recommendations made by B&W in its ICS reliability study was that the reliability of the non-nuclear instrumentation and ICS power supplies should be increased. CEC Ex. 3 at 3-1; Tr. 699, 700 (Karrasch); Tr. 1738 (Thatcher). Licensee has addressed this recommendation. See paragraph 58, infra.

32 Similarly, while the ORNL report stated that a more extensive (whole plant) simulator that allowed complete following of a transient was desirable, it concluded that the simulator used by B&W (which included all the plant systems but was able to duplicate plant parameters only within a limited dynamic range) was adequate and more detailed analysis would not provide more enlightening information for the purposes of the study. Bd. Ex. 1 at 12-13, 16.

evaluate the impact on the plant of each single failure of the system under consideration. Thatcher ICS Testimony at 6, 7; Tr. 1083-1084, 1090 (Karrasch). In any event, the effects of multiple failures of the ICS were bounded by the "worst case" safety analyses performed in preparing the Rancho Seco Final Safety Analysis Report.[33] Tr. 1084, 1109, 1112-1113 (Karrasch). The ORNL report also concluded that a different methodological approach by B&W might have permitted assessment of the significance of multiple ICS failures, but that further analysis of this type might not be economically justifiable given that failures within the ICS do not constitute a significant threat to plant safety.[34] Bd. Ex. 1 at 15.

57. The fourth of ORNL's main criticisms was with the use of functional block diagrams instead of component diagrams in the FMEA. There is no dispute that a component block diagram might have permitted a more "detailed" FMEA to be performed. Tr. 647 (Karrasch). However, the only additional information that would have been gained by utilization of a

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33 The ORNL report suggests that certain ICS failures may go undetected and not become apparent until a second ICS failure takes place. Bd. Ex. 1 at 8; Tr. 695, 696 (Karrasch). However, to the extent that the second failure impacted other systems outside the ICS, it would have been studied in the FMEA. Tr. 697 (Karrasch).

34 ORNL also felt it would have been "instructive" to examine the consequences of single ICS failures occurring when there was less than a full complement of coolant pumps or in the presence of other off-normal plant conditions. However, ORNL concluded that redoing the analysis to take into account such conditions might not be worthwhile. Bd. Ex. 1 at 10-11, 16.

component block diagram would have been an estimate of the probability of failure of each functional block based on the failure rates of each component (assuming these are known). Tr. 1086 (Karrasch). While such information would indeed be useful, it is irrelevant to the purpose of the ICS reliability analysis -- which was to determine the effects on the rest of the plant of a failure in each of the elements in the ICS, regardless of the likelihood of such failure. Tr. 647, 648 (Karrasch). Thus, use of component block diagrams would not have added anything to the FMEA. The ORNL report itself concludes that further ICS reliability analysis utilizing component block diagrams was not economically justifiable. Bd. Ex. 1 at 15.[35]

58. Based on its ICS reliability analysis, B&W made a number of recommendations as to areas, largely outside the ICS, in which improvements could potentially contribute to the overall operation of the facility. Thatcher ICS Testimony at 7; CEC Ex. 3 at 3-1. These recommendations for further study were endorsed by the ORNL report and by the Staff's evaluation of both documents. See Bd. Ex. 1 at 17; Staff Ex. 5 at 6. Licensee has responded to these recommendations and has made

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35 A reason advanced by ORNL for preferring component block diagrams is that there could be undisclosed couplings or interactions between the functional blocks used in the FMEA. Bd. Ex. 1 at 6. This does not appear to be a sound criticism, however, because any failure in one functional block in the ICS would be carried through the other blocks with which it interacts until it impacted the actuated equipment. Tr. 691, 692 (Karrasch).

substantial changes in perhaps the most important area, non-nuclear instrumentation power supplies.[36] These changes, which are now completed, increase the reliability of the power supplies to the ICS. Tr. 3703, 3711 (Capra); Thatcher ICS Testimony at 9.[37]

59. Additional recommendations for further study and action in areas relating to the ICS are currently under consideration by the NRC Staff. For instance, NRC's B&W Reactor Transient Response Task Force recommends that plant control systems "be improved to reduce the number of challenges to the safety systems." Staff Ex. 4 at 5-61. However, control systems in this context have the broader meaning of all plant control systems, including inputs to the ICS. No problems were identified by the task force relating to the ICS itself.[38] Tr. 3716 (Capra).

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36 It was a failure of the non-nuclear instrumentation power supplies that initiated the transient of February 26, 1980, at Crystal River Unit 3 ("CR-3"). Tr. 1737 (Thatcher). In the CR-3 event, the ICS functioned properly, although it took erroneous action based on the failed input signal it received from the non-nuclear instrumentation. Tr. 1737, 1738 (Thatcher).

37 Of the other areas identified in the ICS reliability study, Licensee is considering changes to increase the reliability of the reactor coolant flow input signal to the ICS. Tr. 3703, 3704 (Capra). Licensee has also developed procedures to control ICS and to improve the "tuning" between the ICS and the balance of the plant, and has further trained its operators in ICS control. Tr. 3704, 3705 (Capra).

38 The licensee has already taken actions which implement some of the recommendations by the task force. For instance, it has complied with the intent of the recommendation that the power buses and signal paths for non-nuclear instrumentation and associated control systems be separated and channelized to reduce the impact of failure of one bus. Tr. 3717 (Capra).
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60. From the foregoing, the Board finds, in response to the inquiry in Board Question H-C 16, as did the Staff in evaluating the ICS reliability study, that the Rancho Seco ICS FMEA is complete and, together with the rest of the reliability analysis contained in CEC Exhibit 3, constitutes an adequate assessment of the reliability of the plant's ICS. See Tr. 1290, 1706 (Thatcher). Moreover, the assessment shows that the ICS at Rancho Seco is a highly reliable system, one that prevents or mitigates many more plant upsets than the few it creates, and a superior system to manual or fragmented control schemes. Bd. Ex. 1 at 15. The ICS design concept -- i.e., coordinated control of reactor power, feedwater, and turbine power, runback features on upset events, and cross limits -- is a correct and proper control strategy. CEC Ex. 3 at 5-6. Reliance on the ICS to regulate feedwater and other plant parameters is therefore not a shortcoming but a significant asset to plant availability and safety. Bd. Ex. 1 at 14.

61. Finally, it is clear that the performance of the ICS is not a factor in the ability of Rancho Seco to respond safely to a feedwater transient.[39] Tr. 1107 (Jones). The ICS

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Rancho Seco is also the only plant which already has procedures for loss of one or both channels of non-nuclear instrumentation. Tr. 1256 (Capra).

39 No credit is given to the ICS in safety analyses for the plant's ability to respond to a feedwater transient, even though in reality the ICS helps to mitigate such a transient. Tr. 1279-1281 (Thatcher, Capra). For instance, in an overfeed situation the ICS will try to compensate for the decrease in

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is not and should not be a safety system, Tr. 1312, 1313 (Capra, Thatcher), and is therefore subject to occasional failures that can lead to feedwater transients or accentuate ongoing ones. Tr. 624 (Karrasch); Tr. 1285 (Thatcher). However, ICS failures can be mitigated adequately by the reactor protection system, if not by the ICS's own cross-checking features. Thatcher ICS Testimony at 9; Bd. Ex. 1 at 14, 15. More importantly, there is no interaction between the ICS and the systems upon which the plant relies to respond to feedwater transients.[40] Tr. 1284-1287 (Thatcher, Capra). This lack of interaction has been confirmed by the operating experience of B&W plants, for ICS failures have not produced adverse impacts on the plants' safety systems. Tr. 685, 686 (Karrasch). In fact, there is no failure of the ICS that could leave the Rancho Seco plant in a worse condition (from the

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reactor coolant temperature and pressure by pulling the control rods in order to maintain a stable Tave in the system. Tr. 601, 606 (Jones). The ICS would also attempt in that situation to close the feedwater control valves to reduce feedwater. Tr. 766 (Karrasch).

40 For example, if there is a problem with or malfunction of the ICS the operator can detect it from the presence of an AFW initiation signal accompanied by a continued decline of steam generator level and no flow indication on the feedwater flow meters. Tr. 1480 (Matthews, Novak); Tr. 1481, 1482 (Capra). Upon detection of the ICS failure the operator can establish or decrease AFW flow, as appropriate, by means of manually controlled flow valves which are independent of the ICS. Tr. 1286 (Capra); Tr. 1387 (Matthews); Tr. 1387-1392 (Matthews, Novak); Tr. 1402, 1403 (Novak); Tr. 1512-1514 (Matthews). In the future, of course, with complete independence of the AFW from the ICS, not even that procedure will be necessary. Tr. 1286 (Capra).

safety viewpoint) than it would be if it did not have an ICS at all. Tr. 1113 (Karrasch).

62. Although the ICS and related control systems contain areas which potentially could be improved, the ICS has proven to have a low failure rate and does not appear to initiate a significant number of plant upsets. Thatcher ICS Testimony at 8, 9. Therefore, it is consistent with the enhancement of the public health and safety to continue operation of the Rancho Seco plant with the present ICS configuration until such time as further improvements or refinements of the system, if any, are identified.

B. Feedwater Transients

FOE Contention 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.

63. This contention by FOE apparently was inspired by an NRC Staff study (NUREG-0560), initiated shortly after the Three Mile Island accident, to assess the effect of feedwater transients on B&W reactors. While reviewing the significant feedwater transients that had occurred at B&W plants, the Staff also reviewed the operating experience at all PWR plants from March, 1978, to March, 1979. The events reviewed in this study were simply the cases where forced plant shutdown resulted from

a feedwater system malfunction. The study was described by Staff witnesses as "cursory in nature," designed to see if "a vast difference" in feedwater related malfunctions existed for the various vendors. While the Staff found a somewhat larger number of such events for B&W plants, it was not felt to be an appreciably higher frequency than for the other vendors. The Staff also expressed the thought that the greater number of feedwater transients may have been due to the generally younger age of B&W plants. In any event, a somewhat greater frequency of feedwater related transients was not by itself considered by the Staff to be a safety concern. NRC Staff Testimony of Mark P. Rubin and Thomas M. Novak Regarding the Acceptability of Feedwater Transients Referenced in NUREG-0560 (FOE Contention IIIa), following Tr. 1163 ("Rubin-Novak Feedwater Testimony"), at 3. In a more recent examination of feedwater transients at all operating plants since the Three Mile Island accident, the NRC Staff found that a substantial portion of the reactor trips that occur in all PWRs are associated with feedwater transients and that B&W was second among the three PWR vendors (B&W, Combustion Engineering and Westinghouse) in the number of feedwater transients per plant. Tr. 3754 (Capra).

64. One witness testified that B&W plants historically have been more prone to feedwater transients than other PWRs, but he relied on the same cursory study (NUREG-0560) discussed above in Paragraph 63. Prepared Direct Testimony of Clifford M. Webb Concerning Design Sensitivities

of the Babcock and Wilcox Nuclear Steam Supply System, following Tr. 1801 ("Webb Testimony"), at 5, n.5. Data presented by Licensee for the year 1978 shows that the frequency of feedwater transients causing reactor trip at B&W reactors was less than the corresponding rate for other PWRs. Karrasch-Jones Testimony at 13, 14. The Board finds that a comparison of feedwater transients leading to reactor trip is more relevant to the other issues in this proceeding than the broader data base, feedwater malfunctions leading to forced plant shutdown, used in NUREG-0560.

65. There is no dispute that feedwater transients occur in all pressurized water reactors. There is no reason why the Commission's Order of May 7, 1979, should have commented specifically on recent operating experience which the Staff had reported to the Commission prior to the issuance of the Order. Rubin-Novak Feedwater Testimony at 4. The important consideration, to which we have devoted the remainder of this decision, is whether the Rancho Seco plant is designed to accommodate feedwater transients safely. The Board finds that the Commission Order of May 7, 1979, does not fail to provide reasonable assurance of safety for the reasons set forth in FOE Contention III(a).

C. Once Through Steam Generator Sensitivity

Additional Board Question 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. overcooling *vis-a-vis* a small-break LOCA).

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

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

66. Additional Board Question 3 raises a general concern and a specific concern. The general concern goes to the sensitivity of the once-through steam generator ("OTSG"), a principal component of the B&W designed nuclear steam supply system at Rancho Seco. While the Board will summarize here its findings on this central concern, much of the remainder of this decision is devoted to more specific issues which relate back to the general issue raised in Additional Board Question 3. The more specific concern, operator diagnosis and response to overcooling events and small-break, loss-of-coolant accidents, is addressed here as well as in section II.F of this decision.

67. The OTSG employed by B&W is used for heat transfer from the primary to secondary coolant. Karrasch-Jones Testimony at 16. It is called a "once through" design because

the primary coolant rejects heat to the secondary coolant during a single pass through the unit. Staff Ex. 4 at 5-16. Each steam generator (there are two at Rancho Seco) has approximately 15,000 vertical straight tubes. Primary coolant flows down inside the steam generator tubes, while the secondary coolant flows up from the bottom on the shell side of the OTSG.[41] NRC Staff Testimony of Mark P. Rubin and Thomas M. Novak Regarding the Sensitivity of the Once-Through Steam Generator Design (Additional Board Question 3), following Tr. 1163 ("Rubin-Novak OTSG Testimony"), at 3. Because the tubes are only partially covered with secondary liquid, steam generated by the boiling of secondary liquid is superheated before exiting to the steam piping system. Primary-to-secondary heat transfer is controlled by the rate of feedwater introduction to the generator, which in turn establishes the steam generator level and the tube bundle length which is exposed to liquid secondary coolant.[42] Increasing feedwater flow increases this heat transfer length, and decreasing feedwater flow decreases the length. In other words, the amount of heat removed by an OTSG essentially is directly

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41 See Karrasch-Jones Testimony at 18 (Figure 3) and Staff Ex. 4 at 5-9 (Figure 5.6) for illustrations of a B&W once through steam generator.

42 There is a temperature difference of approximately 45°F between primary coolant entering and exiting the OTSG. The rate of heat transfer is lowest in the upper one-third of the unit and highest in the lower one-third of the unit. Tr. 591, 592 (Jones). See also, Rubin-Novak OTSG Testimony at 4.

proportional to the height of liquid on the secondary side,[43] and results in a rapid primary system response to feedwater flow changes. Karrasch-Jones Testimony at 17; Rubin-Novak OTSG Testimony at 3, 4; Staff Ex. 4 at 5-16.

68. In a U-tube steam generator, used in Westinghouse and Combustion Engineering PWRs, the primary coolant enters at the bottom, flows through inverted U-tubes that are covered by the secondary coolant, and exits at the bottom of the steam generator. The steam produced by the U-tube generator is saturated, rather than superheated. In the U-tube design, only small changes in the primary to secondary temperature difference are needed to accommodate rather large changes in the heat removal rate. Because of this and because the volume of water on the secondary side surrounding the U-tubes is larger,[44] perturbations on the secondary side do not as readily affect the behavior of the primary coolant system. Rubin-Novak OTSG Testimony at 4, 6; Staff Ex. 4 at 5-17.

69. The close coupling of the primary and secondary systems in the B&W design, combined with the relatively small liquid volume in the secondary side, creates the characteristic of the OTSG which has been referred to as "sensitivity" and "responsiveness." The design and operating characteristics of

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43 There is not, however, a one-to-one relationship. Tr. 600 (Jones).

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

the OTSG, including its responsiveness to feedwater transients and available secondary inventory, have been considered in the safety analyses for the plant. Karrasch-Jones Testimony at 16. In addition, there are clear advantages to the OTSG design. The close coupling allows the steam generator secondary to borrow energy from the primary coolant to increase load promptly and to store energy in the primary coolant to decrease load rapidly -- resulting in a nuclear steam supply system which can respond acceptably to load changes and maintain the reactor within the limits of the Reactor Protection System.[45] Karrasch-Jones Testimony at 16; Staff Ex. 4 at 5-1. The OTSG design also results in longer turbine life, an increase in plant efficiency, and favorable tube integrity compared to the U-tube design. Staff Ex. 4 at 5-18; Tr. 1201 (Novak).

70. Because of the NRC Staff's concerns, following the Three Mile Island accident, that B&W reactors appeared to be unusually sensitive to certain off-normal conditions originating in the secondary system (see paragraph 2, supra), steps have been taken at the Rancho Seco plant to minimize the potential for a similar sequence of events and to provide additional assurance that the plant will respond safely to feedwater transients. In response to IE Bulletin 79-05B (April 21, 1979), the high reactor coolant pressure trip setpoint was

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45 Mr. Comstock, a shift supervisor and senior licensed operator at Rancho Seco who has had operating experience at non-B&W PWRs, testified that the OTSG responds to feedwater transients in a more positive way for operators, giving them better control over plant parameters. CEC Ex. 37 at 8, 9.

lowered (from 2355 to 2300 psig) and the setpoint for the pressurizer power operated relief valve ("PORV") was increased (from 2255 to 2450 psig). Staff Ex. 4 at A-4. This will minimize challenges to the PORV and the possibility that the valve will stick open, cause a small-break, loss-of-coolant accident, and aggravate a transient situation. Karrasch-Jones Testimony at 22. In response to item (c) of the short-term actions required by the Commission's Order of May 7, 1979 (see paragraph 5, supra), hard-wired trips on loss of main feedwater and turbine trip were installed prior to the restart of Rancho Seco. This results in the prompt decrease in core heat generation in the event of a loss of feedwater or turbine trip, provides additional time for the auxiliary feedwater system to respond to a loss of feedwater transient, and provides additional margin to avoid a reactor coolant pressure increase which might challenge the PORV. Karrasch-Jones Testimony at 23; Rubin-Novak OTSG Testimony at 7.

71. Prior to the implementation of these post-TMI modifications a loss of main feedwater transient at Rancho Seco would have resulted in reactor coolant pressure increases to the PORV setpoint (with PORV actuation) and to the high reactor coolant pressure trip setpoint and reactor trip. Karrasch-Jones Testimony at 19, 20. As a result of the post-TMI modifications described above (paragraph 70, supra), the same transient at Rancho Seco would cause an anticipatory reactor trip on loss of main feedwater and the PORV would not

be actuated. Id. at 14, 15. Since a reactor coolant pressure excursion is not expected to occur and the PORV is not expected to open, the likelihood of conditions occurring which could aggravate a loss of feedwater has been further reduced. In addition, the anticipatory trip on loss of main feedwater actually serves to decrease primary system fluctuations as a result of secondary system upsets.[46] Karrasch-Jones Testimony at 24, 25. See also, Rubin-Novak OTSG Testimony at 7; Staff Ex. 4 at 1-2.

72. The other actions directed by the Commission in its Order of May 7, 1979, and especially the efforts to upgrade the timeliness and reliability of the auxiliary feedwater system (see section II.G, infra), have also served to decrease the probability that a feedwater transient at Rancho Seco will be aggravated by unexpected occurrences, and to increase the plant's capability to mitigate off-normal situations should such occur. Karrasch-Jones Testimony at 23-25; Rubin-Novak OTSG Testimony at 5-7.

73. Concern has also been expressed about the OTSG design with respect to the response of the primary system to secondary side overcooling transients. Secondary side overcooling resulting in depressurization of the primary system

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46 The Board disagrees, then, with the conclusion of CEC witness Webb that none of the modifications required by the Commission's Order of May 7, 1979, addressed measures to reduce the basic design sensitivities of the B&W reactor. See Webb Testimony at 12.

usually occurs because of overfeeding a steam generator, demanding too much steam from the steam generators, or introducing excessive amounts of relatively cold auxiliary feedwater into the steam generator. Staff Ex. 4 at 5-22. These transients, which are not desirable, will be examined in some detail below in the Board's findings on natural circulation and void formation. See section II.F, infra. To summarize, however, all of the analyses in this record of credible overcooling events show that adequate core cooling is maintained. See Rubin-Novak OTSG Testimony at 8, 9; Karrasch-Jones Testimony at 43-45; Tr. 1208 (Rubin).

74. In Additional Board Question 3, the Board expressed specific concern that in the event of a feedwater transient an operator may not be able to distinguish between an overcooling event and a small-break, loss-of-coolant accident ("LOCA"), or to determine the appropriate response. The record shows that the instrumentation available to the Rancho Seco operator in the control room,[47] as well as the availability of valve and pump controls in the control room, provides assurance that overcooling conditions can be recognized and controlled. Rodriguez Testimony at 30. The operator's response to a

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47 Auxiliary feedwater flow measuring instrumentation was installed at Rancho Seco, in response to the Commission's Order of May 7, 1979, to provide operators with an additional means of confirming auxiliary feedwater flow and diagnosing the flow rate. Main feedwater flow instrumentation has been available as a part of the initial plant design. Licensee's Testimony of Ronald J. Rodriguez, Etc., following Tr. 2948 ("Rodriguez Testimony"), at 29.

decreasing pressure condition will depend upon whether or not it is associated with: (a) decreasing reactor system temperature and abnormal secondary side conditions such as higher than anticipated steam generator level (symptoms of an overcooling transient), or (b) essentially stable average reactor system temperature (indicative of loss of coolant). Rodriguez Testimony at 29, 30; Karrasch-Jones Testimony at 48; CEC Ex. 37 at 32-34. The immediate operator action for the two events, however, is the same -- i.e., assure that high pressure injection has been initiated to restore reactor coolant inventory. Therefore, because of the similarity of response, the inability to differentiate the two events immediately does not compromise core cooling. Rubin-Novak OTSG Testimony at 9; Karrasch-Jones Testimony at 48.

75. Returning to the more general concern raised in Additional Board Question 3 -- the sensitivity or responsiveness of the B&W once through steam generator -- the Board finds in the record varying degrees of concern among some witnesses but no clear proposals advanced by those concerned. Mr. Webb, an engineer with the staff of the CEC, stated his concern that what he views to be the design sensitivities of B&W plants cannot be tolerated in the long term, while at the same time he advanced no solution to meet his concern. Webb Testimony at 14; Tr. 1941 (Webb). Mr. Webb, we should note, has had no experience in the design of nuclear steam supply systems, has performed no direct safety analyses of the design

of nuclear power plants (including Rancho Seco) in his work with CEC, and presented testimony here which constituted nothing more than his opinion of analyses performed by the District, Babcock & Wilcox, the NRC Staff and others. Tr. 1818, 1851, 1982 (Webb). The California Energy Commission also presented the testimony of Dr. Harold W. Lewis, Professor of Physics since 1964 at the University of California, Santa Barbara. Prepared Direct Testimony of Dr. Harold W. Lewis Concerning Natural Circulation Cooling, following Tr. 477 ("Lewis Testimony"). Dr. Lewis has served as Chairman of the American Physical Society's Study Group on Light Water Reactor Safety, and Chairman of the NRC's Risk Assessment Group, and he is currently a member of the NRC's Advisory Committee on Reactor Safeguards and the Advisory Council of the Institute for Nuclear Power Operations. Id.; Tr. 478-450 (Lewis). Dr. Lewis, whose testimony includes the results of his own analysis, concluded ". . . that what hardware problems there were have been largely remedied by the series of orders that have been mandated by the NRC in the aftermath of TMI . . .". Lewis Testimony at 12; Tr. 494-495, 526 (Lewis).

76. There are no criteria, which have been called to our attention, by which the Board might assess the acceptability of the overall response of the B&W system. See Staff Ex. 4 at 5-25 to 5-28. The Staff believes that the actions already undertaken provide the necessary assurance that Rancho Seco will respond safely to loss of feedwater transients, and that

overcooling events at Rancho Seco will not result in loss of adequate core cooling or exceed the fuel damage criteria. Rubin-Novak OTSG Testimony at 7, 8; Tr. 1194 (Rubin). The NRC's B&W Reactor Transient Response Task Force, in its final report, stated that "replacement of the OTSG does not appear to be practical or a necessary action for operating plants, especially when weighed against certain other safety advantages of the OTSG." Staff Ex. 4 at 2-2. One of the general findings of that task force is that while it requires a highly interactive and responsive control system, <u>the once-through steam generator design is basically sound</u>. Staff Ex. 4, transmittal memorandum, May 1, 1980, Robert L. Tedesco to Harold R. Denton, at 1.

77. The NRC Staff is continuing its review of the sensitivity or responsiveness of the B&W design. Rubin-Novak OTSG Testimony at 7, 8; Staff Ex. 4 at 1-3. The Board encourages this effort. The Board finds, however, that the OTSG design is basically sound, and that its responsiveness and characteristically smaller operating secondary inventory have been integrated with the design requirements of supporting control and safety systems. Moreover, the Board finds that the modifications directed by the Commission in its Order of May 7, 1979, have reduced the sensitivity of the OTSG, so that gross disturbance of the primary system is not inevitable for feedwater transients as we indicated in Additional Board Question 3.

D. Anticipatory Reactor Trips

Board Question H-C 9: Has the reliability of the recently installed control grade reactor trip on 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) had been called on to respond four times and had failed once:

a. Is this typical of performance by control grade trips?
b. What are the safety implications for operation of Rancho Seco before such trips are upgraded?

78. Prior to the TMI-2 March 28, 1979 accident, B&W PWRs did not have anticipatory reactor trips upon loss of main feedwater or upon turbine trips. NRC Staff Testimony of Dale F. Thatcher Relative to Direct Initiation of Reactor Trip Upon the Occurrence of Off-Normal Conditions in the Feedwater System (Board Question 9 and Additional Board Question 1), following Tr. 1163 ("Thatcher ART Testimony"), at 2. Since these anticipatory reactor trips were not available, loss of main feedwater would cause the reactor to trip only when the reactor coolant system pressure rose to the reactor high pressure trip setpoint. Thatcher ART Testimony at 2; Licensee's Testimony of Robert A. Dieterich, Etc., dated February 11, 1980, following Tr. 1988 ("Dieterich Testimony"), at 14. In order to reach the high pressure setpoint, the reactor coolant system pressure had

to rise above the then existing setpoint for the pilot operated relief valve ("PORV"), so the PORV was likely to open every time a loss of main feedwater occurred. Thatcher ART Testimony at 2, 3. A similar scenario might take place in the event of a turbine trip, the only difference being that the ICS would attempt a reactor power runback which might prevent a reactor trip from taking place. Karrasch-Jones Testimony at 10; Dieterich Testimony at 14. Regardless of whether a reactor trip took place, a turbine trip might result in sufficiently increased reactor coolant system pressure to open the PORV. Thatcher ART Testimony at 2, 3.

79. During the TMI-2 accident, there was an initial loss of main feedwater and a turbine trip. Reactor trip occurred 8 seconds after these events, since it took that time for the reactor coolant system pressure to reach the high pressure trip setpoint. Thatcher ART Testimony at 2. During the interval, the PORV opened and, as is well known, failed to reclose, leading to a loss of reactor coolant inventory.

80. In order to alleviate the concern that the PORV might open every time a loss of main feedwater or turbine trip occurred, the NRC Staff issued IE Bulletin 79-05B on April 21, 1979, which among other things required the owners of B&W PWRs to reduce the high reactor coolant pressure setpoint from 2355 psig to 2300 psig and raise the setpoint for automatic opening of the PORV from 2255 psig to 2450 psig. Thatcher ART Testimony at 3. These actions minimize the likelihood of

automatic actuation of the PORV by having a reactor trip occur earlier (at a lower pressure), and thus limiting the subsequent reactor coolant system pressure rise. Thatcher ART Testimony at 3.

81. The change in setpoints for the high pressure trip and the PORV opening is the principal mechanism relied upon to avoid PORV actuation during transients. Tr. 1641-1643 (Novak). To provide an additional margin to the new PORV opening setpoint, and further ensure against PORV challenges, Licensee was required by the Commission Order of May 7, 1979, to "[i]mplement a hard-wired control-grade reactor trip that would be actuated on loss of main feedwater and/or turbine trips" as a condition to the restart of the Rancho Seco facility. 44 Fed. Reg. at 27780 (1979); Thatcher ART Testimony at 3; Tr. 1079 (Karrasch); Tr. 1643 (Thatcher, Novak).

82. An additional, and perhaps more important, reason for implementing hard-wired reactor trips upon loss of main feedwater and turbine trip was to minimize the transient response of the plant to secondary system upsets. Tr. 1079 (Karrasch); Tr. 1641 (Novak). As the Board found above (paragraph 70), the presence of the anticipatory trip on loss of main feedwater results in a prompt decrease in core heat generation (8 to .0 seconds earlier than the high pressure trip would provide) so that the steam generator inventory is not depleted as rapidly as it would be if a trip had been delayed until the high pressure setpoint was reached. Tr. 928, 929

(Karrasch). This prompt decrease in core heat generation adds 3 to 4 minutes to the potential steam generator dry-out time. Tr. 588, 589 (Karrasch); Tr. 1443, 1444 (Rubin); Tr. 1753, 1754 (Matthews). In turn, the increased steam generator dry-out time results in at least 3 to 4 additional minutes (and perhaps more) in which to re-establish a heat sink in the system. Tr. 1445, 1446 (Novak).

83. Licensee's compliance with the Commission Order of May 7, 1979, in implementing hard-wired control-grade reactor trips on loss of main feedwater and on turbine trip was verified by the Staff prior to the restart of the facility. See Staff Evaluation at 14-16.

84. The hard-wired trips installed by Licensee are "control grade", i.e., they do not meet the design criteria of the reactor protection system and are therefore not "safety grade."[48] Thatcher ART Testimony at 5, 6. The Commission Order of May 7, 1979, requires, as a long-term modification, that the reactor trips on loss of main feedwater and/or turbine trip "be upgraded so that the components are safety grade." 44 Fed. Reg. at 27779 (1979); Thatcher ART Testimony at 5. When the record in this case closed, Licensee's proposed design for upgrading the trip to safety grade at Rancho Seco had been

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48 The "safety grade" design criteria, contained in the Institute of Electrical and Electronics Engineers (IEEE) Standard 279, addresses design requirements such as single failure, testability, qualifications, independence and automatic removal of operating bypasses. Thatcher ART Testimony at 6; Tr. 1653, 1654 (Thatcher).

approved by the Staff and Licensee was proceeding to implement this modification. Thatcher ART Testimony at 6; Dieterich Testimony at 15.[49]

85. There is ample evidence on the record that the control-grade anticipatory reactor trips installed by Licensee are reliable and that it is safe to operate the Rancho Seco facility pending their modification to meet safety-grade standards.[50] The circuitry utilized on these trips has been designed to the highest industry standards to provide high reliability of operation and is comparable in quality and reliability to other control-grade devices installed at Rancho Seco, such as the turbine generator controls, which have proved extremely reliable in over five years of operation of the facility. Dieterich Testimony at 15. Moreover, the reliability of the control-grade anticipatory trips at Rancho Seco has been demonstrated by operating successfully in three turbine trips and by performing without failure at tests conducted every month since they were installed. Dieterich Testimony at 15; Tr. 2332, 2333 (Dieterich).

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49 Upgrading of these reactor trips to safety grade requires adding redundant power supplies and sensors, and installing seismically qualified instrumentation and power cabling. Tr. 2125 (Dieterich).

50 It is, of course, preferable from a reliability standpoint to use safety-grade equipment to implement these trips. Tr. 1653 (Thatcher); Tr. 2333-2335 (Dieterich). The increased reliability of these trips, when made safety grade, may enhance to some extent the defense in depth of the reactor. Tr. 1654, 1655 (Novak). The extent of any safety enhancement provided by the safety-grade trip will depend upon the implementation of other plant improvements. Id.

86. Additional Board Question 1 expresses some concern about the reliability of the control-grade anticipatory trips because it was learned that one trip failure had been experienced in the first four times the trips were called upon to function.[51] Thatcher ART Testimony at 8. Since that time, however, there have been eight additional activations of the anticipatory trips, all successful. Thus the current operational record (11 successes out of 12 challenges) demonstrates the reliability of the control-grade anticipatory trip. Tr. 1126, 1127 (Karrasch); Tr. 1711, 1712 (Thatcher); Tr. 2128, 2129 (Dieterich).[52]

87. Even if the control-grade anticipatory trips were to fail when called upon at Rancho Seco, there would be only minor safety implications from such failure. Thatcher ART Testimony at 9; Dieterich Testimony at 16; Karrasch-Jones Testimony at 27; Tr. 2127, 2128 (Dieterich). The effect of such a failure, as demonstrated by the Arkansas experience, would be merely to delay the reactor trip for about 8 seconds

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51 The one trip failure that was experienced occurred at the Arkansas Plant 1, and occurred because there was a loose connection in one circuit so that when the turbine tripped the signal was not conveyed to the reactor trip mechanism. Tr. 1712 (Thatcher). At Arkansas, the reactor tripped on high pressure 8 seconds after the turbine tripped; the PORV was not actuated. Thatcher ART Testimony at 9.

52 The early failure experienced at Arkansas was attributed by one witness to the process of operator familiarization and maintenance problem "debugging" that takes place after a system is newly installed. Tr. 2128 (Dieterich); Dieterich Testimony at 16.

and to decrease somewhat the steam generator boil-dry time in the case of a loss of main feedwater trip. Tr. 1713, 1714 (Thatcher). The plant is capable of responding safely to the transient if the trips fail to function.[53] Karrasch-Jones Testimony at 27; Thatcher ART Testimony at 9. Therefore, operation of Rancho Seco with the control-grade anticipatory trips, until they are upgraded to safety grade, is acceptable.

E. 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 the greater capacity in these pieces of equipment?

88. The pressurizer is a cylindrical vessel with the same design pressure as the reactor coolant system and is considered to be an integral part of that system. Karrasch-Jones Testimony at 28; NRC Staff Testimony of Philip R. Matthews [on] Adequacy of the Pressurizer and Pressurizer Relief Tank Size (Board Question 21), following Tr. 1163, ("Matthews Pressurizer Testimony"), at 2, n. 1. The purpose of the pressurizer is to provide a gas volume to accommodate

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53 In particular, a failure of the anticipatory reactor trips would not affect the reliability of the AFW supply because the trips do not affect the AFW system in any way. Tr. 2112 (Dieterich).

pressure and density changes in the reactor coolant system during normal operating conditions as well as during anticipated transients. Karrasch-Jones Testimony at 28; Matthews Pressurizer Testimony at 2. During normal operating conditions, the pressurizer is partially filled with water in saturation with steam. A decrease in reactor coolant system temperature and pressure causes some of the water in the pressurizer to flash to steam, thus assisting to maintain reactor coolant system pressure. Conversely, an increase in reactor coolant system temperature and pressure causes water from the reactor vessel to be sprayed into the steam space of the pressurizer to condense steam and reduce pressure. Matthews Pressurizer Testimony at 3.

89. The pressurizer capacity at Rancho Seco was selected in accordance with the NRC's General Design Criteria.[54] The maximum pressurizer volume was determined by adding the minimum volume of reactor coolant to be maintained following a reactor trip, the maximum volume change to be expected following such a trip from full power, the maximum volume change to be expected during normal operating conditions, and the maximum expected increase in volume due to a turbine trip.

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54 The NRC criterion applicable to pressurizer sizing is General Design Criterion 15 of Appendix A to 10 C.F.R. Part 50, which states that the reactor coolant system (including the pressurizer) shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during normal operating conditions or anticipated transients. Matthews Pressurizer Testimony at 4, 5.

The result of this addition, increased by an appropriately conservative engineering factor, gave the total design volume of 1500 cubic feet for the Rancho Seco pressurizer. Tr. 784 (Karrasch); Karrasch-Jones Testimony at 29, 30.

90. The design basis for pressurizer sizing is to select a size that avoids the pressurizer emptying or becoming "solid" (full with fluid) during normal operations or during anticipated transients.[55] Karrasch-Jones Testimony at 29, 30; Tr. 784, 785 (Karrasch, Jones). The rationale behind these objectives is that the pressurizer is the principal means of maintaining pressure control in the reactor coolant system; however, as long as the pressurizer remains empty or solid it cannot be used for maintaining pressure control. Tr. 786, 787 (Karrasch, Jones).

91. It is uncontested that the Rancho Seco pressurizer meets the current NRC design criteria and that its size is adequate to accommodate reactor coolant fluid volume changes during normal conditions and anticipated transients. Tr. 1460, 1461 (Matthews); Matthews Pressurizer Testimony at 5,

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55 An anticipated transient is one leading to a reactor trip without any subsequent equipment failure. Tr. 780 (Karrasch). Anticipated transients have a frequency of occurrence of more than one per year. Tr. 780, 781 (Karrasch). Anticipated transients considered in sizing the Rancho Seco pressurizer included turbine trips and loss of feedwater transients. Karrasch-Jones Testimony at 29; Tr. 784 (Karrasch). All or most of the conditions leading to overcooling or undercooling situations result from some failure beyond the initiating event that caused the reactor trip and therefore are not regarded as anticipated transients but as "off-normal" conditions. Tr. 770 (Karrasch); Tr. 782-784 (Jones).

8 and 13; Karrasch-Jones Testimony at 31. There are, however, a number of transients or accident conditions that can theoretically result in emptying the pressurizer or causing it to go solid.[56] For instance, emptying of the pressurizer is possible for short periods of time during a depressurization or overcooling transient. See paragraph 109, infra. Emptying of the pressurizer can also occur in an overheating transient in which the "feed and bleed" mode of core cooling (see paragraph 121, infra) has been exercised for an extended period of time and then there is a start of AFW delivery. Tr. 1128 (Jones).[57] 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).[58]

92. Although the above-cited conditions may in theory lead to the pressurizer emptying, analysis of operating

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56 A solid pressurizer may result from an off-normal condition such as the non-nuclear instrumentation failure which occurred during the CR-3 transient of February 26, 1980. Tr. 1685 (Matthews).

57 In such a scenario, HPI would be actuated on reactor coolant low pressure and would cause the pressurizer to refill automatically. Tr. 1129 (Jones).

58 Thus, conditions such as existed during the early phases of the TMI-2 accident (i.e., loss of coolant through a stuck-open PORV) were not included in sizing the pressurizer. Tr. 1681, 1682 (Matthews).

data from all B&W PWRs shows that in every instance of a reactor trip, including those involving overcooling events, the pressurizer liquid volume was maintained, i.e., the pressurizer did not empty.[59] Tr. 771-773 (Karrasch); Tr. 775-777 (Karrasch, Jones); Licensee's Supplemental Testimony of Bruce A. Karrasch and Robert C. Jones, dated February 26, 1980, following Tr. 535, at 2. Thus, the size of the pressurizer at B&W reactors has been proven adequate by the operating experience. Tr. 785 (Karrasch).

93. The pressurizer relief tank (also "pressurizer quench tank" or "PRT") is a vessel located within the containment and which condenses, cools and collects steam discharged from the pressurizer overpressure protection valves (the PORV and the two code safety valves). Matthews Pressurizer Testimony at 4. The PRT is not part of the reactor coolant system pressure boundary, but is an operational convenience whose purpose is to accommodate the fluid discharges produced in the few instances in which the PORV or code safety valves may lift. Tr. 943 (Karrasch); Karrasch-Jones Testimony at 30, 31.

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59 On a number of occasions, pressurizer level indication in the control room was lost, although pressurizer liquid volume existed. Tr. 774 (Karrasch). Staff Exhibit 4 (NUREG-0667) recommends that following a reactor trip the pressurizer level should remain on scale. Staff Ex. 4 at 5-13. This result might be achieved in a number of ways, such as expanding the range of pressurizer level indication or changing the setpoint on the turbine bypass valves; these possible solutions would not require increasing the size of the pressurizer. Tr. 1462, 1463 (Matthews).

94. The PRT is protected against overpressure by a rupture disc sized for the total combined relief capacity of the PORV and the two code safety valves. If steam discharge into the PRT exceeds the disc setpoint, the rupture disc will rupture to avoid the failure of the entire tank. Tr. 1691, 1692 (Matthews); Matthews Pressurizer Testimony at 4.

95. The PRT size was determined in accordance with the sizing criteria of the Staff's Standard Review Plan, Section 5.4.11. Matthews Pressurizer Testimony at 7. The basis for PRT sizing was to accommodate the total steam discharge and discharge rate from the maximum pressure increase that the code safety valves will be subjected to during a design basis accident, which occurs during a control rod withdrawal accident from zero power. Tr. 942 (Karrasch); Matthews Pressurizer Testimony at 6. This accident bounds all other design basis accidents, including loss of feedwater transients, because it results in the largest discharge to the PRT through the PORV and the code safety valves. Karrasch-Jones Testimony at 31.

96. The PRT has a 500 psig design pressure, but the the rupture disc is set to rupture at lower pressures. Tr. 1690 (Matthews). The PRT at Rancho Seco has a volume of 1100 cubic feet and operates with about half of that volume filled with water and the other half filled with nitrogen gas. Tr. 1691 (Matthews). Thus, the PRT is usually not full with fluid when the rupture disc ruptures, for the entry of water[60] from

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60 At Rancho Seco there is a sparger under water at the (footnote continued next page)

the pressurizer compresses the nitrogen cushion and makes the disc fail while there is still a gas phase within the tank. Tr. 1469, 1470 (Matthews).

97. If the transient terminates before the rupture disc in the PRT ruptures, the contents of the tank are cooled to normal temperatures by a cooling water system. Tr. 1694 (Matthews); Matthews Pressurizer Testimony at 4. If the rupture disc ruptures, the contents of the PRT are deposited in the containment building floor and collect in the containment building sump. Tr. 954 (Karrasch); Tr. 1468 (Novak).[61]

98. It is not expected that there will be any fluid discharge to the PRT at Rancho Seco during normal operating conditions or during feedwater transients. Karrasch-Jones Testimony at 31. This is because the modifications made after the TMI-2 accident (changing the PORV and reactor high pressure trip setpoints, and adding an anticipatory trip on loss of feedwater) will result in very rare actuations of the PORV in the future.[62] Tr. 935 (Karrasch); Tr. 1688 (Novak);

(continued)
entrance of the PRT so that the steam released by the PORV and safety valves condenses (at least partially) to water and loses energy. Tr. 1690-1692 (Matthews).

61 The spillage of PRT water onto the containment building floor is not in itself a safety concern as long as the water remains confined to the containment building. Tr. 1686, 1770-1771 (Novak).

62 There were on the order of 149 reactor trips with documented PORV openings at B&W PWRs prior to the TMI-2 accident, and only one reported incident thereafter. Tr. 1689 (Capra); Staff Ex. 4 at 4-15. Three of these incidents resulted in rupture of the PRT rupture disc. Tr. 1687 (Novak).

Karrasch-Jones Testimony at 24; Matthews Pressurizer Testimony at 9, 10. And, in any case, the PRT is sized to accommodate the discharge from most anticipated and off-normal events.[63]

99. In response to the inquiry, then, in Board Question H-C 21, the Board finds that both the pressurizer and the PRT at Rancho Seco meet the NRC sizing criteria and are adequate to accommodate normal and anticipated transient conditions and bounding design basis accidents. In addition, there are no anticipated transients (or transients already experienced elsewhere) which call for greater capacity in these pieces of equipment.

F. Natural Circulation, Void Formation, and Small-Break, Loss-of-Coolant Accidents

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 1-4: Will the failure of safety and/or relief valves in the Rancho Seco primary system result in an unsafe condition despite the modifications and actions of Subparagraphs a-e?

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63 The PRT is not designed to accommodate the flow through a PORV or a code safety valve that remains open and discharges fluid continuously. Tr. 1682 (Matthews); Matthews Pressurizer Testimony at 7. Such an event, which is a type of loss of coolant accident, would result in a very fast pressure rise in the PRT and failure of the rupture disc in a matter of seconds. Tr. 1695-1696, 1771 (Novak). The PRT would also have to be failed in a "feed and bleed" mode of operation in order to provide a discharge path for the reactor coolant fluid. Tr. 1771, 1772 (Novak). On the other hand, it is not anticipated that the PRT will fail in the event of HPI actuation during an overcooling transient, because HPI will be cut out after a short period of operation (once a 50°F subcooling margin is achieved). Tr. 945-948 (Jones).

Board Question
CEC 1-7: Do the operator training actions responding to Subparagraph (d) of Subparagraphs a-e for Rancho Seco fail to give sufficient attention to providing appropriate analytical bases for operator actions?

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?

Additional Board
Question 2: 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
H-C 24: What features of the Rancho Seco system serve to prevent or control bubble formation in the primary system following a loss-of-feedwater transient?

100. Board Questions CEC 1-2, CEC 1-4, CEC 1-7, CEC 1-10, H-C 24 and Additional Board Question 2 address the use of natural circulation (or natural convection) and other cooling modes to assure adequate core cooling during various feedwater and related transient scenarios, and the adequacy of short-term action (d) of the Commission's Order of May 7, 1979 (see paragraph 5, _supra_), which required the completion of analyses

of small-break, loss-of-coolant accidents and the development and implementation of operating instructions to define operator action.

101. Natural circulation is the process by which coolant is circulated in the primary system of a PWR hen all reactor coolant pumps are inoperative. The natural circulation phenomenon occurs as a result of design features inherent to plants such as Rancho Seco. Removing core decay heat from the primary coolant with the steam generators (and increasing its density) at a higher elevation than the elevation at which heat is added in the core (decreasing its density) produces a force (from the density change) which maintains a continuous flow in the primary loop. Karrasch-Jones Testimony at 33, 34 and 36 (Figure 4); NRC Staff Testimony of Paul E. Norian on Natural Circulation (Board Question CEC 1-2), following Tr. 1163 ("Norian Circulation Testimony"), at 2, 3.

102. Analyses have been performed, utilizing conservative assumptions over a wide range of plant conditions, to determine that natural circulation is adequate to maintain core cooling when all of the reactor coolant pumps are inoperative. Natural circulation has also been tested at other operating B&W plants. The testing confirmed that natural circulation can be initiated and maintained over a wide range of plant conditions, and demonstrated that the design analyses conservatively predict the natural circulation capabilities of the plants.[64] Karrasch-Jones Testimony at 32; Norian

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64 While Rancho Seco is one of the "lowered-loop" B&W plants (footnote continued next page)

Circulation Testimony at 3. This analysis and testing show a temperature difference of between 20° and 40°F for the core and in the steam generators, which results in a natural circulation flow rate between 2% and 4% of the normal flow rate with all four reactor coolant pumps in operation.[65] Karrasch-Jones Testimony at 34. Further, unplanned occurrences of natural circulation have been experienced at B&W operating plants and in all of these events, where the reactor coolant pumps were inoperative, natural circulation core cooling maintained the plant in a safe condition. Id. at 32, 33, 35. Perhaps the most prominent example of verification of natural circulation in a B&W plant has been at Three Mile Island where, since April 27, 1979, natural circulation with one steam generator has been removing the core decay heat. Id. at 35, 37; Norian Circulation Testimony at 3.

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(see Staff Ex. 4 at 5-4, Figure 5.1), the low elevation of the steam generator relative to the reactor vessel does not prevent effective natural circulation. Norian Circulation Testimony at 5, 6. The newer B&W plants, with a "raised loop" configuration (see Staff Ex. 4 at 5-6, Figure 5.3), have a higher elevation of the steam generator relative to the reactor vessel, thus providing a greater driving head for natural circulation. Both analyses and testing of natural circulation flow rates at a raised-loop plant, however, have shown that a significant increase (doubling) in the driving head results in an increase in flow rate which is not large (10% to 20%). Tr. 912, 913 (Karrasch).

65 The flow rate of the primary coolant at Rancho Seco with all four reactor coolant pumps in operation is approximately 140 to 150 million pounds per hour, whereas the required natural circulation flow rate to assure core cooling is roughly one million pounds per hour (or 0.6 to 0.7%). Tr. 915, 916 (Karrasch).

103. Operator action is not required at Rancho Seco to establish natural circulation cooling following the anticipated transient loss of main feedwater and loss of forced reactor coolant flow. The plant design requires only that auxiliary feedwater cooling be established, which is an automatic plant function. The operator can monitor these basic parameters to determine that natural circulation has been established: reactor coolant temperature, reactor coolant pressure, steam generator level and pressurizer level. These parameters indicate reactor coolant subcooling and steam generator heat removal. Karrasch-Jones Testimony at 37; Rodriguez Testimony at 52, 53.

104. There is no indication that the level of understanding of natural circulation by the TMI-2 operators contributed to the severity of the accident at that facility. Norian Circulation Testimony at 5. The problem in achieving natural circulation at Three Mile Island resulted from voiding in the primary system created by the failure to maintain reactor coolant inventory. The voids were already present at the time forced circulation was terminated. Id.; Karrasch-Jones Testimony at 37; Lewis Testimony at 6, 11.

105. Additional operating procedures and training on the establishment of natural circulation cooling in the event forced circulation is lost have been provided to the Rancho Seco operators since the Three Mile Island accident. These procedures describe specific parameters which operators can

monitor and provide specific direction on controlling these parameters in the event of a loss of forced circulation. Rodriguez Testimony at 52. In its review of Licensee's compliance with the short-term actions required by the Commission's Order of May 7, 1979, the NRC Staff audited licensed operators at Rancho Seco and initially found a deficiency in the operators' knowledge concerning verification of natural circulation. Consequently, each licensed operator received additional training on this subject by the Rancho Seco training staff and by General Physics Corporation, a training consultant to Licensee. Staff Evaluation, at 23, 24. A subsequent audit by the NRC Staff revealed no deficiencies. See paragraph 125, infra.

106. Single-phase (no voids) natural circulation, which we have discussed above in paragraphs 101-105, is the normal cooling mode that would occur following the tripping of reactor coolant pumps during an anticipated operational transient such as loss of feedwater. During such a transient the reactor coolant is maintained in a subcooled condition and no steam or other gases are expected to be formed in the primary system other than in the pressurizer steam space. Karrasch-Jones Testimony at 42; Norian Circulation Testimony at 3. Abnormal conditions associated with feedwater transients, however, could result in void formation in the reactor coolant system. Karrasch-Jones Testimony at 42. Steam voids would form in the Rancho Seco primary system whenever the reactor

coolant pressure is reduced below the saturation pressure for the fluid, resulting in the flashing of some fluid into steam. Non-condensible gases could accumulate in the primary system following a postulated LOCA. NRC Staff Testimony of Paul E. Norian on Bubble Formation (Board Question CEC 1-10 and Board Question 24), following Tr. 1163 ("Norian Bubble Testimony"), at 2, 3.

107. Natural circulation of the primary system fluid can also occur under two-phase (voided) conditions. If the fluid contains only limited voids, the liquid with the entrained voids will continue to circulate around the system. As the primary system voids increase, the steam will tend to separate from the liquid and would eventually result in the core being covered with a boiling liquid pool. The steam generated is transported to the steam generator and condensed. The condensed liquid travels back to the core to replenish the liquid that is being converted to steam. The resultant flow is another mode of natural circulation, which has also been referred to as reflux boiling. Norian Circulation Testimony at 3. See also, Lewis Testimony at 9. It is possible to interrupt natural circulation if the primary system contains significant non-condensible gas or steam such that the U-bends at the top of the hot legs are blocked. Norian Circulation Testimony at 7; Norian Bubble Testimony at 3, 4; Lewis Testimony at 10, 11.

108. Void formation in the Rancho Seco reactor coolant system might occur in conjunction with a feedwater

transient in the following situations: (a) a reactor trip followed by an overcooling transient (e.g., excessive feedwater addition) causing a potential loss of the pressurizer liquid volume; (b) a reactor trip followed by an overheating transient (e.g., loss of main feedwater followed by delayed actuation of auxiliary feedwater) causing steam to be created in the primary system; and (c) a loss-of-coolant accident causing reactor coolant inventory loss and pressure reduction (e.g., loss of feedwater with a small-break LOCA). Karrasch-Jones Testimony at 43; Lewis Testimony at 10. Each of these three classes of events will be examined in turn to determine if the Rancho Seco plant will respond in a safe manner.

109. Overcooling of the Rancho Seco primary system could occur from excessive addition of feedwater to the steam generators following a reactor trip. This could be caused, for example, by a failure which causes main feedwater pumps and valves to deliver full feedwater flow to the steam generators when the demand for feedwater flow is very low following a reactor trip. In such events, the excessive reduction in reactor coolant temperature causes a contraction of the system inventory and a decrease in pressurizer level and reactor coolant pressure. If the feedwater overfilling continues, reactor coolant pressure decreases to the setpoint for actuation of high pressure injection and the pressurizer may empty for a short period. Displacement of a portion of the pressurizer steam space into the reactor coolant piping and

possible flashing of the fluid within the system could then result in void formation. Karrasch-Jones Testimony at 43, 44.

110. Licensee has analyzed the event described above in paragraph 109 to evaluate the potential for, and impact of, void formation. The analysis shows that while the pressurizer may empty for a short period if feedwater overfeed continues, the steam from the pressurizer is condensed as it mixes with the colder fluid in the reactor coolant piping and no voids are formed in the loops. Consequently, if the reactor coolant pumps are inoperative natural circulation will not be interrupted. In addition, the results of the analysis indicate that high pressure injection is adequate to offset the reactor coolant volume contraction and to restore pressurizer level. This analysis bounds an overcooling transient caused by excess auxiliary feedwater addition. Id. at 44, 45; Tr. 1021-1023 (Jones).

111. CEC witness Webb relied upon, for his testimony on the effect of overcooling events on natural circulation, a transcript of an ACRS meeting at which a Brookhaven analysis of overcooling was discussed. See Webb Testimony at 11; Tr. 1864-1865, 1912 (Webb). While that analysis is not in the record, we were able to garner some information about it from witnesses for Licensee and the NRC Staff who were familiar with the work. Unlike Licensee's analysis, the Brookhaven study concludes that there will be void formation in the primary system (in the upper head of the reactor vessel, but not in the

hot leg where it might inhibit natural circulation). Brookhaven agrees with Licensee, however, that adequate core cooling is maintained. Tr. 728-730 (Jones). The Brookhaven work analyzed a different transient or initiating event than did Licensee. The Brookhaven study starts with a reactor trip and turbine trip; fails the integrated control system causing continuation of full main feedwater flow through both steam generators; initiates auxiliary feedwater flow roughly one second after the reactor trip; and malfunctions the steam bypass system when it becomes operational. The two studies also employed different computer codes. Tr. 733-734, 1093-1095 (Jones). NRC Staff witness Novak, who presented the Brookhaven work to the ACRS, testified that the use of the IRT code by Brookhaven for the specific transient analyzed was a misapplication because the model does not adequately treat phase differences and density differences. Tr. 1416, 1417 (Novak). In addition, the study did not properly represent the transient the Staff intended to analyze. Id. No single failure could result in the event sequence analyzed by Brookhaven. Tr. 733, 1094 (Jones); Tr. 1418, 1419 (Novak). In the Staff's view, the transient analyzed by Brookhaven is not a reasonably credible event. Tr. 1418 (Novak).

112. While a feedwater transient causing excessive feedwater addition to the steam generators is not desirable, it is clear that such a transient does not result in void formation in the reactor coolant system and does not interrupt

natural circulation. Karrasch-Jones Testimony at 45. Even for severe overcooling accidents (steam line breaks) where some void formation is expected to occur, adequate natural circulation is maintained to keep the core cooled. Tr. 1093-1095 (Jones); Tr. 1324 (Rubin); Tr. 1325 (Norian).[66] Consequently, the Board finds that there is reasonable assurance that the Rancho Seco plant will respond safely to overcooling events.

113. The second class of events to be examined for potential void formation is overheating transients. Overheating of the primary system could occur if multiple equipment failures reduce secondary side cooling capability below that required to remove core decay heat following a reactor trip. In such an event, the primary coolant temperature could increase and cause saturated conditions, and the system pressure could rise to the pressurizer relief and/or safety valve setpoints. Steam would be created within the hot leg piping and core outlet regions, and would then be dispersed throughout the primary system unless corrective action (restoration of feedwater and/or actuation of high pressure injection) is initiated. Licensee has performed analyses for a loss of main feedwater with various time delays for the delivery of auxiliary feedwater. These analyses show that adequate core cooling will be maintained if auxiliary feedwater is provided

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66 The Staff witnesses relied upon, in addition to the analyses performed by Licensee and Brookhaven, a third analysis conducted by the NRC Staff itself. Tr. 1325 (Norian).

to the steam generators or high pressure injection is delivered to the reactor coolant system within 20 minutes. Karrasch-Jones Testimony at 45-46, 51, 58; Staff Ex. 2 at 2-4. CEC witness Lewis essentially confirmed this conclusion with his own rough calculations, which show that in the case of a reactor trip associated with a loss of all feedwater, and with no other heat losses from the primary system (such as through a relief valve), the operators have between 10 and 20 minutes available to initiate an alternate cooling mode. Lewis Testimony at 4, 5. In answer, then, to the specific inquiry made in Board Question H-C 24, the Board finds that, given the reliability of the auxiliary feedwater system at Rancho Seco (see section II.G, infra), a loss of all feedwater is very unlikely and, even if such an event were to occur, that the operators have sufficient time to undertake corrective action to prevent bubble formation from threatening the adequacy of core cooling.

114. For the third class of events identified, postulated loss-of-coolant accidents, void formation in the reactor coolant system is a predicted result of the analysis required by applicable NRC regulations. The loss of reactor coolant inventory results in a reduction in reactor coolant pressure, and voids are formed as saturated coolant conditions are reached. Licensee had conducted LOCA analyses, prior to the Three Mile Island accident, using the Emergency Core Cooling System ("ECCS") evaluation model which satisfies the

requirements of Appendix K to 10 C.F.R. Part 50, for the full range of potential loss-of-coolant accidents. The results of these analyses, which include the effects of steam accumulation in the primary system, meet the criteria of 10 C.F.R. § 50.46 and show that adequate core cooling is maintained due to the automatic initiation of emergency core cooling. Karrasch-Jones Testimony at 46-47, 50-51. See also, Tr. 1035-1037, 1149 (Jones); Tr. 1754 (Norian).

115. Subsequent to the Three Mile Island accident and in response to the Commission's Order of May 7, 1979, Licensee performed additional analyses of small-break LOCAs.[67] The following are among the more significant events analyzed. In the event of a small-break LOCA with loss of all feedwater, the analysis shows that ECCS may not be actuated automatically. In this situation, the results show that operator action within 20 minutes either to establish auxiliary feedwater flow (which will in turn result in automatic ECCS actuation) or to actuate high pressure injection manually assures that the core remains

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67 Slow system depressurization resulting from small-break LOCAs had not previously received detailed analytical study comparable to that devoted to large breaks. Typically, the smallest break size analyzed was one that would produce system depressurization without uncovering the core, in accordance with the single failure criterion and other requirements imposed by Appendix K to 10 C.F.R. Part 50. While these analyses were sufficient to show compliance with 10 C.F.R. § 50.46, in the Staff's view they did not provide the information needed for operator action following a small break. Staff Ex. 2 at 1-1. The additional analyses performed go beyond the scope of Appendix K to 10 C.F.R. Part 50 in order to develop an improved set of operator guidelines. Tr. 1036, 1037 (Jones).

covered and adequately cooled. The analyses also show that automatic initiation of high pressure injection is sufficient to assure adequate core cooling in the case of a loss of main feedwater followed by a stuck open PORV and in the case of a stuck open PORV followed by a loss of all feedwater. Karrasch-Jones Testimony at 51-52, 59-61. See also, Staff Evaluation at 16-18.

116. Analyses performed by Licensee show that if the reactor coolant pumps operate continuously throughout a LOCA, or if they are tripped promptly upon receipt of a low reactor coolant pressure safety signal, adequate core cooling is provided for all break sizes. While continued pump operation provides forced circulation cooling of the core, the analyses show that for certain break sizes it also causes more fluid inventory to be discharged through the break than would otherwise occur. Because of this greater inventory loss the fluid in the reactor coolant system will evolve to a high void fraction. If for some reason the reactor coolant pumps are tripped after a high void fraction is reached, for this limited range of break sizes available inventory may not be sufficient to keep the core covered. Consequently, the NRC issued IE Bulletin 79-05C to require immediate tripping of the reactor coolant pumps upon reactor trip and initiation of high pressure injection caused by low reactor coolant system pressure.[68]

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68 This phenomenon is not unique to B&W plants and the same requirement has been imposed upon all PWR operating plants. Tr. 1073, 1074 (Jones).

Karrasch-Jones Testimony at 53-54, 63; NRC Staff Testimony of Paul E. Norian on Logic for Reactor Coolant Pump Trip in Small-Break LOCA (Additional Board Question 2), following Tr. 1163 ("Norian Trip Testimony"), at 2-4; Rodriguez Testimony at 27.

117. Board Question CEC 1-4 inquires about the effects of failure of the safety and/or relief valves on the Rancho Seco primary system. First, because of modifications made at Rancho Seco -- the revised PORV and high pressure trip setpoints (see paragraph 70, supra) and the added anticipatory reactor trip signals (see section II.D, supra) -- it is very likely that the PORV and/or safety valves will not be challenged for a loss of main feedwater and/or turbine trip transient. Before these modifications were made, it was expected that the PORV would be challenged for all such transients initiated during power operation. NRC Staff Testimony of Paul E. Norian on Adequacy of Safety and Relief Valves (CEC Contention 1-4), following Tr. 1163 ("Norian Valve Testimony"), at 4; Staff Ex. 2 at 3-7. As we have found, however (paragraph 115, supra), failure of the PORV will be safely mitigated by high pressure injection. In the event that the pressurizer safety valves were opened to relieve reactor coolant system pressure, and one or both of these valves stuck open, the resultant break would be bounded by existing LOCA analyses which show that the core will remain covered and adequately cooled. Karrasch-Jones Testimony at 54-56; Norian

Valve Testimony at 5. Consequently, the Board finds that the failure of safety and/or relief valves in the Rancho Seco primary system will not result in an unsafe condition.

118. The effect of non-condensible gases that could be released into the reactor coolant system during a LOCA has also been examined to assess the effect of gas accumulation on the heat removal process by the steam generators. Licensee's analysis shows that the quantity of non-condensible gas produced will neither prevent natural circulation nor impair the condensation heat transfer process in the steam generators. Karrasch-Jones Testimony at 47; Tr. 965, 966 (Jones). The NRC Staff has also estimated the effect of non-condensible gases and does not believe that enough non-condensible gases would enter the system to disrupt natural circulation flow or significantly degrade steam generator heat transfer. Staff Ex. 2 at 4-5 and 4-72; Tr. 1421, 1422 (Norian). To summarize, then, our findings on Board Question CEC 1-10, the Board finds that the physical configuration of the Rancho Seco primary system is not such as to permit unsafe accumulation of steam or other gases.

119. The results of the NRC Staff's review of the generic small-break LOCA analyses performed by B&W on behalf of operating plants with B&W systems, including Rancho Seco, are presented in Staff Exhibit 2. The Staff's conclusions are stated as follows:

> B&W has performed a sufficient spectrum of small break LOCA analyses to identify the anticipated system performance for breaks in this range. These analyses serve as an adequate basis for developing improved operator guidelines for handling small break LOCAs. In addition, these analyses provide an adequate basis for demonstrating that proper operator action coupled with a combination of heat removal from the primary system through the break, the steam generators and with the HPI system, assure adequate core cooling. The required operator actions are: (1) tripping of the RCPs shortly after initiation of a small break LOCA; (2) termination of HPI in the event of primary system repressurization, provided there is adequate subcooling; and (3) manual restoration of AFW flow to the steam generators in the event of a failure of the AFW system. With regard to tripping of the RCPs, B&W estimates at least three minutes are available for the operator to perform this action.

Staff Ex. 2 at 4-25. See also, Norian Trip Testimony at 4, 5; Lewis Testimony at 6.

120. The NRC Staff believes that improved understanding of small-break LOCAs is necessary and supports the current NRC research programs at the Semi-scale and LOFT facilities which during 1980 will explore the sensitivity of operation with and without reactor coolant pump use. Norian Trip Testimony at 6. In the meantime, however, since all analyses have confirmed that the plant can be maintained in a safe condition (as defined by 10 C.F.R. § 50.46) during a small-break LOCA without the reactor coolant pumps operating, provision for prompt tripping of the pumps upon indication of a LOCA (receipt of a low reactor coolant system pressure safety features actuation signal) assures that adequate core cooling

is provided. While other (non-LOCA) events may lead to a low pressure safety signal, tripping of the reactor coolant pumps for these events still provides adequate core cooling. Consequently, while further analyses and tests may be performed to understand more exactly the effect of continued reactor coolant pump operation during a small-break LOCA, current procedures assure the safe operation of Rancho Seco.[69] Norian Trip Testimony at 6; Karrasch-Jones Testimony at 54.

121. Natural circulation normally will be relied upon to cool the core in any transient that results in tripping of the reactor coolant pumps. Norian Circulation Testimony at 4, 5. CEC witness Lewis testified that given operator adherence to current instructions, natural circulation ought to be reliable. Lewis Testimony at 9. The Board agrees. In addition, Rancho Seco operating procedures provide specific direction to the operator in the event natural circulation cannot be confirmed. Rodriguez Testimony at 28. In the event of a continued loss of natural circulation, B&W plants with a lowered loop design (such as Rancho Seco) could still provide adequate core cooling in a "feed and bleed" mode which utilizes both high pressure injection trains to inject water into the reactor coolant system, while bleeding water out of the system through the pressurizer relief and safety valves. Norian Circulation Testimony at 7; Staff Ex. 2 at 4-26; Tr. 1111

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69 Paragraphs 116 and 120 constitute the Board's findings on Additional Board Question 2.

(Jones). CEC witness Lewis testified that ". . . with respect to feedwater transients and small break LOCAs I generally believe that there is sufficient redundancy in plant equipment, and options open to the operator, to provide adequate assurance of core cooling (provided the operator takes appropriate action)." Lewis Testimony at 7.

122. The B&W analyses of small-break LOCAs show that some operator action, both immediate and follow-up, is required. Immediate operator action is defined as those actions committed to memory by the operators which must be carried out as soon as the problem is diagnosed. Follow-up actions require operators to consult and follow the steps in written and approved procedures which must always be readily available in the control room for the operators' use. Staff Evaluation at 19. On the basis of its analyses, B&W developed operating guidelines, as required by the Commission's Order of May 7, 1979, to define operator actions during a small-break LOCA and to provide a description of plant behavior during a small-break LOCA and the effect of the defined operator actions. Id.; Karrasch-Jones Testimony at 56. Revisions recommended by the NRC Staff were incorporated in the guidelines. Staff Evaluation at 20.

123. Licensee then applied these guidelines to Rancho Seco, and developed and implemented procedure changes to provide for appropriate operator action. These procedures define the required operator action in a spectrum of break

sizes for a loss-of-coolant accident in conjunction with various equipment availability and failures. The NRC Staff reviewed the procedures to determine conformance with the B&W guidelines; comments generated in the course of the review were incorporated in further revisions; and the procedures were approved by the Staff prior to the resumption of operation at Rancho Seco on July 5, 1979. Staff Evaluation at 20-23; Rodriguez Testimony at 26; NRC Staff Testimony of Bruce A. Wilson on Operator Training and Competence, Etc., following Tr. 3788 ("Wilson Operator Testimony"), at 2; Tr. 3812, 3813 (Wilson).

124. In the immediate action requirements of Rancho Seco's procedure for loss of reactor coolant, strong emphasis is placed on maintaining reactor coolant system pressure-temperature relationships to assure that a subcooling condition of at least 50°F exists. Specifically, the procedure requires that upon automatic initiation of high pressure injection all reactor coolant pumps are tripped and high pressure injection shall not be terminated unless: (1) low pressure injection pumps are in operation and flowing at a rate of not less than one thousand gallons per minute each and the situation has been stable for twenty minutes; or (2) all hot and cold leg temperatures are at least 50°F below the saturation temperature for the existing reactor coolant system pressure and the hot leg temperatures are not more than 50°F greater than the secondary side saturation temperature. If 50°F subcooling cannot be

maintained, the procedure requires the high pressure injection system to be reactivated. The Rancho Seco procedure for plant shutdown and cooldown has also been modified to address specifically additional operator action to be taken in a small break accident with a loss of forced circulation. This procedure directs the operator to verify that auxiliary feedwater is supplying the steam generators and that generator levels are being maintained at 50 percent on the operating range. The procedure identifies reactor coolant system differential temperatures which confirm natural circulation, identifies differential temperatures at which the operator must take additional action to improve natural circulation flow, and provides specific direction to the operator in the event that natural circulation cannot be confirmed. Rodriguez Testimony at 27, 28.

125. The general subject of Rancho Seco operator training, including that provided immediately after the accident at Three Mile Island, is addressed below in section II.I. It is relevant to note here, however, that the post-TMI training (provided prior to the restart of Rancho Seco) gave a great deal of attention to recognition and understanding of the symptoms unique to small break conditions and the reasons for immediate operator actions to mitigate the consequences of small-break LOCAs. Each licensed operator completed TMI training on the B&W simulator. Licensee conducted special training sessions on the concepts and use of the small-break

LOCA procedure. Each licensed operator passed a written examination on TMI, administered by Licensee, which was audited by the NRC Staff for content and grading. Deficiencies revealed in an NRC audit of operators resulted in additional training by a Licensee consultant, an additional audit by the consultant, and a final audit by the NRC Staff during which no deficiencies were uncovered. Rodriguez Testimony at 15-18, 28-29; Wilson Operator Testimony at 4-7; Staff Evaluation at 23-25; Tr. 3821 (Wilson).

126. In answer to the specific inquiry made in Board Question CEC 1-7, no party has identified a specific operator training action responding to small-break LOCAs which fails to give sufficient attention to providing appropriate analytical bases for operator actions. See, e.g., Tr. 1853 (Webb). Our examination of the record of the depositions in this proceeding of three licensed Rancho Seco operators gave us no reason to disagree with the Staff's testimony that the Rancho Seco licensed operators adequately understand the analytical bases of the actions they may be required to take pursuant to Subparagraph (d) of the Commission's Order of May 7, 1979.[70] Wilson Operator Testimony at 7.

127. In conclusion, the Board finds that the B&W nuclear steam system has been proven by analysis, testing and

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70 They displayed a good understanding, for example, of the requirement to trip the reactor coolant pumps. See CEC Ex. 36 at 136; CEC Ex. 37 at 52, 53; CEC Ex. 38 at 10, 11. See also, Tr. 3277-3279 (Rodriguez).

operating experience to have adequate capability to establish and maintain natural circulation following loss of forced reactor coolant flow. Operator guidance and training for monitoring and controlling natural circulation have been provided. Consequently, in response to the inquiry in Board Question CEC 1-2, the Board finds that there is adequate understanding of natural circulation in the Rancho Seco system and that it is a reliable means of cooling the core. Void formation in the reactor coolant system does not occur during normal plant operation or anticipated transients such as a loss of feedwater. Void formation associated with feedwater transients can occur, however, where off-normal conditions are involved -- such as a loss-of-coolant accident. For those events, and in response to the Commission's Order of May 7, 1979, analyses have been performed which demonstrate adequate core cooling, and appropriate operator guidance has been developed and provided. Consequently, the Board finds that Rancho Seco can safely mitigate events, including small-break LOCAs, which may involve the accumulation of steam or other gases in the primary system.

G. 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?

128. The AFW system is an emergency system designed to supply feedwater to the steam generators in order to remove

heat from the reactor coolant system in the event of a loss of main feedwater. Dieterich Testimony at 6; NRC Staff Testimony of Philip R. Matthews on Reliability and Timeliness of the Emergency Feedwater System (Board Question CEC 1-6), following Tr. 1163 ("Matthews AFW Testimony"), at 2. The AFW system is not designed by the NSSS vendor, but by the architect-engineer that furnishes the "balance of plant" design. Dieterich Testimony at 7; Tr. 576 (Jones). Thus, there are wide differences in the AFW designs of the various plants having a B&W PWR, although all the AFW systems at those plants meet certain criteria specified by B&W.[71]

129. The AFW system must be able to operate over a time period sufficient either to hold the plant at hot standby for several hours or to cool down the reactor coolant system to the temperature and pressure conditions which permit the low pressure decay heat removal system to replace the AFW system for reactor coolant system heat removal. Matthews AFW Testimony at 2. The latter period of time is also on the order of a few hours.[72]

130. The primary water source for the AFW system at Rancho Seco is the condensate storage tank, which has a

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71 The design criteria imposed by B&W require that the AFW system supply feedwater to the steam generators at a rate of 760 gpm within 40 seconds of receiving an actuation signal. Dieterich Testimony at 7. As will be seen, the Rancho Seco AFW system meets this criterion.

72 For example, during the TMI-2 accident that point was reached after 9 1/2 hours of AFW system operation. Tr. 1629, 1630 (Matthews).

capacity of approximately 400,000 gallons -- sufficient to supply AFW for a period of at least 24 hours. Tr. 1491 (Matthews); Tr. 2057, 2324 (Dieterich). Two alternate water sources are the Folsum South Canal and an on-site reservoir. Matthews AFW Testimony at 3. Located in close proximity to the condensate storage tank are two pumps which take suction from the bottom of the condensate storage tank through separate pipes and feed independent but interconnected piping systems. Id.; Tr. 2323 (Dieterich). One of the pumps is driven by an electric motor; the other pump has two independent drives, an electric motor and a steam turbine. The electric motor drives are powered by the plant's AC system. The steam turbine drive runs on steam provided by the steam generator. Each pump is capable of delivering AFW flow against the maximum steam generator pressure to piping supplying both generators. Matthews AFW Testimony at 3; Tr. 1491 (Matthews).

131. Each pump, and the piping and other devices connected to it, constitutes an independent AFW subsystem or "train". Each train is capable of supplying AFW to either or both steam generators under automatic or manual initiation and control.[73] Matthews AFW Testimony at 2. For each AFW train, the pump discharges into two parallel lines. In one of the lines there is an air-operated AFW flow control valve whose

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73 There are cross-connections in both the suction and the discharge line of each pump, permitting either pump to feed one or both of the generators. Matthews AFW Testimony at 3; Tr. 1491 (Matthews).

operation is currently controlled by the ICS. In the other line there is a motor-operated AFW flow control valve which operates independently of the ICS. These two valves on each train control the flow of AFW to the steam generator. Tr. 1491, 1492 (Matthews); Matthews AFW Testimony at 3. Beyond the valves, each pair of parallel lines rejoins and goes on to the steam generator. Steam from the generators is in turn transmitted to the steam lines and from there it is available for driving the steam-turbine driven pump. Tr. 1492, 1493 (Matthews).

132. Each train of the AFW system normally derives its power from a separate electric bus fed from off-site power. Each of these buses is backed up by an on-site diesel generator. In the event of a loss of both off-site and diesel power, the AFW system is driven by the turbine-driven pump, which needs no electric power to operate. Tr. 1495-1497 (Matthews).

133. The AFW system at Rancho Seco is initiated automatically by either of two events: (a) loss of all reactor coolant pumps; or (b) low main feedwater pump pressure. Each of these events starts both pumps and opens the air-operated AFW flow control valves on both trains. Tr. 1512 (Matthews); Matthews AFW Testimony at 3, 4. In addition, the turbine-driven pump is automatically started, and the motor-driven AFW flow control valves on both trains are opened, by a safety features actuation signal (SFAS) generated by

either a low primary system pressure or a high reactor building pressure. Tr. 1512, 1513 (Matthews); Matthews AFW Testimony at 4; Dieterich Testimony at 7. These automatic initiation features reduce the dependence on operator action. Rodriguez Testimony at 40. However, the AFW system can be manually initiated and controlled by the operators at all times from the control room. Tr. 1512, 1534 (Matthews); Matthews AFW Testimony at 4.

134. The automatic initiation of AFW is accomplished by means of control-grade devices.[74] The automatic SFAS actuation is safety-grade. Tr. 1493, 1494 (Matthews). With the exception of the automatic control-grade initiation, the Rancho Seco AFW system meets all NRC requirements for safety-grade systems. Tr. 1314, 1493-1494 (Matthews).

135. The reliability of the AFW system at Rancho Seco and other B&W PWRs was the subject of Staff scrutiny following the TMI-2 accident. The reasons for this attention were a recognition that reliable AFW operation is an integral part of a plant's safe response to a loss of feedwater transient, and the perception that B&W PWRs respond faster to transients initiated on the secondary side, making timely and

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74 The automatic start of the AFW pumps is accomplished independently of the ICS. The ICS, however, controls the position of the air-operated flow control valves. Tr. 1386, 1387 (Novak, Matthews). As noted above (see paragraph 43, supra), procedures have been implemented to ensure AFW flow control independent of the ICS; ultimately the Rancho Seco AFW system will be upgraded to safety grade and will be totally independent of the ICS. Tr. 1243-1245 (Capra); Tr. 1286, 1296-1297 (Thatcher).

reliable AFW initiation and delivery important in those plants. Tr. 1485, 1486 (Matthews).[75] The Rancho Seco AFW system meets the Staff acceptance criteria contained in Section 10.4.9 of the NRC Standard Review Plan. Matthews AFW Testimony at 4-6.

136. As a result of that scrutiny, the Staff and Licensee agreed on a number of short-term action items to upgrade the timeliness and reliability of the Rancho Seco AFW system in light of the lessons learned from the TMI-2 accident. Tr. 2077, 2078 (Dieterich). The short-term actions were primarily intended to make sure that operators could recognize abnormal conditions and respond to them appropriately, so that the AFW system was initiated and running in a timely manner. Part of this effort consisted of giving the operators additional instrumentation in the control room to enable them to diagnose those abnormal conditions early. Tr. 1486 (Matthews).

137. Item (a) of the five short-term items proposed by Licensee in its April 27, 1979 letter (CEC Ex. 25), and confirmed by the Commission in its Order of May 7, 1979, dealt with the AFW system and called for "[u]pgrad[ing] of the timeliness and reliability of delivery from the Auxiliary Feedwater System by carrying out items 1 through 9 identified

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75 The Staff was also concerned about the fact that during the first eight minutes of the TMI-2 accident AFW was unavailable because it had been valved out of service. While this delay in delivery of AFW may not have led to core damage at TMI, it focused the Staff's attention on that system, which previously had been of less interest because it was not classified as a "safety" system. Tr. 1529 (Capra); Dieterich Testimony at 6.

in enclosure 1". CEC Ex. 25 at 1. The first of these nine items required Licensee to "[r]eview 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." Id., Enclosure 1 at 1. This requirement stems from the fact that the motor-driven AFW pump at Rancho Seco does not presently load automatically onto the electric buses powered from diesel generators upon the loss of all off-site power. Tr. 1523, 1524 (Matthews).[76] Since no automatic transfer of the motor-driven pump to the diesel-powered buses exists, this action has to be taken by the operator from the control room. Tr. 1533 (Matthews).

138. The loading operation itself is a simple, brief procedure involving the insertion of a key into a lock and starting the pump. Tr. 1526 (Matthews); Tr. 1528 (Capra). However, a certain degree of operator judgment is required, since he must examine the control room panels and satisfy himself that the plant conditions permit the additional

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76 One of the long-term plant modifications that Licensee has committed to make to improve further the reliability of the Rancho Seco AFW system is to implement an automatic loading of the motor-driven AFW pump onto the diesel generator bus upon a loss of all off-site power. Matthews AFW Testimony at 17, 18; Tr. 1524 (Matthews). Licensee has submitted a proposed design for this modification to the Staff. This design is being analyzed by the Staff to examine what impact loading this pump onto the diesel generator will have on the voltage and frequency of the bus. Tr. 1156, 1524 (Matthews); Tr. 1528 (Capra); Tr. 2086, 2087 (Dieterich).

electric load of the pump to be added to the diesel. Tr. 1432, 1433 (Novak).[77] The testimony shows that prior to this required action the operators at Rancho Seco had an understanding of the manual loading procedure. Tr. 1525, 1526 (Matthews); Tr. 2044 (Dieterich); Tr. 3247 (Rodriguez). Still, this requirement established a formal written procedure providing specific direction for the operator on the steps required to load the motor driven pump onto the diesel generator buses. Staff Evaluation at 3. The Staff restart team that visited Rancho Seco reviewed the new procedure, gave written tests to all operators, and conducted a "walk through" of the procedure with a random sample of senior reactor operators and reactor operators. Tr. 1531, 1532 (Capra, Novak).[78] As a result of this evaluation, the Staff concluded that the procedure was adequate, the operators were properly trained to apply it, and that this part of the May 7 Order had been satisfied. Staff Evaluation at 3; Matthews AFW Testimony at 11; Tr. 1620, 1621 (Matthews).

139. The second action required of Licensee was to implement procedures and conduct training "to provide an operator at the necessary valves in phone communication with

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77 Tests have been conducted at Rancho Seco which show that the pump can be loaded onto the diesel generator buses without causing an electrical disturbance or load shedding. Tr. 2302, 2303 (Dieterich).

78 This auditing process was also conducted with respect to the other procedural changes instituted as a result of the Commission Order of May 7, 1979. Tr. 1531 (Capra).

the control room during the surveillance mode to carry out the valve alignment changes upon AFW demand events." CEC Ex. 25, Enclosure 1 at 1. The reason for this requirement is that during the quarterly surveillance and in-service testing of the AFW pumps, the cross-tie valves are open and both AFW trains are out of service. Tr. 1504 (Matthews); CEC Ex. 20 at 9. Prior to implementation of the change, if AFW was needed during surveillance testing an operator had to be dispatched to close the full flow recirculation valve (FWS-055), or otherwise feedwater flow would recirculate between the condensate storage tank and the pumps instead of being delivered to the steam generators. Tr. 1504, 1505 (Matthews); Tr. 3247, 3248 (Rodriguez). Under the new surveillance procedures instituted by Licensee to comply with this requirement, an operator is now stationed at this valve, FWS-055, during surveillance testing of the pumps.[79] He is in continuous telephone communication with the control room so that the valve can be closed immediately if AFW flow is needed. In addition, he is to ensure that the valve is closed after completion of the test. Tr. 1505 (Matthews); Tr. 2045, 2046 (Dieterich); Tr. 3247, 3248 (Rodriguez). Independent verification is also provided of correct valve alignment after completion of testing. This is accomplished by sending another operator to check that all

79 Continuous presence of an operator at FWS-055 was not required prior to this change. Tr. 2045 (Dieterich); Tr. 3247, 3248 (Rodriguez).

valves have been left in the proper position after completion of testing. Tr. 1700 (Matthews); Matthews AFW testimony at 11. The Staff reviewed the new surveillance procedures, verified that the operators are familiar with them, and concluded that the procedures and their implementation complied with the Commission Order of May 7, 1979. Staff Evaluation at 4, 5; Matthews AFW Testimony at 11.

140. The third requirement directed Licensee to institute procedures and train its operators "to provide for control of steam generator level by use of safety grade AFW bypass valves in the event that ICS steam generator level control fails." CEC Ex. 25, Enclosure 1 at 1. This requirement, which to some extent overlaps with item (b) of the May 7 Order, was intended to verify that operators would control the steam generator level manually in the event of an ICS failure. Tr. 1537, 1538 (Novak). Again, the operators were capable of taking this action before the procedures were instituted. Tr. 1539, 1540 (Capra); Tr. 2047 (Dieterich); Tr. 3248, 3249 (Rodriguez). However, there were no written procedures specifically describing the steps that the operator would have to take to maintain control of steam generator level. Tr. 1539, 1540 (Capra); Tr. 2047, 2048 (Dieterich); Tr. 3248 (Rodriguez). In response to this requirement, Licensee developed an emergency procedure for loss of feedwater simultaneous with an ICS failure. The procedure instructs the operator on restoring feedwater, taking manual control of the

AFW bypass valves from the control room and using them to maintain specified steam generator levels.[80] Staff Evaluation at 5, 6. The Staff reviewed the procedures and audited the operators' training with respect to them, and concluded that both the procedures and the training were adequate to comply with the Commission Order of May 7, 1979. Id. at 6; Matthews AFW Testimony at 11.

141. The fourth required action was to verify "that the Technical Specification requirements of AFW capacity are in accordance with the accident analysis" and to verify "[p]ump capacity with mini flow in service." CEC Ex. 25, Enclosure 1 at 1. This requirement was intended to verify that the AFW pumps would supply the flow for which credit was taken in the accident analysis in the Rancho Seco Final Safety Analysis Report. Tr. 1540 (Matthews); Tr. 2048 (Dieterich). To comply with this requirement, Licensee had B&W perform an analysis and verify that a total flow rate of approximately 760 gpm to either or both steam generators would be sufficient to accommodate a decay heat level of 4.5 percent of rated power[81] plus

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80 Essentially, the operator must adjust the valve position periodically so that the AFW makeup flow is equal to the amount being boiled off by the steam generator. Since changes in steam generator level after a reactor trip and AFW initiation are relatively slow, all that is required to maintain constant level is to make small adjustments in valve position while checking the steam generator level indication. Tr. 2073-2075, 2307-2308 (Dieterich).

81 Analysis shows that 35 to 40 seconds after a reactor trip, the total decay heat from the core (residual heat and delayed fission heat) is down to 4.5 percent of rated power. Since it (footnote continued next page)

the heat input from the reactor coolant pumps. Staff Evaluation at 7. Then, tests were conducted which showed that each of the AFW pumps has the capability of delivering a minimum of 780 gpm to the steam generators, in addition to 60 gpm of recirculating water ("mini flow") to preclude pump overheating. Tr. 1540-1541, 1747-1748 (Matthews); Tr. 2050, 2067-2071 (Dieterich). The Staff reviewed the AFW flow rate test results and concluded that sufficient AFW capacity had been demonstrated for compliance with the May 7 Order. Staff Evaluation at 7, 8; Matthews AFW Testimony at 13.[82]

142. The fifth short-term requirement placed on Licensee was to make modifications "to provide verification in the control room of AFW flow to each steam generator." CEC Ex. 25, Enclosure 1 at 1. Prior to this modification, the operators could determine indirectly whether AFW flow was established by monitoring steam generator level; however, there was no direct AFW flow indication. Tr. 2053, 2054 (Dieterich); Tr.

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takes at least 35 to 40 seconds for the AFW flow to initiate, the calculated flow is adequate to meet the steam generator demands. Tr. 1750-1752 (Matthews); Tr. 2329 (Dieterich).

82 While the Staff believes that the flow capacity of the Rancho Seco AFW system is adequate, it is seeking on a long-term basis more detailed information from all PWR licensees, including SMUD, as to what transients or accidents were considered in establishing the AFW flow requirements of each plant. Tr. 1541-1546 (Matthews); CEC Ex. 21, Enclosure 1 at 10-11. At the hearing, Licensee witness Dieterich testified that the transient considered in sizing the AFW flow was a reactor trip simultaneous with loss of main feedwater, since only under those circumstances will the AFW be called upon to remove core decay heat. Tr. 2051, 2052 (Dieterich).

3249 (Rodriguez). To comply with this portion of the Order, Licensee installed flow meters on each of the AFW lines -- as close as possible to, but outside of, the steam generators -- to give a reliable indication in the control room of the existence and amount of AFW flow to the steam generators. Tr. 1546-1548 (Matthews, Novak); Tr. 2053, 2054 (Dieterich). Calibration tests were conducted on the flow meters after installation. The test results showed that the meters indicated flow rate within the acceptance criteria for accuracy. Based on these tests, the Staff concluded that Licensee had complied with this part of the May 7 Order. Staff Evaluation at 8; Matthews AFW Testimony at 12.

143. The AFW flow meters are not intended to replace the steam generator level meters, but to supplement them. Tr. 1549 (Matthews); Tr. 2054 (Dieterich).[83] Licensee has committed to upgrade the AFW flow indication to safety grade. Tr. 2053, 2054 (Dieterich).

144. The sixth action item was to "[r]eview and revise, as necessary, the procedures and training for providing alternate sources of water to the suction of the AFW pumps." CEC Ex. 25, Enclosure 1 at 1. In response to this requirement, Licensee modified its emergency procedures to provide specific

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83 AFW flow meters are particularly useful in B&W PWRs because the AFW is showered from above on the steam generator tube bundle; so it is possible to have established a heat sink without the generator level yet returned within the range of the instrument. Tr. 1548, 1549 (Matthews); Tr. 2322 (Dieterich).

guidance to the operator on how to obtain an alternate source of water for the AFW system. Tr. 2056 (Dieterich); Tr. 3250 (Rodriguez); Staff Evaluation at 9. In addition, since transfer from the condensate storage tank to the plant reservoir[84] or the Folsum South Canal[85] will not be necessary for at least 24 hours after transient initiation, this procedure is one for which there would be sufficient time for operator familiarization. Tr. 2056, 2057 (Dieterich); Tr. 3250 (Rodriguez); Staff Evaluation at 9. The Staff reviewed the modified procedure, verified the operators' training on it, and concluded that Licensee had met the requirements of this part of the May 7 Order. Staff Evaluation at 10; Matthews AFW Testimony at 11, 12.

145. The seventh short-term action required of Licensee was to conduct a design review and implement modifications, as necessary, "to provide control room annunciation for all auto start conditions of the AFW system." CEC Ex. 25, Enclosure 1 at 2. Prior to this modification, there would be annunciation in the control room of SFAS actuation of the AFW

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84 The supply of water to the AFW pumps from the reservoir does not require use of transfer pumps because the water moves by the force of gravity. Tr. 1552 (Matthews); Tr. 2108 (Dieterich).

85 The Staff identified a possible omission in the procedure for obtaining water from the Folsum South Canal in that the procedure does not call for starting a transfer pump to draw water from the canal. CEC Ex. 21, Enclosure 1 at 6; Tr. 1551 (Matthews). Licensee has committed to review the procedures and ensure that they describe adequately how to obtain water from this source. CEC Ex. 22, Attachment at 2.

system, but not of automatic AFW initiation upon loss of all reactor coolant pumps or upon main feedwater pump low pressure, nor of manual AFW initiation. Tr. 1553 (Capra); Tr. 3250 (Rodriguez). While the operator would have inferred the automatic start of AFW from indication of loss of the reactor coolant pumps or loss of main feedwater, the addition of annunciators of AFW initiation would serve to draw the operator's attention to that event. Tr. 1553-1555 (Capra, Novak); Tr. 2057 (Dieterich); Tr. 3250, 3251 (Rodriguez). In response to this requirement, Licensee provided indication of all automatic and manual AFW initiations on an annunciation panel inside the control room. Staff Evaluation at 10. The Staff reviewed this design modification and concluded that it complies with this part of the May 7 Order. Id.; Matthews AFW Testimony at 12.

146. The eighth required action was to institute procedures and conduct training "to provide guidance for timely operator verification of any automatic initiation of AFW." CEC Ex. 25, Enclosure 1 at 2. This modification is related to the previous one; once the operator is alerted by the annunciators (which are both visual and audible) that AFW has been initiated, he is required to verify that the pumps are running and that AFW flow to the steam generators has been established. Tr. 2058, 2059 (Dieterich); Tr. 3252 (Rodriguez). While the operators would have been expected to verify AFW flow once they learned of AFW initiation, this modification further ensures

that the operators will take that action. Tr. 2058, 2059 (Dieterich). In response to this requirement, Licensee and the Staff confirmed that existing emergency procedures specifically direct the operators to verify AFW flow to the steam generators upon initiations of AFW. The Staff also verified that the Rancho Seco operators are adequately trained in these procedures, and thus concluded that Licensee had complied with this part of the May 7 Order. Staff Evaluation at 11; Matthews AFW Testimony at 12, 13.

147. The last short-term action of item (a) of the May 7 Order required Licensee to verify that "the air operated level control valves (a) Fail to the 50% open position upon loss of electrical power to the electrical to pressure converter, and (b) Fail to the 100% open position upon loss of service air." CEC Ex. 25, Enclosure 1 at 2. This was merely a verification procedure intended to confirm that the air-operated valve at Rancho Seco fails in an open position so that a flow path is always open to the steam generator without needing to dispatch an operator to open the valve manually.[86] Tr. 1557-1559 (Matthews); Tr. 2060 (Dieterich). Verification tests for the failure mode of the valve were conducted, and the test results confirmed that the valve fails to the indicated

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86 If necessary, AFW flow can be reduced after this valve has failed in a fully or partially open position in at least two ways: by turning off one or both pumps, and by dispatching an operator to reposition the valve manually. Tr. 2060 (Dieterich).

position. Thus the Staff determined that Licensee was also in compliance with this requirement of the May 7 Order. Staff Evaluation at 12; Matthews AFW Testimony at 13, 14.

148. Clearly, the above nine short-term items were not of equal importance,[87] but all contribute to ensuring a faster and more reliable delivery of AFW flow to the steam generators, which is the sole purpose of the AFW system. Tr. 1559 (Matthews). It is difficult to determine the significance of the enhancement of the timeliness and reliability of the Rancho Seco AFW system provided by these modifications, because the system at Rancho Seco has a perfect operating history. Tr. 3255 (Rodriguez). It has been called upon to operate a total of 101 times, both under actual transient and test conditions, and in every instance it has provided feedwater. Rodriguez Testimony at 49. In a general sense, however, it is clear that these modifications can only enhance the timeliness and reliability of the AFW system.

149. Some time after the May 7, 1979 Order, the Staff decided to conduct a generic study of the reliability of the AFW systems at all PWRs. Tr. 1485-1486, 1569 (Matthews). It first conducted such a study for the AFW systems of the

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87 Licensee witness Rodriguez testified that, in his opinion, the most important of these modifications were the provision of AFW flow indication (No. 5), stationing of an operator at the FWS-055 valve location during surveillance testing of the AFW pumps (No. 2), installation of visual and audible annunciation in the control room of AFW initiation (No. 7), and verification of procedures that require the operator to confirm AFW flow to the steam generator (No. 8). Tr. 3257 (Rodriguez).

Westinghouse ("W") and Combustion Engineering ("CE") PWRs during May and June, 1979. Tr. 1578, 1579 (Matthews). The Staff then requested the owners of B&W PWRs to perform reliability studies of the AFW systems at their plants, utilizing the same methodology, assumptions and data base that went into the W and CE studies so that the study results would be comparable. Tr. 1573, 1574 (Matthews); Tr. 2082 (Dieterich). The B&W PWR owners selected B&W to perform the reliability studies for them. The study for the Rancho Seco AFW system was formally submitted to the Staff in December, 1979, and is in the record of this proceeding as CEC Exhibit 20.

150. The analysis in CEC Exhibit 20 was conducted utilizing a "fault-tree" technique[88] and three scenarios: loss of main feedwater ("LMFW"), loss of main feedwater accompanied by loss of all off-site power ("LOOP"), and loss of main feedwater accompanied by loss of all AC power ("LOAC"). Tr. 1582 (Matthews). These three cases were chosen because, when

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88 The "fault-tree" technique is a systematic method for establishing the dominant contributors to the failure of a system. Tr. 1561 (Matthews). It takes an initial scenario and looks at all combinations of human errors and equipment failures that may ensue, assigning probability estimates to each and seeking to determine which of these combinations defeat the operation of the system, and their relative likelihood. Tr. 1561-1564 (Matthews). The main value of the fault-tree technique is that it permits a determination of the dominant sequences contributing to the failure of a system. Tr. 1564 (Matthews). Another value of the technique is that it permits a relative comparison to be made between the reliability of two systems (*e.g.*, between the AFW systems at two plants). Tr. 1562, 1563 (Matthews).

taken together, they encompass the vast majority of the instances in which AFW system operation is required. Tr. 1582, 1583 (Matthews). The postulated initial conditions were a loss of main feedwater followed by reactor trip, accompanied (in the second and third cases) by some power failure. From there, AFW initiation was assumed, and fault-trees were developed to determine which human errors or equipment failures, both within and outside the AFW system, can preclude delivery of AFW to the steam generators. Tr. 1565-1566, 1583-1584 (Matthews). The data base for equipment and human error failure rates was supplied by the Staff and was the same data base utilized in the W and CE AFW system reliability studies. This data base was derived from the Reactor Safety Study, as updated, and by information compiled from general industry sources as updated by Licensee Event Reports. Tr. 1573-1577 (Matthews); Tr. 2082 (Dieterich).

151. In the analysis, the reliability of the Rancho Seco AFW system was calculated in terms of the probability that an operator will be able to take corrective action to restore AFW flow within a given period of time after the initiating event, assuming the AFW system has failed to operate. Tr. 1591 (Matthews). The time intervals chosen were 5, 15 and 30 minutes. They were selected because NRC-supplied operator reliability data for these times were available. CEC Ex. 20 at 2; Tr. 1728, 1729 (Matthews). Mission success was defined in the study as attainment of flow from at least one pump to at

least one steam generator. CEC Ex. 20 at 2. Achievement of mission success within 5 minutes is roughly equivalent to the mission success criterion utilized by the Staff in its study of W and CE plants' AFW system reliability, i.e., avoidance of steam generator dryout, because the steam generator dryout time for a B&W PWR is approximately 5 minutes,[89] assuming actuation of an anticipatory reactor trip on loss of feedwater.[90] Tr. 1593, 1594 (Thatcher, Matthews); Tr. 1754 (Matthews); CEC Ex. 20 at 2. Achievement of mission success within 15 or 30 minutes would be important to the overall safety of the plant because adequate core cooling can be maintained for periods in excess of 20 minutes without AFW flow, provided at least one HPI pump is operating. CEC Ex. 20 at 2; CEC Ex. 21, Enclosure 1 at 1; Tr. 492-494, 519-522 (Lewis); Tr. 1484 (Novak); Tr. 1586-1587 (Matthews).

152. The Staff and Licensee agree that while steam generator dryout is an undesirable event because it results in challenging the plant's safety systems, it is not an event of safety concern.[91] Tr. 1595 (Matthews); Tr. 2010-2011,

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89 As a refinement on previous analyses, the steam generator dryout time assuming an anticipatory reactor trip has been reduced from 5 to 4 minutes. Tr. 2089-2090, 2112 (Dieterich). This reduction should have little impact on the results of the analysis. Tr. 2090-2092, 2107 (Dieterich); Tr. 1659 (Matthews).

90 If no anticipatory reactor trip takes place, the steam generator dryout time is approximately 1.5 minutes. Tr. 1594, 1753-1754 (Matthews).

91 If the steam generator boils dry, the primary system loses its heat sink, primary pressure and temperature increase, and (footnote continued next page)

2088-2089 (Dieterich). However, the Staff insists that, for the purpose of assessing AFW system reliability alone and without regard to the behavior of other systems available to protect the reactor, the mission success criterion should be delivery of AFW to the steam generator before it boils dry. Tr. 1597 (Matthews); CEC Ex. 21, Enclosure 1 at 1. Licensee disagrees and argues that the ultimate measure of AFW system reliability is the ability to remove decay heat from the core to prevent core damage. Tr. 2088-2089, 2093, 2107 (Dieterich).[92]

153. It is unnecessary to resolve this controversy in order to interpret the results of the analysis contained in CEC Exhibit 20. Figure 6 of that document shows a comparison between the Rancho Seco AFW system reliability and a range of reliability values for W PWRs in the three cases studied for the 5, 15 and 30 minute operator reaction time periods. For the 5 minute case, the Rancho Seco AFW system is shown to be approximately midway in the range of reliability for W plants in the LMFW and LOAC cases and in the low to medium portion of the range in the LOOP case.[93] Tr. 1618 (Matthews); CEC Ex. 20

(continued)
in a short time the PORV and the code safeties open to relieve primary pressure. Tr. 1610 (Matthews).

92 Licensee, however, is willing to accept a mission success criterion based on ability to provide AFW flow to the steam generator within a given time frame (say five minutes), but appears to be unwilling to agree to a definition that ties mission success to an event not taking place (steam generator dryout). Tr. 2321 (Dieterich).

93 The main reason the Rancho Seco AFW system reliability is
(footnote continued next page)

at iii, 12. These results would not be affected by adopting an avoidance of steam generator dryout criterion for mission success, because the range of reliability values for W plants is based on such a criterion and the plants that lie at the lower end of the range have some type of vulnerability that potentially inhibits the ability of the AFW system to provide feedwater to the steam generator. Tr. 1607-1610, 1657-1665 (Matthews).

154. The main significance of the 15 and 30 minute results is that they show an increased likelihood of successful operator action to restore AFW at Rancho Seco with increased time. There is a significant improvement in going from 5 to 15 minutes and a smaller improvement thereafter. Tr. 1591, 1592 (Matthews). These results show the converse of the generally accepted principle that the shorter the time allowed for an operator to do something, the lower the likelihood of his taking the correct action and the higher the probability of human error. Tr. 1666, 1667 (Matthews).

155. To summarize, the study of AFW system reliability conducted by B&W for the Rancho Seco plant indicates that the reliability of the Rancho Seco AFW system is

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(continued)
lower for the LOOP case is that the motor-driven pump does not load automatically onto the diesel generator powered buses in the event of a loss of off-site power. CEC Ex. 20 at 12, 13; Tr. 1619 (Matthews). When the plant modification is implemented to load automatically the motor-driven pump onto the diesel buses, the reliability of the system will be improved. Tr. 1619 (Matthews).

quite comparable to, and is bracketed by, that for W plants. Tr. 1606 (Matthews). Long-term modifications committed to by Licensee will improve the relative reliability of Rancho Seco's AFW system even further. Tr. 1619 (Matthews).

156. Some of these long-term modifications have already been mentioned. One will be a safety-grade initiation and control of the AFW system (which will entail installing a new control system entirely separate from the ICS, with safety-grade instrumentation and power supplies). Tr. 2099 (Dieterich).[94] Another modification is the automatic loading of the motor-driven AFW pump onto the diesel generator buses upon loss of all off-site power. This proposed action is awaiting Staff design approval. CEC Ex. 21, Enclosure 1 at 4-5; Matthews AFW Testimony at 17, 18; Tr. 1156 (Matthews). Another pending modification is the upgrade of the AFW flow indication to safety grade. Tr. 2116 (Dieterich); Matthews AFW Testimony at 18, 19. A modification of the AFW system piping to provide a remotely operated valve operable from the control room instead of the local, manually operated FWS-055 is also planned. This change will permit keeping an AFW train operable while the other one is under test, and the restoration of the AFW train under test back to operable status from the control room without the need to station an operator next to the valve.

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94 The proposed design for this modification has not yet been approved by the Staff; it is estimated that the modification will be implemented during the first half of 1981. Tr. 2098, 2099 (Dieterich).

Matthews AFW Testimony at 18; CEC Ex. 21, Enclosure 1 at 5; Tr. 2116, 2117 (Dieterich). Licensee also expects to implement an upgrade of the existing condensate storage tank level indication and low level alarm to safety-grade standards. Matthews AFW Testimony at 18; CEC Ex. 21, Enclosure 1 at 7; Tr. 2117 (Dieterich). The Staff has concluded, and the Board agrees, that these long-term modifications will improve further the reliability of the Rancho Seco AFW system. Matthews AFW Testimony at 19. In particular, upgrading the AFW system to safety grade will make it "extremely reliable" as called for by the B&W Reactor Transient Response Task Force. Staff Ex. 4 at 5-10; Tr. 2095 (Dieterich).

157. The foregoing review of the evidence indicates that, as CEC witness Lewis testified, the Rancho Seco AFW system in its present configuration is among the more reliable of such systems, and the likelihood of its failure is quite low. Lewis Testimony at 3, 4. The Staff is satisfied with the reliability of the system; the operating history shows that the system has provided feedwater in a timely manner[95] on every occasion in which it has been called upon to function. Tr. 1522 (Capra). The short-term modifications performed in accordance with the Commission's Order of May 7, 1979, have improved the reliability of the system "sufficiently to assure

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95 In fact, in every instance in which auxiliary feedwater has been called upon, AFW flow has been provided early enough to avoid steam generator dryout. Tr. 2119 (Dieterich).

safe plant shut down following loss of main feedwater." Matthews AFW Testimony at 19. Long-term modifications now being undertaken will further enhance the reliability and timeliness of the AFW system. Id. Because of these considerations, and in response to the inquiry in Board Question CEC 1-6, the Board finds that the timeliness and reliability of the AFW system at Rancho Seco were adequate and have been enhanced by the modifications directed by the Commission.

H. Safety System Challenges

Issue CEC 1-1: 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?

Issue CEC 1-12: 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?

158. The modifications and actions taken as a result of the Commission's Order of May 7, 1979 -- the addition of anticipatory reactor trips on loss of feedwater and turbine trip -- combined with the changes to the PORV and high pressure trip setpoints required by IE Bulletin 79-05B (see paragraph 70, supra), are expected to increase the number of reactor

trips at Rancho Seco. Karrasch-Jones Testimony at 39; NRC Staff Testimony of Mark P. Rubin and Thomas M. Novak Regarding the Design Basis for Rancho Seco Safety Systems (CEC Contentions 1-1 and 1-12), following Tr. 1163 ("Rubin-Novak Safety Systems Testimony"), at 3. The addition of the anticipatory reactor trip on loss of feedwater is not expected to increase the number of reactor trips, since such an event normally has resulted in a reactor trip on high reactor coolant system pressure. The addition of the anticipatory trip on turbine trip, however, is expected to increase the number of reactor trips, since the system previously was capable of avoiding a reactor trip during such an event. Karrasch-Jones Testimony at 10, 40.

159. Based upon Licensee's tabulation of data compiled from NRC publications, it appears that prior to the accident at Three Mile Island the yearly average of reactor trips at B&W plants was below the averages for Combustion Engineering and Westinghouse PWRs. Id. at 40. Licensee now expects the trip frequency for B&W units to increase and approximate the industry average. Id. at 41.

160. This expected increase in the number of reactor trips will not result, however, in the design and licensing basis of safety systems being exceeded. Id. at 39; Rubin-Novak Safety Systems Testimony at 3. During the course of designing Rancho Seco, certain criteria were established for the allowable number of plant transients which would result in thermal

cycles and stress on the reactor coolant pressure boundary. These criteria are detailed in design information supplied to the Rancho Seco operating staff in reports which describe the number of transients of each category allowed for in the basic plant design and include such plant responses as reactor trip and high pressure safety injection. Rubin-Novak Safety Systems Testimony at 4; Tr. 1449, 1450 (Novak).

161. To assure that challenges to safety systems do not exceed their design and licensing basis, Licensee has established administrative procedures to monitor these design basis transients. Rubin-Novak Safety Systems Testimony at 4; Rodriguez Testimony at 50. Among those transients for which specific data are recorded and monitored are reactor trips caused by loss of feedwater, loss of feedwater to one steam generator resulting in a dry OTSG, cooldowns from hot conditions to 140 degrees, and high pressure injection into the reactor coolant system. If the number of design cycles is approached, then corrective action can be taken.[96] Rodriguez Testimony at 50, 51; Rubin-Novak Safety Systems Testimony at 4. In addition, it should be noted that these safety systems are subjected to periodic testing and maintenance to assure that they are capable of performing their functions if required. Rubin-Novak Safety Systems Testimony at 4; Tr. 995 (Karrasch); Tr. 1448, 1449 (Rubin); Tr. 1451 (Novak).

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96 See, e.g., Tr. 1746 (Novak); Tr. 3358-3359, 3409-3410 (Rodriguez) (new operational procedures to avoid adding thermal cycles to HPI nozzles).

162. There is no evidence in the record to indicate that the modifications directed by the Commission in its Order of May 7, 1979, or any other current operating practices at Rancho Seco, will result in increased challenges to safety systems beyond the design and licensing basis of the facility. The Board finds in the negative, then, in answer to the questions posed in CEC Issues 1-1 and 1-12.

I. Operator and Management Competence

Issue CEC 3-1: Whether personnel adequately understand the mechanics of the facility, basic reactor physics, and other fundamental aspects of its operation?

Issue 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?

Issue 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?

FOE Contention III(d): The NRC orders in issue do not reasonably assure adequate safety because no procedures have been taken to assure facility management competence.

FOE Contention
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.

163. CEC Issues 3-1, 3-2 and 3-3, FOE Contentions III(d) and III(e), and Board Questions H-C 32 and H-C 34 raise various issues concerning the adequacy of the competence of Licensee's operators and management to provide reasonable assurance that the Rancho Seco plant will respond safely to feedwater transients.[97] One of the short-term actions and one of the long-term modifications required by the Commission's Order of May 7, 1979, directed Licensee to undertake additional training of its plant operators in light of the experience gained from the accident at Three Mile Island. See paragraphs 5 and 7, supra. The Board will examine the adequacy of those measures in the context of the training which has been, and is being, provided to the Rancho Seco management and operators.

164. By regulation of this Commission, no person may perform the function of an operator or senior operator at a nuclear reactor except as authorized by a license issued by the Commission. 10 C.F.R. § 55.3. An "operator" is defined as any

97 Board Question CEC 1-7, which addresses the adequacy of training actions for responses to small-break, loss-of-coolant accidents, is related to these issues, but was addressed above in section II.F along with the Board's findings on the adequacy of the small-break LOCA analyses which underlie operator actions. See, particularly, paragraphs 122-126, supra.

individual who manipulates a control of a facility, including the direction of another to manipulate a control. 10 C.F.R. § 55.4(d). "Controls," in turn, are defined as apparatus and mechanisms the manipulation of which directly affect the reactivity or power level of the reactor. 10 C.F.R. § 55.4(f). A "senior operator" is any individual designated by a facility license under 10 C.F.R. Part 50 to direct the licensed activities of licensed operators. 10 C.F.R. § 55.4(e). See also, 10 C.F.R. § 50.54(l). There are 24 licensed personnel (18 senior operators and 6 operators) on the operating staff at Rancho Seco. Tr. 3047, 3048 (Rodriguez), amending Rodriguez Testimony at 22. See also, Tr. 3400, 3401 (Rodriguez). Because, by Commission regulation, the licensed operators are the key personnel involved in the facility's response to feedwater transients and any associated off-normal conditions, the Board will examine their training and performance before turning to Licensee's management and unlicensed operators.[98]

165. Eight of the senior licensed operators at Rancho Seco do not normally stand control room watch, but serve in various supervisory and facility management positions. Each

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98 During a loss of feedwater, as well as any other abnormal operating condition, the NRC licensed operators assigned to the shift operating crew at the time, and not SMUD management, are responsible for responding to the event in accordance with established procedures and for taking the necessary action to control the reactor and associated plant systems. NRC Staff Testimony of Allen D. Johnson Relative to the Competency of SMUD to Operate the Rancho Seco Facility (FOE Contention III(d) and Board Question 32), following Tr. 3920 ("Johnson Testimony"), at 5; Tr. 3981-3984 (Johnson).

crew assigned to an eight-hour control room watch includes three licensed personnel. The control room operator holds an operator's license, while the shift supervisor and the senior control room operator each hold senior operator licenses. The NRC currently requires that two licensed personnel be in the control room at all times during plant operation.[99] Rodriguez Testimony at 22. There are four Rancho Seco training programs which are relevant to the competence of these licensed operators: "cold license" training, "hot license" training, the requalification program, and special post-TMI training.

166. The "cold license" training program was provided from 1970 to 1974 to the personnel initially licensed to operate Rancho Seco when it received a facility operating license in 1974. Rodriguez Testimony at 7. More than one-half of the presently licensed operating personnel received all or most of the cold license training. Wilson Operator Testimony at 3. The program included: 13 weeks of observation at an operating nuclear power plant; a 520-hour course in basic reactor physics and engineering; a 6-week PWR technology course

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99 The NRC is considering an expansion of the shift crew composition to require a second senior reactor operator on each shift, with a requirement that one of the senior operators be in the control room at all times. Tr. 3939-3940, 3949-3950 (Allenspach); Testimony of Frederick R. Allenspach Relating to Management and Technical Competence (FOE III(d) and Board Question 32), following Tr. 3920 ("Allenspach Testimony"), at 7. As we have found, however, the practice at Rancho Seco already is to have two senior operators on the crew; and as long as two licensed personnel must be in the control room, one of them will be a senior operator. See Tr. 4096 (Allenspach).

and a 10-week simulator course presented by B&W at its headquarters; a final review training course (including a simulator refresher course); and participation in plant start-up activities. Id.; Rodriguez Testimony at Appendix I.

167. The "hot license" training program is used to prepare operator candidates for licensing since the facility operating license was issued in 1974. Because candidates eligible for this training program normally have been employed in the Operating Division at Rancho Seco for two or more years, and thereby gain practical training in plant operations, the hot license training program does not include the observation course at another plant which was necessary for the cold license program. Rodriguez Testimony at 7; Wilson Operator Testimony at 4. Individuals eligible for this training program are selected for participation on the basis of a math and science written examination, an interview and an evaluation of previous work performance.[100] Rodriguez Testimony at 7.

168. The first part of the hot license training program consists of 600 hours of academic training and includes a mathematics couse, a physics course and a related technologies course. The next phase of the program involves

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100 The District has attempted to hire into the Nuclear Operations Department, for unlicensed operator positions and licensing candidates, individuals with a two-year college degree in the electrical-mechanical area or equivalent experience. Tr. 3393, 3484 (Rodriguez). The NRC Staff is in the process of implementing revised criteria for license examination eligibility, including minimum experience. CEC Ex. 49, Enclosure 1 at 1, 2. See also, Tr. 3075 (Rodriguez).

in-plant operations training at Rancho Seco and includes systems and operations training in the control room, the application of procedures to systems under operating conditions, and fuel handling training. The third part of the prospective operator's preparation is simulator training. This includes a one-week review course at Rancho Seco and a three-week course at the B&W simulator in Lynchburg, Virginia.[101] Finally, the candidate undergoes a pre-license review course, including a comprehensive oral and written examination administered by the District. The NRC's license examination is then given to the candidate only if Licensee certifies that the candidate is prepared. Rodriguez Testimony at 7-10 and Appendix II. Requirements for approval of the operator license application are set forth in the Commission's regulations at 10 C.F.R. § 55.11. The scope and content of the NRC's written examinations and operating tests are set forth at 10 C.F.R. §§ 55.20 through 55.23.[102]

169. By regulation, the Commission has imposed, as a condition of facility operating licenses, the requirement that

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101 Simulator training at the B&W facility typically is divided equally between classroom presentations and actual simulator operation. Rodriguez Testimony at 9, 13.

102 The NRC Staff has specified criteria which increase the scope of the NRC license examinations. See CEC Ex. 49, Enclosure 1 at 4. In addition, Licensee is considering revisions to its hot license training program, in response to NRC Staff guidance, to increase the level of detail for training in heat transfer, fluid flow, thermodynamics and mitigating core damage. Tr. 3075 (Rodriguez); CEC Ex. 49, Enclosures 2 and 3.

the licensee shall have in effect an operator requalification program which shall, as a minimum, meet the requirements of Appendix A to 10 C.F.R. Part 55. 10 C.F.R. § 50.54(i-1). Each operator and senior operator license expires two years after the date of issuance. 10 C.F.R. § 55.32. Requirements for the renewal of operator licenses are set forth at 10 C.F.R. § 55.33, and include successful completion of the requalification program. The requalification training program for licensed personnel at Rancho Seco is conducted continuously and on a two-year cycle. The program includes regularly scheduled lectures,[103] assigned individual study, on-the-job training including reactor control manipulation,[104] an annual one-week simulator course, an annual oral exam administered by Rancho Seco management,[105] and an annual written examination of

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103 During the course of the two-year cycle an average of 60 different hours of lectures are scheduled and repeated to accommodate all licensed operating personnel. Rodriguez Testimony at 11; Tr. 3078-3079, 3087 (Rodriguez). Individuals who score sufficiently high in a particular subject area on the written requalification examination are not required to attend lectures in that subject area. Tr. 3079, 3080 (Rodriguez).

104 Each licensed operator is required to manipulate the controls a minimum of ten times during the term of the license. Each licensed senior operator must manipulate the controls or direct the activities of operators during control evolutions a minimum of ten times during the term of the license. In meeting these requirements, credit is given for control manipulations at the B&W simulator. Rodriguez Testimony at 13, 14; Appendix A to 10 C.F.R. Part 55, at paragraph 3.

105 Members of the NRC's Performance Appraisal Branch testified that Licensee had not fully implemented the training program for licensed operators in that for a couple of operators this oral exam was not administered within the time frame specified by the Rancho Seco administrative procedure.
(footnote continued next page)

comparable scope to the NRC licensing exam. Rodriguez Testimony at 11-15; Wilson Operator Testimony at 4. The Rancho Seco administrative procedure governing the requalification training program may be found in the record as CEC Exhibit 35.[106] Licensee has modified the program to include training on the lessons learned from the Three Mile Island accident. Rodriguez Testimony at 12; CEC Ex. 35 at 3. This program is audited regularly by the NRC's Office of Inspection and Enforcement and the Operator Licensing Branch. Wilson Operator Testimony at 4; Tr. 3813-3815 (Wilson).

170. Special training was provided to the Rancho Seco operators after the accident at Three Mile Island. Item (e) of the short-term actions required by the Commission's Order of May 7, 1979, directed Licensee to "[p]rovide 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." One of the long-term modifications required by the May 7 Order directed as follows:

(continued)
Supplemental Testimony of NRC Performance Appraisal Branch Regarding SMUD Management Controls, following Tr. 4233 ("PAB Testimony"), at 3; Tr. 4254-4256 (Hinckley, Gagliardo). Licensee explained, however, that the delay in administering the exam to one or two operators was because the exam became due, under the administrative procedure, during a plant refueling outage which required the services of the operators. Tr. 3447, 3448 (Rodriguez).

106 The NRC Staff is in the process of implementing additional criteria for requalification programs. See CEC Ex. 49, Enclosure 1 at 5-7.

> 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.

44 Fed. Reg. at 27779 (1979). Both of these requirements were met prior to the restart of the Rancho Seco plant on July 5, 1979. Staff Evaluation at 25, 26; Rodriguez Testimony at 15; Testimony of Robert A. Capra on Implementation of Long-Term Modifications Established in the Commission Order of May 7, 1979 (FOE Contention III(c)), following Tr. 1163 ("Capra Testimony"), at 5, 6.

171. Special post-TMI training of Rancho Seco operators has been addressed by the Board above at paragraph 125. In addition to the special B&W simulator training, this included further training by the Rancho Seco training staff[107] and by General Physics Corporation,[108] a consultant to the District, on the sequence of events and causes of the TMI accident, procedure changes made to reflect the lessons learned from the TMI accident, requirements of NRC IE bulletins, plant modifications made as a result of the TMI accident, small-break

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107 The staff includes a training supervisor and four instructors. In addition, other engineering personnel employed by the District present some of the training lectures. Tr. 3394-3397 (Rodriguez).

108 General Physics Corporation is very active and experienced in providing training support services for the nuclear industry. Tr. 3398, 3399 (Rodriguez).

LOCAs, void formation theory, saturated and subcooling operations curves, initiation and recognition of natural circulation, safety features actuation system operation, auxiliary feedwater system operation, control of the reactor trip relay, clarification of technical specifications, and requirements for notification of the NRC. Rodriguez Testimony at 16-18 and Appendix III. As we found earlier, each licensed operator was tested on this training by Licensee and audited by the NRC Staff. See paragraph 125, supra.

172. Training on the B&W simulator has been an important part of the training provided to the licensed operators at Rancho Seco. The B&W simulator is very similar in design and layout to the Rancho Seco control room. The arrangement of controls, and the types of controls in the areas that deal directly with feedwater control and reactor coolant system control, are essentially identical to those at Rancho Seco. Rodriguez Testimony at 9; Tr. 3854 (Wilson). This is clearly an advantage in the training of Rancho Seco operators. Lewis Testimony at 13; Tr. 3564 (Bridenbaugh). The simulator training provides the opportunity for the operator to participate in plant operations as a control room operator and as a supervisor of control room operators. The simulator has the capability of introducing over 60 individual casualties in reactor plant systems, including the coolant makeup system, the reactor and its instrumentation, the reactor coolant system, the steam and turbine system, the condensate and feedwater

system, and various auxiliary systems. The individual casualties can be combined to create multiple failure accidents or the instructor may fail equipment sequentially. Thus, the simulator gives the operator the opportunity to practice his training and diagnostic skills on complex problems.[109] Rodriguez Testimony at 13, 14. In the case of the post-TMI training, operators were able to observe the course of the various plant parameters while the Three Mile Island accident was demonstrated on the simulator, and in a second simulation to exercise control to mitigate the accident. Id. at 16. Rancho Seco management personnel often are able to observe personally the training of the operators on the simulator, and the Training Supervisor reviews the written report provided by B&W on the performance of each operator. Tr. 3228, 3229 (Rodriguez).

173. CEC Issue 3-1 questions whether personnel adequately understand the mechanics of the facility, basic reactor physics, and other fundamental aspects of plant operation. As we have found, the cold and hot license training programs, and the ongoing requalification program, include instruction in the fundamentals of nuclear technology, and the theory and principles of plant operation. As a part of these training programs, the operators are examined to assess their

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109 During the requalification simulator training, most of the time on the simulator is spent practicing abnormal situations. CEC Ex. 37 at 74, 75.

understanding of nuclear technology fundamentals. Rodriguez Testimony at 23.

174. CEC witness Bridenbaugh concluded his written testimony on this question with the observation that he[110] found ". . . no assurance that SMUD operators have an analytical understanding significantly better than that of the TMI operators." Prepared Direct Testimony of Dale G. Bridenbaugh and Gregory C. Minor Concerning Operator Training and Human Factors Engineering, following Tr. 3496 ("Bridenbaugh-Minor Testimony"), at 9. The careful wording of this conclusion raises several questions: (1) Do the Rancho Seco operators have a level of understanding which, in the opinion of the witness, is better but not "significantly better" than the operators at TMI; (2) Did the witness compare operators at the two facilities on an equivalent basis; (3) Is the comparison based upon adequate personal information; (4) Is the comparison valid today; and (5) Is the absence of "assurance" on the part of the witness a well founded conclusion based on the results of the comparison or a reflection of inadequacies in the comparison.

175. The record does not reveal an answer to the first question posed above by the Board, but there is ample evidence on the remaining questions. Mr. Bridenbaugh's

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110 While the testimony was jointly sponsored, the Board and the parties were directed to Mr. Bridenbaugh for the testimony on operator training and to Mr. Minor for the testimony on human factors engineering. Tr. 3498.

knowledge of the level of analytical understanding of the Rancho Seco operators is based upon his review of the transcripts of the depositions in this proceeding of three licensed Rancho Seco operators, Licensee's report describing the hot license training program, Licensee's administrative procedure governing the requalification program, and Licensee's responses to discovery requests in this proceeding. Tr. 3505 (Bridenbaugh). His knowledge of the TMI training program, on the other hand, is based upon a review of reports published as a part of various studies made of the Three Mile Island accident. There is no indication that Mr. Bridenbaugh personally reviewed testimony by TMI operators or the Metropolitan Edison procedures which govern its training programs. Tr. 3506, 3507 (Bridenbaugh). Consequently, it is clear that the witness did not investigate the analytical levels of understanding of operators at the two facilities to an equivalent extent.

176. While he has reviewed governing procedures, Mr. Bridenbaugh has not looked at the actual training materials used to instruct the Rancho Seco operators, the lesson plans used or the written examinations given; he has not orally examined an operator, observed training of Rancho Seco operators, or compared the training staffs at Rancho Seco and Three Mile Island. Tr. 3508-3510, 3610-3612 (Bridenbaugh). The Board finds that a review of training procedures and other descriptive reports is not, by itself, an adequate basis for

reaching a conclusion on the level of understanding of the individuals who have received the training.[111] Mr. Bridenbaugh himself acknowledged that there could be differences in the quality and content of the training conducted at the two facilities which would not be revealed by his evaluation. Tr. 3537, 3538 (Bridenbaugh). In response to a question on this point by the Board, Mr. Bridenbaugh testified:

> I guess I would say that I haven't had the opportunity to make any extensive qualitative analysis of the two programs. I think in order to do that, for example, you would find it necessary to do many of the things that the NRC does, and that is to -- or should be doing, at any rate, and that is to sit in on training programs and observe them in operation.
>
> I did not have the opportunity to do that. I think your point is a very valid one, though, and that is, you know, that -- comparing absolute hours is not necessarily, you know, a total picture of things . . ..

Tr. 3610, 3611 (Bridenbaugh). Yet, Mr. Bridenbaugh appears to have based his conclusion on the similarity of training at the two facilities largely on a comparison of the number of hours devoted to classroom and simulator training. Tr. 3568-3570 (Bridenbaugh).

177. Mr. Bridenbaugh's comparison also appears to rest upon his finding that Licensee's training program complies with only the letter of existing requirements, and that since

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111 A reading of CEC Exhibit 35, the Rancho Seco procedure on requalification training, for example, does not by itself tell us what the operators actually know and understand.

the TMI program met the same regulations, and they were both approved by the NRC, there is no substantial difference in the training provided to the operators at the two facilities. Bridenbaugh Testimony at 6; Tr. 3534 (Bridenbaugh). The witness went so far as to apply this reasoning to most, if not all, of the utilities with licensed reactors. Tr. 3534 (Bridenbaugh). This basis for comparison suffers from two important flaws. First, there is no evidence to support the hypothesis that since the same regulatory standards apply to all operating licensees, then the training provided to the operators at all of these facilities cannot be substantially different. In fact, for the reasons given above in paragraph 176, even if an investigation showed that each training program in the country was identical on paper, it would not follow that the training experience and the level of understanding of the operators would be the same. Second, Mr. Bridenbaugh's conclusion that the Rancho Seco program meets only the letter of existing requirements reflects a shallow inquiry and is contradicted by the record. The witness could not clearly identify the requirements to which he referred and testified that his conclusion rested on a general impression of the Rancho Seco training program, and not on a careful comparison of the program with existing requirements on a point-by-point basis. Tr. 3517-3523 (Bridenbaugh). The record shows, in fact, that the annual simulator training provided in the requalification program is beyond existing NRC requirements.

Tr. 3230 (Rodriguez); Tr. 3524 (Bridenbaugh). The NRC Staff has found, in the past, that the content of the written requalification examination exceeded requirements. Tr. 3824 (Wilson). And the annual oral requalification exam administered by Rancho Seco management is not a requirement of the NRC. Wilson Operator Testimony at 14; Tr. 3448 (Rodriguez). Finally, the Board observes that many of the requirements of 10 C.F.R. Part 55 are qualitative, and not quantitative, so that they do not lend themselves to a conclusion that only the "letter" has been met. See Tr. 3525 (Bridenbaugh). While the post-TMI training requirements directed by the Commission's Order of May 7, 1979, are specific, they are but one aspect of the training which contributes to the operators' level of analytical understanding.

178. While he did not perform a comparable review of TMI operators, Mr. Bridenbaugh also relies on his review of the Rancho Seco operator depositions for his comparative conclusion. Bridenbaugh-Minor Testimony at 7, 8. The cross-examination of Mr. Bridenbaugh and the Board's own review of the depositions,[112] however, show that the few examples cited by Mr. Bridenbaugh as evidence of a substantial amount of uncertainty and lack of understanding cannot be substantiated from the testimony of the deponents or are irrelevant to CEC Issue 3-1. See, e.g., Tr. 3529-3532 (Bridenbaugh on

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112 The transcripts of the depositions are in evidence as CEC Exhibits 36, 37 and 38.

uncertainty regarding conflicts between procedures and technical specifications); Tr. 3582 (Bridenbaugh unaware whether vessel weldments issue is a resolved concern at Rancho Seco); CEC Ex. 36 at 56 (Bridenbaugh's citation to Tipton deposition does not substantiate uncertainty over any alleged conflicts between procedures).

179. Finally, Mr. Bridenbaugh compared the current Rancho Seco training programs with the training provided to TMI operators prior to the accident. Tr. 3589, 3590 (Bridenbaugh). Since, as we have found, the Rancho Seco operators received additional training after the TMI accident and the requalification program has been modified to provide training on the TMI lessons learned, Mr. Bridenbaugh's comparison is not valid today. <u>See</u> paragraphs 125, 169-172, <u>supra</u>.

180. There is no evidence in the record to support a finding that the Rancho Seco operators receive training which is not substantially different than that provided to the TMI operators. Staff witness Wilson <u>assumed</u> that prior to the TMI accident the training provided to the operators at the two facilities were fairly similar, although he had not made a detailed comparison. Tr. 3811, 3812 (Wilson). Mr. Rodriguez, who established the initial phases of the training program at Rancho Seco, testified that Licensee did not merely adopt the training programs at other plants, but developed its own program. Tr. 3050-3052 (Rodriguez). Mr. Bridenbaugh's comparison is out of date, reflects an unequal examination of

the two programs, is not based upon adequate first-hand knowledge, and, in any case, does not answer the question raised in CEC Issue 3-1. In addition, the only specific recommendation advanced by Mr. Bridenbaugh with respect to the Rancho Seco training program was a suggestion to make the annual requalification simulator training a requirement. Tr. 3613-3615 (Bridenbaugh).[113] Staff witness Wilson from the NRC's Operator Licensing Branch has personally audited Rancho Seco operators and observed them in training. Wilson Operator Testimony at 2; Tr. 3821, 3822 (Wilson). Mr. Wilson testified that on the basis of the tests the NRC has conducted and the requalification training which he has witnessed personally, the Rancho Seco operators adequately understand the mechanics of the facility, basic reactor physics, and other fundamental aspects of its operation. Wilson Operator Testimony at 7; Tr. 3827 (Wilson). The Board finds Mr. Wilson's testimony to be reliable and endorses his conclusion as the Board's finding on CEC Issue 3-1.

181. CEC Issue 3-2 questions whether Rancho Seco operators are properly apprised of relevant new information, including operating experience at other reactors. Licensee

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113 The NRC is already considering a requirement that all operator licensees participate in simulator programs as part of the requalification programs. CEC Ex. 49, Enclosure 1 at 7. In any event, the record shows that all of the licensed operators who stand shift at Rancho Seco receive this training every year. The only occasional exception is for management personnel. Tr. 3397, 3398 (Rodriguez).

receives new information relevant to the safe operation of Rancho Seco from vendors, the NRC, and from the plant's own operating experience. In the case of significant operating events at Rancho Seco, reports prepared for submission to the NRC, if pertinent, are provided to operating crews through the Special Order program.[114] As a result of screening by Rancho Seco management, this information may also be reflected in revisions to operating procedures or communicated in memoranda for reading and information. The periodic issue of licensee event reports by the NRC is distributed to the Rancho Seco Plant Superintendent and Operations Supervisor. Experiences at other units which are deemed by Rancho Seco management to be directly pertinent to plant operation can then be communicated to the operators through the Special Order program or through short lectures by the Operations Supervisor. In addition, B&W produces a weekly summary of occurrences at B&W reactors, which is provided to Rancho Seco operating crews. Finally, the requalification training program, including B&W simulator training, is used to acquaint operators with operating experience at other plants. Rodriguez Testimony at 34, 35.

182. The nuclear industry and the NRC have both undertaken additional efforts, since the Three Mile Island

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114 The Special Order procedure requires that each shift supervisor discuss with his operating crew the content of each order issued. The shift supervisor must document that this discussion was accomplished. Rodriguez Testimony at 32; Tr. 3402 (Rodriguez).

accident, to improve the dissemination and use of nuclear power plant operational data. The Electric Power Research Institute's Nuclear Safety Analysis Center is developing a capability to review systematically available plant event reports and data for transmission to applicable licensees.[115] The new Institute for Nuclear Power Operations is also to review and analyze operating experience and relay this information to licensees for incorporation into their training programs. Wilson Operator Testimony at 10; Rodriguez Testimony at 35. In addition, the Commission has established an agency-wide Operational Data Analysis and Evaluation Office to provide coordination and an overview of all operational data analysis activities performed within the NRC.[116] Wilson Operator Testimony at 9.

183. CEC witness Bridenbaugh's written testimony on this subject emphasizes what he feels is the absence of a procedure requiring that pertinent new information is communicated to operating crews. Bridenbaugh-Minor Testimony at

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115 EPRI provided Rancho Seco with relevant information on the February 26, 1980 event at Crystal River-3, another plant with a B&W NSSS, which was then communicated to the Rancho Seco operators. Tr. 3300, 3301 (Rodriguez).

116 The Board also takes notice of the fact that since the accident at Three Mile Island the Commission has: (1) issued an advanced notice of proposed rulemaking to consider requiring mandatory participation of power reactor licensees in the Nuclear Plant Reliability Data System, 45 Fed. Reg. 6793 (1980); and, (2) amended its regulations to require timely and accurate information from licensees to NRC following significant events at operating nuclear power reactors, 45 Fed. Reg. 13434 (1980).

9, 10. The implementation of a more rigid procedure for the communication of such information was one of only two specific changes Mr. Bridenbaugh identified for the Board's consideration. Tr. 3615, 3616 (Bridenbaugh). On cross-examination, however, Mr. Bridenbaugh testified that the absence of a formal procedure was of the second order, and that the most significant failing was that the TMI accident was the only transient communicated to Rancho Seco operators. Tr. 3545 (Bridenbaugh); Minor-Bridenbaugh Testimony at 9. Yet this "significant" conclusion by Mr. Bridenbaugh was based solely on the testimony of Mr. Tipton, a Rancho Seco operator, who stated at his deposition that he could not recall, at that time, a transient other than TMI which his shift supervisor had discussed with him. Tr. 3545 (Bridenbaugh). See also, CEC Ex. 36 at 96, 97. The Board does not agree that this lack of recollection is proof that no other transients were discussed.[117] We also note that in another deposition shift supervisor Comstock testified that it is among his responsibilities to inform operators of

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117 Mr. Bridenbaugh also testified that no system exists to make NRC NUREG reports readily available to the operators, citing CEC Ex. 36 at 139 (Bridenbaugh-Minor Testimony at 10), even though Mr. Tipton, in the testimony cited, stated that while NUREGs are not in the control room, he could get them if he requested them. See Tr. 3550-3553 (Bridenbaugh). In any event, the Board finds that in the case of the NUREG report about which Mr. Tipton was asked (NUREG-0623, on the reactor coolant pump trip for small-break LOCAs), the operators have a good understanding of the basis for the requirement. See n.70, supra. This is the important point, and not whether operators gain that knowledge through reading NUREG reports or from what we would expect to be more efficient training vehicles. See, e.g., CEC Ex. 37 at 50-55.

significant events at other reactors, and that he or the Operations Supervisor discuss such events with his crew. CEC Ex. 37 at 75, 76. The record also shows that information has been communicated to Rancho Seco operators on recent events at the Crystal River and Oconee plants. Tr. 3300-3302 (Rodriguez).

184. Mr. Bridenbaugh concludes his written testimony on CEC Issue 3-2 with the observation that "[a]t a minimum, there needs to be a means to ensure that new procedures and significant events are promptly communicated to operators in a manner designed to make certain that the events and procedures are thoroughly understood by operators." Bridenbaugh-Minor Testimony at 10. This is a reasonable objective, but the evidence indicates that such means already exist and that improvements are being made. Excessive formality and rigid criteria are not necessarily advantageous here. Licensee's Manager of Nuclear Operations testified that the facility management staff does not want to overload operators with information which is not new or does not add to their understanding of plant operation. Tr. 3305 (Rodriguez). Careful screening of new information by the Manager of Nuclear Operations, the Plant Superintendent and the Operations Supervisor -- who are personally aware of the operators' needs for information, yet sensitive to the overall burdens on the operators -- is in our view superior to the establishment of rigid criteria for the communication of new information. See

Tr. 3446, 3447 (Rodriguez). The Board finds, then, on CEC Issue 3-2, that Rancho Seco personnel are properly apprised of new information pertinent to the facility's safe operation and ability to respond to transients, and particularly of information on the operating experience of other reactors.

185. CEC Issue 3-3 questions whether emergency procedures at Rancho Seco are understood by and available to plant personnel so that they will be implemented effectively in an emergency. Licensee maintains emergency procedures at Rancho Seco in a single volume red binder, distinct from other plant procedures, one copy of which is located in a desk immediately behind the control console in the control room. Consequently, the emergency procedures are available in a manner that allows quick and effective implementation during an emergency. Rodriguez Testimony at 32; Wilson Operator Testimony at 12.

186. Administrative procedures are in place to ensure that the emergency procedures are kept up-to-date. Wilson Operator Testimony at 12. Licensee has changed a number of its emergency procedures since the Three Mile Island accident and made what the Staff believes to be significant improvements to them in response to the Commission's Order of May 7, 1979. Id. at 15; Tr. 3850, 3851 (Wilson). Changes to the emergency procedures normally are communicated to operating personnel through the Special Order program. See n.114, supra. Each licensed operator must review an emergency procedure

change and document completion of that review. Rodriguez Testimony at 32. The emergency procedures are also the subject of training in the requalification program, where operators practice the procedures during simulator training and are selectively tested on them in the annual oral and written examinations. Id. at 33; Wilson Operator Testimony at 12, 14. Through its examination process, the NRC Staff also determines whether emergency procedures are understood by licensed personnel. Wilson Operator Testimony at 13; Tr. 3840-3845 (Wilson).

187. The testimony of CEC witness Bridenbaugh on CEC Issue 3-3 reached no conclusion and made no recommendations. Rather, Mr. Bridenbaugh asserted that there may be confusion over the memorization of emergency procedure immediate action steps and the use of written procedures, and observed that operators need to memorize the immediate action steps. Bridenbaugh-Minor Testimony at 10, 11; Tr. 3561, 3562 (Bridenbaugh). There is no confusion on this point, however. The record shows that the operators do commit the immediate action steps to memory as Mr. Bridenbaugh suggests; after those steps are taken they refer to the written emergency procedures to determine the subsequent actions which should be taken and to verify accomplishment of all of the immediate actions. Tr. 3443 (Rodriguez); Wilson Operator Testimony at 12; Tr. 3842 (Wilson). See paragraph 122, supra. The NRC Staff, on the basis of its examinations conducted at Rancho Seco, is

satisfied that licensed personnel understand the emergency procedures. Wilson Operator Testimony at 13. The Board finds, on CEC Issue 3-3, that the NRC and SMUD adequately assure that emergency instructions are understood by and are available to plant personnel in a manner that allows quick and effective implementation during an emergency.

188. FOE Contention III(e) and part of Board Question H-C 32 question whether there are adequate procedures to determine and test the competence of Rancho Seco operators. As we have found, individuals who manipulate the controls of a nuclear reactor must first be licensed by the Commission, and the issuance of such a license requires the successful completion of an initial licensing examination administered by the NRC Staff. Paragraphs 164, 168, supra. See also, Wilson Operator Testimony at 18. As a part of the requalification training program, Licensee administers annual oral and written examinations to licensed personel. Paragraph 169, supra. See also, Wilson Operator Testimony at 16, 17. Since the accident at Three Mile Island, the Rancho Seco operators have been audited or examined by Licensee, Licensee's consultant training organization and the NRC Staff. Paragraphs 125, 171, supra. See also, Wilson Operator Testimony at 19-21; Rodriguez Testimony at 19-21. There is no evidence that this program of testing has been inadequate.[118] The Board finds that these

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118 CEC witness Bridenbaugh advanced a general observation that assessment of training should be more exact, but his testimony (footnote continued next page)

testing procedures are adequate to determine the competence of Rancho Seco licensed operators to respond to feedwater transients.

189. Through the licensing requirements and training directed by the Commission in its regulations, in the Order of May 7, 1979, and by the Staff, the competence of licensed operators clearly is a matter of regulation by this agency. The cold license, hot license, and requalification training programs at Rancho Seco have been reviewed and approved by the NRC. Wilson Operator Testimony at 3. CEC witness Bridenbaugh is the only witness who questioned the competency of the Rancho Seco operators, although his conclusion merely was that he is not sure they are better than the operators who were at TMI. In the foregoing findings of fact we have concluded that Mr. Bridenbaugh's testimony on that score is not reliable. Further, Mr. Bridenbaugh does not question that Licensee's training of its licensed operators complies with NRC requirements. Rather, he asserts that the NRC standards are inadequate. Bridenbaugh-Minor Testimony at 7, 11 and 12; Tr. 3520, 3570 (Bridenbaugh). To the extent that Mr. Bridenbaugh finds Commission regulations to be in need of revision, his complaint must be taken to the Commission and cannot be entertained by this Board. See 10 C.F.R. § 2.758.

(continued)
does not reflect any personal familiarity with the testing conducted of Rancho Seco operators nor assert any deficiencies in that testing. See Bridenbaugh-Minor Testimony at 11, 12.

190. It is significant that Mr. Bridenbaugh's written testimony did not present, and that on cross-examination he could not recall, a specific instance in which a Rancho Seco operator has demonstrated a lack of understanding of what would have to be done to respond to a feedwater transient. See, e.g., Tr. 3586, 3587 (Bridenbaugh). NRC Staff witness Wilson, who has audited and given licensing examinations to Rancho Seco operators, observed requalification training of Rancho Seco operators, and who has examined hundreds of operators at other plants over a period of six and one-half years, testified that the Rancho Seco operators stack up very favorably with other operators in training programs with which he has experience. Tr. 3878-3881 (Wilson). The Board finds, on the basis of the foregoing findings of fact and in partial response to Board Question H-C 34, that the training actions required by the Commission's Order of May 7, 1979, in the context of the other training provided, provide reasonable assurance that the licensed operators at Rancho Seco will operate the plant safely in response to feedwater transients.

191. The remainder of Board Question H-C 34 questions whether unlicensed operating personnel will respond adequately to feedwater transients. Unlicensed operators at Rancho Seco assist the licensed operators by starting and stopping motorized equipment, opening and shutting valves, conducting periodic maintenance or checking of equipment, and maintaining plant records. These various activities are

directed and supervised by the licensed operators, who assist the unlicensed personnel if necessary. Written procedures are located at equipment operating stations to instruct these personnel in their assigned tasks. Unlicensed personnel are allowed to manipulate apparatus and mechanisms which may affect reactivity and the power level of a nuclear power plant only under the direct supervision of a licensed operator present at the controls and only for purposes of training such individuals to obtain necessary experience to become licensed.[119] NRC Staff Testimony of Philip J. Morrill on Training of Unlicensed Operators (Board Question 34), following Tr. 4141 ("Morrill Testimony"), at 3.

192. The role of unlicensed operators, however, is minimal in operating the Rancho Seco plant safely in response to a feedwater transient. The auxiliary feedwater system, required in the event of a loss of main feedwater, can be operated from the Rancho Seco control room by licensed operators. In the event that the 24-hour water supply for the auxiliary feedwater system in the condensate storage tank reached a low level, unlicensed operators might be called upon to operate manual valves outside the control room to align the off-site water supply to the auxiliary feedwater pumps. See paragraph 144, supra. Unlicensed operators have been given training, since the Three Mile Island accident, to perform this

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119 See 10 C.F.R. § 55.9(b).

manual valving. Each shift supervisor has conducted specific training for the unlicensed operators on his crew, including a "walk through" to affirm valve location and operation, to assure that they can locate and reposition the valves in the unlikely event they are directed to do so to assure an adequate supply of auxiliary feedwater. Unlicensed operators have also been instructed on the proper procedure for taking local control of the auxiliary feedwater system control valve to each steam generator in the unlikely event of a loss of control to all four of the available auxiliary feedwater level control valves.[120] Rodriguez Testimony at 37, 38; Morrill Testimony at 5; Tr. 4224, 4225 (Morrill).

193. CEC witness Bridenbaugh testifed, on unlicensed operator training at Rancho Seco, that "on-the-job" training means unlicensed operators may not know how or where to perform certain actions the first time they are called upon to perform them, citing the deposition of Rancho Seco licensed senior operator Tipton (CEC Ex. 36 at 113, 114). Bridenbaugh-Minor Testimony at 13. Mr. Tipton, however, did not testify that unlicensed operators may not know how or where to perform certain actions the first time they are called upon to perform them. To the contrary, he testified that ". . . they are instructed either the first time they have to do a task or

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120 Unlicensed operators also receive more general training aimed at their other duties at the plant. See Morrill Testimony at 5-8.

again if they need refresher." CEC Ex. 36 at 113, 114. On cross-examination, Mr. Bridenbaugh explained his reliance upon this testimony for his conclusion as "an interpretation" of Mr. Tipton's statement in the context of the interchange. Tr. 3574 (Bridenbaugh). Mr. Bridenbaugh then testified, surprisingly, that ". . . the point of my testimony is not that he may not know how to do things the first time he is called upon to do them", Tr. 3575 (Bridenbaugh), but that the absence of a formal training program increases the probability that the unlicensed operators will not know how to perform a task. Tr. 3577, 3578 (Bridenbaugh). In short, Mr. Bridenbaugh could neither explain nor support his testimony, which is refuted by the evidence.

194. Concluding the inquiry into Board Question H-C 34, the Board finds that there is reasonable assurance that unlicensed operating personnel will respond adequately to feedwater transients.

195. FOE Contention III(d) and part of Board Question H-C 32 question whether there are adequate procedures to determine and test the competence of Rancho Seco or facility management. The Commission reviews the technical qualifications of applicants for facility operating licenses to engage in the proposed activities in accordance with Commission regulations. See 10 C.F.R. § 50.40(b). The NRC reviews the management and technical organization of the applicant and its technical contractors to assure that on-site facility management and personnel are qualified to act responsibly and

competently in the event of an emergency or abnormal occurrence at the plant, to assure that clear management control and effective lines of communication exist between the organizational units involved in the management, operation, and technical support for the operation of the facility, and to assure that the applicant has the necessary technical support for the operation of the facility. Allenspach Testimony at 2-5. The District's organizational structure, personnel requirements and technical qualifications were reviewed and found to be acceptable in a Safety Evaluation, dated June 8, 1973, issued in support of the operating license application for Rancho Seco.[121] Id. at 5, 6.

196. The key members of Rancho Seco management -- the Manager of Nuclear Operations, Plant Superintendent, Engineering and Quality Control Supervisor, Chairman of the Plant Review Committee and Operations Supervisor -- each have a senior operator license issued by the NRC. They each participated in the Rancho Seco cold license training program, in the special post-TMI training, and continue to participate in the requalification training program. See paragraphs 166, 169-172, supra. Consequently, as licensed senior operators, these facility management personnel are regularly trained and tested on their knowledge and competence to operate the plant safely.

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121 The NRC is considering upgraded requirements in the area of management and technical capabilities for licensees of operating reactors. Allenspach Testimony at 7-9.

Rodriguez Testimony at 19, 20. See also, Johnson Testimony at 4. In addition, Rancho Seco management and supervisory personnel have begun participation in a command and control training program, being presented by a consultant to the District, which will provide further training in the command and control aspects of mitigating various accidents. Rodriguez Testimony at 20, 21; Tr. 3385, 3386 (Rodriguez).

197. Three reactor inspectors from Region V of the NRC's Office of Inspection and Enforcement testified on the competency of SMUD to operate the Rancho Seco facility. Mr. Johnson has had responsibilities with respect to inspection at Rancho Seco for about nine years. He was the responsible NRC inspector for Rancho Seco during the final phase of construction and during preoperational and power ascension testing. Since that time he has assisted the responsible inspectors in performing the routine NRC inspection program and was the responsible inspector from January, 1979, to August 1, 1979, when the current resident inspector assumed his duties at the Rancho Seco site. Johnson Testimony at 2, 3. Mr. Johnson testified that, based on Licensee's operating activities from January 1, 1978, through July 31, 1979, ". . . the personnel operating and managing the Rancho Seco Nuclear Generating Station have demonstrated their willingness to evaluate any and all problems brought to their attention and have shown their capability and fitness to operate the station safely." Id. at 6; Tr. 3997, 3998 (Johnson). Mr. Johnson described the

reportable events and items of noncompliance which had occurred during that time frame and the manner in which Licensee responded to the identified problems. Johnson Testimony at 7-10. Based upon this record, and his experienced judgment and personal knowledge of the SMUD operation, Mr. Johnson concluded as follows:

> During the many discussions I have had with personnel of SMUD and its contractors along with opportunities to observe personnel response to planned and unplanned events, my conclusion is that the SMUD organization and personnel are competent to safely operate the Rancho Seco nuclear generating station.

Id. at 11. See also, Tr. 4028-4029, 4085-4086 (Johnson).

198. Mr. Zwetzig was the Project Manager for Rancho Seco in the NRC Division of Operating Reactors from February, 1978, until approximately February, 1979. Since August 1, 1979, he has been the back-up inspector for Rancho Seco. Mr. Zwetzig reported on items of noncompliance and reportable events at Rancho Seco from August 1, 1979, to January 31, 1980, and endorsed Mr. Johnson's testimony, described above, on the competence of the SMUD organization and personnel. NRC Staff Testimony of Gerald B. Zwetzig Relative to the Competency of SMUD to Operate the Rancho Seco Facility (FOE Contention III(d) and Board Question 32), following Tr. 3920; Tr. 3918 (Zwetzig). Mr. Canter has been the NRC's Senior Resident Inspector at the Rancho Seco site since August 1, 1979. Prior to that assignment, he had inspected Rancho Seco as an assistant to the principal inspector. Mr. Canter also discussed enforcement

activities with respect to Rancho Seco from August 1, 1979, to January 31, 1980, and reported on items of noncompliance and reportable events since January 31, 1980. Mr. Canter also endorsed Mr. Johnson's testimony on the competence of the SMUD organization and personnel, and testified that he had personally witnessed this demonstrated competence during a reactor trip and in normal plant operations and maintenance since August, 1979. NRC Staff Testimony of Harvey L. Canter Relative to the Competency of SMUD to Operate the Rancho Seco Facility (FOE Contention III(d) and Board Question 32), following Tr. 3920.

199. By coincidence, the Performance Appraisal Branch of NRC's Office of Inspection and Enforcement had completed the major portion of a management appraisal inspection at Rancho Seco while these hearings were in session. Because preliminary findings from this inspection resulted in a number of concerns to the performance appraisal team which might be relevant to the issues before the Board, two members of the team testified on the team's preliminary findings.[122] The performance appraisal team's concerns relate to management controls in seven of the eleven functional areas reviewed, PAB Testimony at 2, some of which have only tangential, if any, relevance to the subject matter of this proceeding.

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122 The witnesses prefaced their testimony by observing that these preliminary concerns may be resolved by the subsequent inspection and review efforts of the team. PAB Testimony at 1.

200. In the area of training, the witnesses testified that Licensee had not fully implemented requirements in its own procedures for the training of non-licensed personnel. PAB Testimony at 3, as corrected at Tr. 4252 (Hinckley). This observation, however, did not go to the unlicensed operators on the Rancho Seco operating crews which the Board addressed above in its findings on Board Question H-C 34, but to maintenance and other technical staff personnel. Tr. 4276, 4277 (Hinckley). The testimony that the training program for licensed operators had not been fully implemented, PAB Testimony at 3, is addressed above in footnote 105.

201. In the area of design changes and modifications, while the testimony was not specifically related to systems at issue here, the witnesses from the Performance Appraisal Branch testified that Licensee's procedures for the review of design changes to Class I systems pursuant to 10 C.F.R. § 50.59 did not comport with the requirements of the license technical specifications in that a second level safety evaluation was not provided where the first level of review by the Supervisor of Engineering and Quality Assurance made a negative determination under 10 C.F.R. § 50.59. PAB Testimony at 3; Tr. 4274 (Gagliardo). Licensee described its procedures for such reviews and disagreed with the performance appraisal team's conclusion that they did not comply with the technical specifications. Tr. 3448-3450 (Rodriguez). Mr. Johnson, an inspector from Region V of the Office of Inspection and

Enforcement, also described the procedure for such changes required by 10 C.F.R. § 50.59 and the Rancho Seco technical specifications. Tr. 4118, 4119 (Johnson). He is familiar with Licensee's procedures for reviewing design changes to Class I systems, and testified that his position and the position of Region V, which has been communicated to Licensee, is that the District's procedure complies with the technical specifications.[123] Tr. 3921, 3922 (Johnson). In any event, inspectors from Region V and the performance appraisal team reviewed some 176 design changes implemented pursuant to a negative § 50.59 determination by the Supervisor of Engineering and Quality Assurance and found that all of the determinations were correct. Tr. 4275 (Gagliardo); Tr. 4276 (Hinckley).

202. The Performance Appraisal Branch concerns are involved with Licensee's management control systems in the areas reviewed by its team. The fact that concerns exist with a licensee's management controls does not indicate that the licensee's management is not competent to manage their reactor facility. PAB Testimony at 1, 2; Tr. 4270 (Gagliardo). While the team spent approximately one man-year of effort in its investigation of Rancho Seco (Tr. 4234 (Hinckley)), it found no

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123 This differing interpretation by the I&E regional office and its Performance Appraisal Branch apparently persists. See Tr. 4275 (Gagliardo). The Board recognizes, and is mindful in attaching weight to statistics on enforcement activities, that licensees may face different interpretations of a single requirement by different NRC inspectors. See Tr. 3971-3973 (Johnson, Zwetzig); Tr. 4280 (Gagliardo).

weaknesses in management controls which warranted immediate action. PAB Testimony at 5; Tr. 4271, 4272 (Hinckley). Significantly, the team found no concerns in the area of plant operations -- i.e., the manner of instructing operating crews, defining responsibilities, providing for communication between operations personnel and management, and providing for management knowledge of problems in the field and their resolution. Tr. 4235, 4236 (Hinckley).

203. The record on Licensee's management competence, an issue raised by the intervenors who have withdrawn from this proceeding, is complete and uncontradicted. The Board and the parties have examined at length Licensee's Manager of Nuclear Operations, the senior management person on the facility site. Five inspectors from the NRC Office of Inspection and Enforcement have testified, providing a wealth of historical and ongoing evaluations of the competence of Rancho Seco's management to operate the plant safely. No testimony has been presented challenging that competence. Returning to FOE Contention III(d) and Board Question H-C 32, we find that as licensed senior operators, Rancho Seco management is tested for knowledge and competence in operating the plant. While "tests" in the formal sense are not given on the broader subject of managerial skills and capabilities, the extensive evaluations by the NRC Office of Inspection and Enforcement have been more than adequate for reaching a determination on the competence of facility management. The Board finds that the Rancho Seco

facility management is sufficiently competent to provide reasonable assurance that the plant will respond safely to feedwater transients.

J. Instrumentation

Board Question CEC 5-3a: Are the special features and instruments installed at Rancho Seco adequate to aid in diagnosis and control after an 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?

204. The Rancho Seco plant is designed to respond safely to a loss of feedwater transient basically by means of three systems: the integrated control system, the reactor protection system and the auxiliary feedwater system. The ICS is designed to initiate a runback (i.e., a reduction in power) of the reactor and turbine to within the capacity of the remaining feedwater in the event of a partial loss of feedwater. NRC Staff Testimony of Bruce A. Wilson on Instrumentation for Diagnosis and Control of Off-Normal Conditions (CEC Issue 5-3a), following Tr. 3788 ("Wilson Instrumentation Testimony"), at 3. See also, paragraph 39, supra. The reactor protection system (in response to the

anticipatory trip) will shut down the reactor in the event of a loss of both feedwater pumps, and the auxiliary feedwater system is designed to start automatically and deliver water to the steam generators for decay heat removal following the loss of main feedwater and reactor shutdown. Wilson Instrumentation Testimony at 4.

205. Information necessary for the operator to diagnose and respond to a loss of feedwater transient includes: (1) the extent of the loss of feedwater (i.e., whether one or both pumps have been lost or whether control of feedwater flow has been lost); (2) whether the ICS is responding as required; (3) whether the reactor protection system has been called upon to shut the plant down; and, (4) whether the auxiliary feedwater system, if required, is functioning as designed. Id. As a result of its review of the Three Mile Island accident, the NRC Staff decided that operators should have additional indication of auxiliary feedwater flow beyond the already available steam generator level indication. Consequently, as one of the short-term actions undertaken to upgrade the timeliness and reliability of the auxiliary feedwater system in response to the Commission's Order of May 7, 1979, Licensee installed, prior to the restart of Rancho Seco, flow meters on each of the auxiliary feedwater lines to give a more direct indication in the control room of the existence and amount of auxiliary feedwater flow to each steam generator. Id. at 5; Tr. 1546-1549 (Novak, Matthews). See paragraphs 142 and 143,

supra. Various other additions to and modifications of the instrumentation in the Rancho Seco control room have been implemented since the Three Mile Island accident which go beyond the requirements of the Commission's Order of May 7, 1979. Tr. 2962-2963, 3351-3356 (Rodriguez).

206. Specifically, feedwater transient diagnostic instrumentation is available to the operators in the Rancho Seco control room to provide indication of the following parameters: auxiliary feedwater flow; reactor coolant system hot leg, cold leg and average temperature; steam generator level (six channels); steam generator outlet pressure; pressurizer level (three separate temperature compensated level indication channels); reactor coolant system makeup flow; reactor coolant pressure (four narrow range channels and three wide range channels); main feedwater flow to each steam generator; high pressure injection flow; and reactor coolant system loop flow. Rodriguez Testimony at 41, 42, as amended at Tr. 3351 (Rodriguez).

207. As an additional operator aid, two saturation meters were installed in the Rancho Seco control room during the 1980 refueling outage and are now in operation. These meters provide the operator with a continuous and direct display of the amount of subcooling present in the reactor coolant system, which the operator previously determined through a comparison of pressure and temperature to a saturation curve. Tr. 3405 (Rodriguez); Rodriguez Testimony at 44,

45. In partial response to Board Question H-C 22, then, instrumentation is available to give positive indication as to whether or not the coolant is subcooled throughout the core at all times.[124] Each meter receives a wide range pressure signal of 0-2500 pounds from the safety features instrumentation and two hot leg temperature inputs (a range of 120-920°F; one from each reactor coolant loop). The meter itself auctioneers and selects the highest temperature reading it receives, and feeds the temperature and pressure data into a computer for a calculation of subcooling in degrees Fahrenheit. The meter displays to the operator the number of degrees Fahrenheit of subcooling. Tr. 3422, 3423 (Rodriguez); Rodriguez Testimony at 47; Testimony of Paul E. Norian on Adequacy of Pressurizer Instrumentation (Board Question 22), following Tr. 1163 ("Norian Instrumentation Testimony"), at 5. CEC witness Minor, in his written testimony, recommended the installation of a saturation meter at Rancho Seco (Bridenbaugh-Minor Testimony at 16, 17 and 19) because he was unaware of the installation at the time he prepared his testimony. Tr. 3593, 3594 (Minor).

208. Operators at Rancho Seco have the capability to control the following equipment from the control room if needed in response to a feedwater transient: both auxiliary feedwater

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124 It is possible that at the hottest point in the core there might be some local boiling which is not detected by existing instrumentation used to measure subcooling. These bubbles, however, would be capable of mixing and condensing with the surrounding fluid and would not threaten core cooling. Tr. 1141 (Jones).

pumps (startup and shutdown); normal and safety features actuated flow control feedwater valves in the auxiliary feedwater system; normal main feedwater pump turbines; normal steam generator feedwater flow control valves; pressurizer heaters; all three high pressure injection pumps; high pressure injection control valves and makeup control valve; and main feedwater isolation valves. Rodriguez Testimony at 42, 43.

209. While the pressurizer level instrumentation at Three Mile Island provided a reliable indication of pressurizer level during the accident, under the specific conditions of that accident the pressurizer level did not accurately indicate the status of the primary system inventory. During the TMI-2 accident the pressurizer PORV was stuck open and provided a leakage path for the primary system fluid. Subcooling was lost within a few minutes and the coolant began to flash. Since the leakage path was at the top of the pressurizer, there was an insurge of fluid from the hot leg which maintained a large inventory in the pressurizer. Consequently, the pressurizer level was maintained even though the primary system inventory was continuously depleted until the PORV block valve was closed. Norian Instrumentation Testimony at 3, 4. It is undisputed that pressurizer level provides an accurate indication of primary system inventory and reactor vessel level when the primary system fluid is subcooled, but that under saturation conditions significant voids may exist in the system which are not indicated by the pressurizer level indication. Id.; Tr. 933 (Jones); Tr. 1369 (Norian).

210. First, however, it should be noted that the pressurizer level indication at Rancho Seco covers the normal operating level range of the pressurizer and provides sufficient margin above and below that operating range to allow the operators time to restore pressurizer level in the event of an off-normal condition. This level indication also provides the operators with low and high level alarms to annunciate the occurrence of an off-normal condition. Rodriguez Testimony at 46. Pressurizer level, reactor coolant system temperature and reactor coolant system pressure indications (supplemented by the newly installed saturation meters) enable the operators to diagnose whether adequate core cooling is maintained and whether the reactor coolant system is in a subcooled condition. Id. at 44.

211. By maintaining the reactor coolant system pressure and temperature within the allowable operating range, the operator is assured that the reactor vessel is in a solid water condition without any significant vapor. Rancho Seco Emergency Procedure D.5, "Loss of Reactor Coolant/Reactor Coolant System Pressure" (CEC Ex. 46), provides specific guidance to the operator to maintain the reactor coolant in a subcooled condition in the event of a loss of coolant accident. Id. at 47. High pressure injection control from the control room allows the operator to add inventory as necessary to maintain reactor coolant system pressure and to promote adequate subcooling. Id. at 44. By maintaining a minimum of

50°F subcooling in the reactor coolant system and operating high pressure injection pumps to provide an indicated level in the pressurizer -- as the procedures direct, and as all of the post-TMI training has taught the operator to do -- void formation in the reactor coolant system will be prevented. Id. at 47. Because saturation conditions will occur before the core becomes uncovered, Tr. 1755 (Norian), the key indication the operator needs to guide his actions is the existence or loss of subcooling. It is this condition, and not reactor vessel level, which dictates required operator actions. Tr. 1755, 1756 (Capra); Tr. 800-802 (Jones). In the event conditions degrade to the point where voids are formed, the operator can recognize adequate core cooling by observing installed in-core temperature thermocouples which are located at the top of the reactor core. Rodriguez Testimony at 47, 48; Tr. 1369, 1370 (Norian); Tr. 3331 (Rodriguez). In answer, then, to the last portion of Board Question H-C 22, there is no instrumentation which gives reliable information on the water level in the core when the primary coolant is not subcooled. However, there is no evidence to indicate that the operators need such information to undertake the required immediate actions -- which are dictated by the presence or absence of subcooling, and not by vessel level. See Tr. 1370-1372 (Norian).

212. The NRC Staff and Licensee, however, are studying the existing instrumentation for inadequate core

cooling to determine if any supplemental devices, such as reactor vessel water level indication, should be installed. Norian Instrumentation Testimony at 5; Tr. 1369 (Norian). To date, Licensee has not found a design which it believes meets the criterion of unambiguous indication. Tr. 3332 (Rodriguez). CEC witness Minor testified that "the ability to quickly diagnose the Rancho Seco plant would be enhanced" by the installation of vessel level indication, Bridenbaugh-Minor Testimony at 17, although he was not able to put forward a readily available method of accomplishing this goal. Tr. 3600 (Minor). Rather, he suggested that it be carefully studied. Id. CEC witness Lewis, on the other hand, does not recommend core level indicators. Tr. 484 (Lewis). Dr. Lewis believes that void detectors at high points in the primary system would be more useful. Id. Licensee, however, testified that a void detector would not give information, beyond what already exists, upon which to base a response which is different than the actions based upon existing instrumentation. Tr. 872-874 (Jones); Tr. 3333, 3334 (Rodriguez). NRC Staff witness Wilson testified that vessel level indication is not required and that it might even induce inappropriate operator action. Tr. 3877, 3906-3907 (Wilson). On the question of void detectors, Mr. Wilson testified that present operating guidelines and instrumentation are adequate. Tr. 3892, 3893 (Wilson). Obviously, the question of improved instrumentation for inadequate core cooling is unsettled. The important point, however, is that the existing instrumentation is sufficient for the operators to evaluate the state of the primary coolant system and initiate corrective action as needed. Any additional instrumentation to be installed would provide backup to the existing systems and provide further assurance that the core is adequately cooled. Norian Instrumentation Testimony at 6.

213. CEC witness Minor also suggested the need for a dedicated indication of natural circulation. Bridenbaugh-Minor Testimony at 16, 19. On examination by the Board, however, Mr. Minor testified that he did not know whether it was practical to measure such small flow rates, and that additional study would be required to ensure that his proposal would be satisfactory under various conditions. Tr. 3619 (Minor). Licensee's witness testified that existing temperature instrumentation is adequate to verify natural circulation, and the Staff's witness questioned the practicality of a natural circulation meter. Tr. 3444 (Rodriguez); Tr. 3894, 3895 (Wilson). Beyond the fact that the record does not support the imposition of a requirement for such additional instrumentation, the Board has already found that the difficulty in achieving natural circulation during the Three Mile Island accident was not caused by an inability to recognize or verify that mode of cooling. See paragraph 104, supra.

214. While the Board raised questions here about the adequacy of instrumentation at Rancho Seco and a few witnesses made specific suggestions of their own, no party before or

during the hearing advocated the need for any specific modifications. CEC witness Lewis testified that ". . . at Three Mile Island enough information was available on the panel to have allowed those operators to diagnose that event early in the game." Tr. 524 (Lewis). The record here does not support a finding by this Board recommending that the Commission amend its Order of May 7, 1979, to require the installation of any specific additional instrumentation. The Board finds, in answer to Board Question CEC 5-3a, that the existing instrumentation at Rancho Seco is adequate to aid in diagnosis and control after an off-normal condition engendered by a loss of feedwater transient. See Wilson Instrumentation Testimony at 5. Any improvements in instrumentation, we believe, should be addressed by the Commission as a part of its integrated TMI-2 Action Plan.

K. 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?

215. The configuration of the control room has very little effect on whether or not a loss of feedwater transient will occur. NRC Staff Testimony of Bruce A. Wilson on Control Room Design (Board Question 31), following Tr. 3788 ("Wilson Control Room Testimony"), at 2; Rodriguez Testimony at 39. The configuration of the control room could have an effect,

however, on the operators' ability to diagnose and respond to a loss of feedwater transient. Wilson Control Room Testimony at 3.

216. The Rancho Seco control room configuration includes a compact set of 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. Rodriguez Testimony at 40. Licensee's witness was questioned at some length about a report entitled "Human Factors Review of Nuclear Power Plant Control Room Design," published in 1976 by the Electric Power Research Institute. One of the five plants reviewed in this study was Rancho Seco, and the study apparently found some weaknesses as well as many strengths in the control room design at Rancho Seco. Tr. 2965-3004, 3017-3031, 3438-3442 (Rodriguez).[125] CEC witness Minor testified that while it has some weaknesses, the Rancho Seco control room appears to have several significant advantages from a human factors point of view compared to the TMI-2 control room and that, on the whole, the advantages appear to outweigh the disadvantages. Bridenbaugh-Minor Testimony at 17. NRC Staff witness Wilson testified that, on the basis of a comparison with other control rooms, it appears Licensee devoted

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125 Licensee has contracted for a human factors study of its control room to be undertaken this year. Tr. 2974 (Rodriguez).

considerable attention to its control room design. Mr. Wilson believes the Rancho Seco control room to be far superior to the one at TMI-2, and testified that he ". . . would rate the Rancho Seco control room design among the best." Wilson Control Room Testimony at 4, 5. See also, Tr. 3877, 3878 (Wilson).

217. In response to Board Question H-C 31, the Board finds that there are no features of Rancho Seco's control room design and configuration which make it difficult for operators to avoid, or to diagnose and respond to, a loss of feedwater transient.

L. Long-Term Modifications

FOE Contention 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.

218. In addition to the short-term actions which were completed prior to the restart of Rancho Seco on July 5, 1979, the Commission directed Licensee to accomplish as promptly as practicable certain long-term modifications to further enhance the capability and reliability of the reactor to respond to various transient events. Commission Order of May 7, 1979, 44 Fed. Reg. 27779, 27780 (1979). See paragraph 7, supra, and Capra Testimony at 3. It was the judgment of both the Commission and Licensee that these modifications,

unlike the short-term actions, were not required immediately to provide reasonable assurance that Rancho Seco would respond safely to feedwater transients. Dieterich Testimony at 25; Capra Testimony at 8. No witness who appeared at this hearing testified that this judgment was in error. And, while the Commission directed the prompt implementation of the long-term modifications, some of them require detailed engineering analysis and assessment by Licensee, review by the NRC Staff, procurement of components and equipment (some of which may require extensive lead time), installation of equipment and additional training of operators. Capra Testimony at 7. Consequently, these modifications could not reasonably have been subjected to a specified, rigid implementation schedule dictated by the Commission on May 7, 1979. Licensee and the NRC Staff, however, both presented testimony on the status of implementation of the long-term modifications.

219. The first of the four specific long-term modifications directed by the Commission's Order of May 7, 1979, required Licensee to provide the NRC Staff a proposed schedule for implementation of identified design modifications which specifically related to the nine short-term actions undertaken to increase the reliability of the auxiliary feedwater system. The nine AFW actions undertaken as a part of item (a) of the short-term actions were completed prior to the restart of Rancho Seco on July 5, 1979. See paragraphs 137-147, supra, and Capra Testimony at 3, 4. Licensee's review

of these actions indicated that the AFW system long-term design requirements were satisfied as a result of these short-term actions. Capra Testimony at 4. There is no evidence to contradict Licensee's position that it has complied with the first of the four long-term modifications. In addition, however, Licensee undertook to study systematically the reliability of its AFW system. As a result of this study, Licensee provided the NRC Staff with an identification of additional planned modifications to the Rancho Seco AFW system and a schedule for completing these modifications. The Board has found that these changes, coupled with other AFW modifications imposed by the Staff as a result of its evaluations of the TMI-2 accident, will further improve the system's reliability. <u>See</u> paragraphs 149-156, <u>supra</u>, and Capra Testimony at 4. <u>See also</u>, CEC Ex. 21.

220. The second long-term item required Licensee to submit a failure mode and effects analysis of the integrated control system. Licensee submitted such an analysis in August, 1979. Dieterich Testimony at 26; Capra Testimony at 4, 5. The NRC Staff and the Board have determined that this analysis is complete. <u>See</u> paragraphs 52-60, <u>supra</u>.

221. The third long-term modification required Licensee to upgrade to safety grade the anticipatory reactor trip upon loss of main feedwater and/or trip of the turbine. The NRC Staff approved Licensee's preliminary design for the proposed upgrade on December 20, 1979, which allowed Licensee

to proceed toward installation. Dieterich Testimony at 26; Capra Testimony at 5. When the record in this case closed, Licensee was proceeding toward the prompt implementation of this modification. See paragraph 84, supra.

222. The fourth long-term item required Licensee to have two licensed operators on shift with TMI-2 simulator training by June 1, 1979, with at least one operator with such training assigned to the control room. This was accomplished, along with the short-term actions required by the May 7 Order, prior to the restart of Rancho Seco on July 5, 1979. Capra Testimony at 5, 6; Dieterich Testimony at 27. See also, paragraph 170, supra.

223. Contrary to FOE Contention III(c), then, the Board finds that the Commission's Order of May 7, 1979, did not fail to provide reasonable assurance of adequate safety because of the absence of a reasonable time for implementing the long-term modifications. The Commission directed that these modifications be accomplished as promptly as practicable, but could not have specified at the time of the order a rigid schedule for completion. The long-term modifications are not required to provide reasonable assurance that Rancho Seco will respond safely to feedwater transients, but are intended to further enhance that capability. In any event, the record shows that three of the four long-term modifications have now been accomplished and that the remaining modification is in the process of being implemented.

M. Hydrogen Control

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 TMI-2?
b. proper radiological protection of the surroundings if purging is depended upon?

224. Feedwater transients will not normally result in hydrogen being produced inside the containment building. NRC Staff Testimony of Thomas A. Greene on Hydrogen Recombiner (Board Question 20), following Tr. 2783 ("Greene Hydrogen Testimony"), at 5; Dieterich Testimony at 20. Unless a LOCA occurs, or a feedwater transient is aggravated by subsequent equipment failures and/or human errors, there is no need to put into effect measures for controlling hydrogen because there is no mechanism for hydrogen to be generated or released to the containment. Id.

225. In the event of a severe LOCA resulting in core uncovery, hydrogen gas may be generated in the containment through one or more of the following mechanisms: (1) a chemical reaction between the zircalloy fuel rod cladding and steam; (2) corrosion of containment materials by alkaline core spray solutions; (3) radiolytic decomposition of the cooling water in the vicinity of the reactor core and in the containment sump. Greene Hydrogen Testimony at 3. The hydrogen generated by the rod cladding-steam reaction occurs rapidly, in

a matter of minutes; the other two mechanisms generate hydrogen slowly, over a period of days. Id. at 3, 4.

226. Hydrogen accumulation in the containment is a safety concern because hydrogen is flammable. The lower flammability limit of hydrogen is reached when the hydrogen concentration reaches about four percent of the containment volume. Tr. 2176 (Dieterich); Greene Hydrogen Testimony at 4. If the concentration increases to about 12 percent of the containment volume, the detonation point is reached and large amounts of hydrogen may combine suddenly with the oxygen in the containment atmosphere producing an explosion. Tr. 2176, 2177 (Dieterich).

227. There are two methods considered acceptable by the NRC for removing hydrogen from a containment and thus reducing the possibility of a hydrogen fire or explosion. One method uses a "hydrogen recombiner", which is a device that causes hydrogen and oxygen to react chemically to form water vapor and thereby reduces the hydrogen concentration. Greene Hydrogen Testimony at 2; Dieterich Testimony at 20. A second method is to "purge" the hydrogen, i.e., to release it to the atmosphere outside containment after passing it through a filtered purge system. Greene Hydrogen Testimony at 2, 3; Dieterich Testimony at 20. If a recombiner is available, it can be put into use as soon as hydrogen starts accumulating in the containment, because the containment atmosphere is never released to the environment, but only pumped to the recombiner

and circulated back to the containment building. Tr. 2842-2844 (Greene).[126] On the other hand, if a hydrogen purge system is used, purging must take place at a point in time far enough into the accident that the short-lived radioactive isotopes have decayed so that the radiological exposure of the public is minimized. Dieterich Testimony at 20; Tr. 2843 (Greene).

228. The Rancho Seco facility has a hydrogen purge system but does not have a recombiner. Greene Hydrogen Testimony at 2; Dieterich Testimony at 21. The purge system consists of a mixing system,[127] a sampling system, piping, valves, instrumentation, filters to absorb radioactive iodine, and blowers to vent the hydrogen to the atmosphere outside containment. Tr. 2862-2864 (Greene); Tr. 2151, 2152 (Dieterich). The purge system at Rancho Seco is designed to accommodate at least five times the amount of hydrogen generated in the design basis accident, which is a LOCA in which one percent of the zircalloy cladding reacts with steam to generate hydrogen.[128] Tr. 2156, 2157 (Dieterich); Greene

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126 This ability to put the recombiner in use immediately is only of limited value since, as will be seen in paragraph 232 below, early hydrogen releases from the zirconium-steam reaction occur so rapidly and in such large amounts that a recombiner will be unable to process them until many days into an accident comparable to that which occurred at TMI-2. Tr. 2844 (Greene).

127 The mixing system is intended to ensure that the hydrogen concentration is uniform in the containment atmosphere to avoid the formation of localized high hydrogen concentrations or "pockets." Rancho Seco utilizes the fan coolers and the spray system to distribute the hydrogen uniformly in the containment. Tr. 2864 (Greene).

128 Thus, the purge system at Rancho Seco has the ability to
(footnote continued next page)

Hydrogen Testimony at 3. Possession of a hydrogen recombiner is not required at plants, such as Rancho Seco, for which a construction permit was noticed for hearing prior to December 22, 1968, and whose combined radiation dose produced by purging and the postulated LOCA is less than 10 C.F.R. Part 100 guidelines. 10 C.F.R. § 50.44(g); Greene Hydrogen Testimony at 6.

229. A five percent cladding-steam reaction would not in itself result in the hydrogen concentration reaching the lower flammability limit. Tr. 2178 (Dieterich). However, the amount of hydrogen produced from that reaction, plus the hydrogen generated by the long-term sources such as radiolytic decomposition of water, would result in a slow buildup in hydrogen concentration in the containment building. Tr. 2178, 2179 (Dieterich). For Rancho Seco, the lower flammability limit of four volume percent is reached several weeks after the start of the design basis accident. Greene Hydrogen Testimony at 4; Tr. 2846 (Greene); Tr. 2173-2174, 2337 (Dieterich). However, Licensee intends to activate the hydrogen purge system when the hydrogen concentration reaches 3.5 volume percent, that is, 770 hours after the start of the design basis accident. Dieterich Testimony at 21, 23; CEC Ex. 31; Tr. 2337 (Dieterich).

(continued)
filter and release all the hydrogen resulting from five percent of the cladding reacting with steam. Tr. 2157 (Dieterich).

230. If the hydrogen purge system at Rancho Seco were activated when the hydrogen concentration reached the 3.5 volume percent level after the design basis accident, the radiation doses at the exclusion area boundary (calculated in Appendix 14C of the Rancho Seco Final Safety Analysis Report) would be 5.4 rem to the thyroid and 1 rem to the whole body; these doses are well within 10 C.F.R. Part 100 guidelines. Greene Hydrogen Testimony at 6; Licensee's Supplemental Testimony of Robert A. Dieterich in Response to Board Question H-C 20, following Tr. 1988 ("Dieterich Hydrogen Testimony"), at 2, 3; Tr. 2856-2858, 2860 (Greene).

231. Thus, the Rancho Seco hydrogen purge system meets all the applicable NRC standards for combustible gas control systems in light water cooled power reactors, as set forth in 10 C.F.R. § 50.44(g) and 10 C.F.R. Part 100. Greene Hydrogen Testimony at 6; Tr. 2848, 2857-2858 (Greene). Moreover, Licensee has taken the additional precautionary measure (not required by the NRC) of contracting with the Arizona Public Service Company ("APS") to obtain from APS, on a loan basis, its hydrogen recombiners currently in storage at the Palo Verde Nuclear Generating Station. Dieterich Testimony at 21; Tr. 2152 (Dieterich); Tr. 2848 (Greene). The APS recombiners could be delivered to the Rancho Seco site in approximately 24 hours and would therefore be available within sufficient time to assist in reducing the hydrogen concentration in the containment.[129] Id.; Dieterich Testimony at 22.

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129 The APS recombiners would be connected by maintenance personnel
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232. Even though a recombiner would be available for use at Rancho Seco within a short period of time after its need was perceived,[130] early use of a recombiner would be of limited value in the event of an accident, such as that at TMI-2, in which a substantial amount of the fuel cladding combined in a short period of time with steam to produce hydrogen.[131] This is because hydrogen was generated at TMI-2 approximately five hundred times faster than the rate at which recombiners available today can recombine hydrogen.[132] Dieterich Testimony at 22; Greene Hydrogen Testimony at 4, 5; Tr. 2844, 2855, 2886 (Greene); Tr. 2352-2353, 2363 (Dieterich). Therefore, the answer to the first part of Board Question H-C 20 is that no PWR, including Rancho Seco, possesses a hydrogen control system

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to 1-inch reactor building penetrations. Since the recombiners use 4-inch lines, pipe reducers would be used to implement the connection. Tr. 2153-2155 (Dieterich); CEC Ex. 31. Connection of the recombiners to the containment penetrations would be a simple procedure. Tr. 2848 (Greene).

130 Having a recombiner installed at an early point in an accident would make faster reduction of the concentration of hydrogen in the containment possible, but would not necessarily prevent the hydrogen level from reaching the point where purging would be necessary. Tr. 2853-2854, 2909-2912 (Greene).

131 At TMI-2, it is calculated that on the order of 30 percent of the zircalloy cladding reacted with steam to produce hydrogen. Tr. 2885 (Greene).

132 For typical recombining processing of about 50 cubic feet of gas per minute, it would take 27 days to process all of the containment atmosphere. Tr. 2844 (Greene). At TMI-2, the entire zircalloy cladding-steam reaction is believed to have taken place in a two-hour period, 1.5 to 3.5 hours after the start of the accident. Greene Hydrogen Testimony at 4.

which, by recombination or otherwise, can accommodate safely on a short-term basis the massive hydrogen buildup experienced at TMI-2.[133]

233. Generation of a greater amount of hydrogen than that produced in the design basis accident, however, is highly improbable, particularly in view of the steps taken by the NRC and Licensee to prevent a feedwater transient from developing into an accident in which hydrogen would be produced. Tr. 2364 (Dieterich); Greene Hydrogen Testimony at 5. Moreover, even the amount of hydrogen generated in an accident of the severity of TMI-2 would not result in a dangerous challenge to the integrity of the Rancho Seco containment.[134] Thus, the inability to dispose of the accumulated hydrogen early in the accident has no adverse consequences, and continued utilization of Rancho Seco's purging system presents no safety problem. Tr. 2909 (Greene).

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133 One could theoretically increase the size of the purging system so that the containment atmosphere could be filtered and vented at a faster rate. Such a modification, however, would be of little use in a severe accident of the TMI-2 type because venting could not take place until the highly radioactive, short-lived fission products had decayed. Thus the venting process could not be initiated until many days into the accident. Tr. 2157-2159 (Dieterich); Tr. 2885, 2886 (Greene).

134 The hydrogen combustion at TMI produced a pressure spike of approximately 28 psig. Dieterich Testimony at 23; Tr. 2885 (Greene). The Rancho Seco containment has a design pressure of 59 psig and can probably withstand twice the design pressure. Tr. 2174, 2175 (Dieterich); Tr. 2885 (Greene). Licensee witness Dieterich testified that if all the zircalloy cladding available in the core were to react with steam to produce hydrogen, and if all that hydrogen burned without exploding, the containment would be able to accommodate the resulting overpressure. Dieterich Testimony at 23; Tr. 2175-2178 (Dieterich).

234. The second part of Board Question H-C 20 asked whether Rancho Seco's hydrogen purge system provides proper radiological protection of the surroundings in the event of purging. As discussed above, hydrogen purging will only take place when the short-lived fission products have decayed sufficiently for radioactive releases after purging to be within 10 C.F.R. Part 100 guidelines. Dieterich Hydrogen Testimony at 3. In fact, an analysis in the Rancho Seco Final Safety Analysis Report shows that the radiation doses at the site boundary, in the event of purging after an accident in which five percent of the fuel cladding reacts to form hydrogen, would be well within those guidelines. Id. Therefore, the answer to the second part of Board Question H-C 20 is in the affirmative.

N. Venting Into Containment

Issue CEC 5-1: Whether those systems identified as contributing to the releases of radioactivity during the TMI accident, which are outside containment, should be changed to vent into the containment building?

235. The last two issues to be considered -- CEC Issues 5-1 and 5-2 -- pertain to modifications which CEC asks be considered for possible implementation at Rancho Seco above and beyond the short-term and long-term modifications set forth by the Commission in its May 7, 1979 Order. In admitting these two issues into the proceeding the Board made it clear that CEC

had the burden of coming forward "with evidence that these additional measures will be required." Order Ruling on Scope and Contentions, dated October 5, 1979, at 14; see also, Order Ruling on CEC's Motion of October 24, 1979 Relative to Burden and Going Forward, dated December 17, 1979, at 2. Thus, consideration of these issues requires that the Board determine whether CEC has satisfied its evidentiary burden. For the reasons stated below, the Board finds that CEC has not met this burden.

236. The only evidence offered by CEC in support of Issue CEC 5-1 was the testimony of Bruce J. Mann. See Prepared Direct Testimony of Bruce J. Mann Concerning Release of Radioactivity from Containment (CEC Issue 5-1), following Tr. 2926 ("Mann Testimony"); and see, Tr. 2924-2945 (Mann). However, nowhere in his written testimony or in his oral testimony at the hearing did Mr. Mann express the opinion that systems outside the Rancho Seco containment should be changed to vent into the containment building. Thus, while suggesting that "venting back" might be worthy of study as a way to improve the safe response of the Rancho Seco plant to a severe accident, Mr. Mann admitted that he had not studied the "venting back" concept, and did not know what it would take to implement it or whether it might create problems instead of helping to solve them. Tr. 2932-2936 (Mann). Indeed, Mr. Mann repeatedly disclaimed being a proponent of requiring implementation of this system at Rancho Seco. The following excerpt from Mr. Mann's testimony at the hearing is representative:

> ...The point which I would make in this regard is that I would view such a [venting] capability as a discretionary matter...and my position on this matter of venting would be -- even though I am not prepared to suggest that it be required of anyone at this time -- but even if such an analysis were performed which found it to be potentially beneficial, I would recommend only that the capability be available, and whether or not systems would be vented back into the containment would require the deliberation of the senior staff at the facility....

Tr. 2941 (Mann).

237. Thus, Mr. Mann was not prepared to recommend implementation of the "venting back" concept at Rancho Seco· in fact, no other testimony or documentary evidence in this record support the concept. Instead, several witnesses testified that venting back into containment was unnecessary. Dieterich Testimony at 19; NRC Staff Testimony of Jack N. Donohew on Changing the Systems Outside Containment to Vent Into Containment Building (CEC Issue 5-1), following Tr. 3168 ("Donohew Testimony"), at 9; NRC Staff Testimony of James Wing on Changing the Systems Outside Containment to Vent Into Containment Building (CEC Issue 5-1) following Tr. 2740 ("Wing Testimony"), at 9; Tr. 2129, 2136 (Dieterich); Tr. 2762-2764 (Wing); Tr. 3173, 3174 (Donohew). Given the state of the record on this issue, the Board finds that there is no support for having systems located outside containment at Rancho Seco changed to vent into the containment building.

238. In addition to the "venting back" concept, testimony was presented at the hearing on the operation of the containment isolation system at Rancho Seco and the steps being taken by Licensee to improve the degree of isolation provided by it. Because the failure to maintain adequate isolation during the TMI-2 accident was one of the safety concerns raised by that accident, the Board will review the evidence offered regarding the Rancho Seco isolation system and its similarities to and differences from the system at TMI-2.

239. The containment isolation system is designed to minimize the leakage of radioactive materials out of the containment building in the event of a LOCA or similar accident. CEC Ex. 29 at 5.2-32. The system is fairly simple. It consists of a closed piping network and a number of valves which penetrate containment and which close or open, as the case may be, upon an isolation actuation signal. Tr. 2130 (Dieterich). The valves are redundant, for there are at least two valves (often of different types) on each penetration, although only one valve is needed to provide isolation. Tr. 2137, 2138 (Dieterich); Tr. 2766, 2767 (Greene); CEC Ex. 29 at 5.2-32, 5.2-49 to 5.2-52. This design feature insures that the isolation system is single failure proof; i.e., the system can withstand any single failure of an active component and still maintain containment isolation.[135] Tr. 2149 (Dieterich); CEC Ex. 29 at 5.2-32.

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135 Isolation valves are tested periodically to verify that the isolation system operates as intended. Tr. 2766 (Greene).

240. Initiation of containment isolation at Rancho Seco is provided by a safety features actuation signal generated by "diverse" parameters, i.e., by either of the following two conditions: (a) low reactor coolant system pressure (below 1600 psig); or, (b) high reactor building pressure (above 4 psig). Tr. 2144 (Dieterich); Tr. 2749 (Greene); Tr. 2928, 2929 (Mann); Dieterich Testimony at 18; Mann Testimony at 12; Donohew Testimony at 4. Initiation of containment isolation at Rancho Seco is significantly different from that at TMI-2, for the containment at TMI-2 isolated only on high reactor building pressure. Tr. 2927, 2929, 2933 (Mann). As a result, isolation did not occur at TMI-2 until 4 hours and 20 minutes into the accident, while for the same transient isolation would have been achieved at Rancho Seco on low coolant system pressure very soon (probably about two minutes) after transient initiation. Tr. 3178 (Donohew); Mann Testimony at 12.

241. Upon a containment isolation initiation signal, the lines connecting "non-essential" systems across the containment walls will be isolated by means of the isolation valves; "essential" systems will not be isolated. Donohew Testimony at 3, 4; Tr. 3179 (Donohew). Licensee defines an essential system as one either needed immediately after a SFAS or one whose continued operation will not cause accident recovery problems and whose continued operation may aid in accident recovery. A non-essential system is one falling into

neither of those two categories.[136] Tr. 3207, 3208 (Donohew); Tr. 2151 (Dieterich); Donohew Testimony at 3. By isolating the non-essential systems after an accident, it is possible to minimize the release of radioactive matter to areas outside of containment. Tr. 2753 (Wing).

242. An example of a system that would be isolated at Rancho Seco because of its non-essential nature is the letdown portion of the makeup and purification system.[137] Tr. 2143, 2144, 2336 (Dieterich); Tr. 2774, 2777 (Wing); Tr. 3171 (Donohew). The letdown system was the most significant pathway for radiation to escape out of containment at TMI-2. Tr. 3172 (Donohew); Tr. 2937 (Mann).[138]

243. The containment isolation valves do not reopen automatically if the containment isolation signal clears;

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136 Licensee performed in late 1979 a detailed review of the systems associated with each containment penetration for the purpose of verifying that "essential" and "non-essential" categories were properly defined and that all non-essential systems would be isolated upon an isolation signal. Tr. 2141, 2142 (Dieterich). This effort was undertaken as part of Licensee's response to the TMI-2 "Short-Term Lessons Learned" recommendations in NUREG-0578. Licensee's conclusions were reviewed by the Staff and found acceptable. Donohew Testimony at 5; Tr. 3184 (Donohew).

137 "Let down" of the reactor coolant system is constantly taking place so that reactor coolant water may be processed and cleaned. Because of let down and minor leakage out of the primary system, there is always need to supply makeup water to the primary system. Makeup is provided by one of the three HPI pumps. Tr. 2335, 2336 (Dieterich). The HPI portion of the makeup and purification system is an essential system and is not isolated. Tr. 2777 (Wing).

138 At TMI-2, the operators chose to operate the letdown system for a considerable period of time after isolation had been achieved. Tr. 2928, 2937 (Mann); Tr. 3172 (Donohew).

instead, manual action is required to open the valves. Donohew Testimony at 4. The operator can override the isolation signal on an individual penetration basis. Tr. 2142 (Dieterich). Manual opening of the valves after containment isolation requires a two-step process -- placing the SFAS system in the "manual" mode and pushing open the button for the valve in question. Donohew Testimony at 4, 8; Wing Testimony at 7. There are procedures requiring the operator to consult the Technical Specifications prior to defeating isolation of any system.[139] Wing Testimony at 7. Retaining the operators' capability, under carefully specified conditions, to override the containment isolation and activate non-essential systems is desirable for safe plant operation. Tr. 3169-3171 (Donohew).

244. An additional way to protect against the release of radioactive materials to the environment after an accident is to minimize or eliminate leakage from those systems outside containment likely to contain radioactive materials. Donohew Testimony at 3, 4. Licensee has identified those systems at Rancho Seco that may contain radioactive fluids in the event of a serious transient or accident, and has implemented a leak reduction program for these systems to reduce their present leakage rate.[140] Donohew Testimony at 5. The

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139 Under certain conditions, operators may be required to open a non-essential penetration after isolation has been achieved. There are, however, procedures establishing the conditions under which this may be done. Tr. 2142 (Dieterich).

140 Licensee already had in place a leak reduction program, (footnote continued next page)

Staff has reviewed Licensee's leak reduction program and determined it to be acceptable. Tr. 3186 (Donohew). Licensee has made permanent its new leak reduction program in order to keep future leakage from those systems to levels which are as low as reasonably achievable.[141] Donohew Testimony at 6.

245. Any contaminated water discharged out of the reactor building or leaking out of systems outside containment will be collected in the radwaste system in the auxiliary building. The radwaste system is designed to contain radioactive materials, although not necessarily to accommodate the amount of fluid discharged in a LOCA if containment isolation fails. Donohew Testimony at 7. However, Rancho Seco has an above average amount (300,000 gallons) of tankage at the site available to contain radioactive waste; so the radwaste system at Rancho [illegible]uld be able to accommodate a large discharge of radioactive waste, perhaps comparable to the amounts produced as a result of the accident at TMI-2. Tr. 3189-3190, 3205-3206 (Donohew). This large storage capacity, plus the provisions for prompt and continued isolation of non-essential

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which was upgraded based on the recommendations of NUREG-0578. Tr. 3185, 3186 (Donohew).

141 As a result of a review of the Rancho Seco design conducted by Licensee to identify potential radiation release paths from systems outside containment, Licensee has identified plant modifications that will reduce the possibility of such releases. One is to alter the floor slope of the auxiliary building so that fluids on the floor do not move toward doorways and other entries. Tr. 3203 (Donohew); Donohew Testimony at 6.

systems after SFAS, ensure that the radwaste system will be able to accommodate the radioactive waste generated in a severe accident. Id.; Donohew Testimony at 7.

246. The record shows that Rancho Seco's containment isolation system will provide isolation early enough in a transient to minimize the possibility of radioactive releases outside of containment and to ensure that the capacity of the radwaste system is not exceeded. Tr. 2144, 2145 (Dieterich). The capability to provide a high degree of isolation at Rancho Seco has been enhanced by a leak reduction program, identification of non-essential penetrations, and plant modifications. It is the Board's view that placing reliance on a dependable isolation system,[142] and improving the performance of the system where possible, is a better way to enhance plant safety than implementing a major containment modification program (such as "venting back") whose need has not been demonstrated and whose usefulness and potential shortcomings are yet to be determined.[143] Tr. 2136 (Dieterich).

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142 No shortcomings of the Rancho Seco containment isolation system were identified by witnesses at the hearing. CEC witness Mann testified that no release paths out of containment at Rancho Seco have been identified; only potential release paths, in light of the TMI-2 experience, have been suggested. Tr. 2938 (Mann). Those paths would mainly be created if an operator decided to operate an isolated system and thus deliberately defeated containment isolation. Tr. 2939 (Mann). As noted above, operators are allowed to do this only under certain specified conditions.

143 Implementation of a "venting-back" system would require making a number of new penetrations into the containment, and the addition of valves, pumps and pipes subject to leakage. (footnote continued next page)

O. 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?

247. This issue, like CEC Issue 5-1, is one for which CEC has the burden of showing that the proposed modification is required at Rancho Seco to protect public health and safety. See paragraph 235, supra. The Board views this burden as one of showing that a significant risk exists to public health and safety which can be effectively reduced by such a modification without otherwise compromising plant safety.

248. Licensee argued, however, that this issue, unlike CEC Issue 5-1, was not appropriate for consideration at the hearing. Licensee moved for summary disposition of this issue on January 24, 1980, asserting, among other reasons, that the controlled filtered venting system proposed by CEC would be intended to mitigate accidents more severe than the design basis accident for the Rancho Seco containment, and that the proposal constituted an impermissible challenge to a Commission regulation -- the General Design Criteria for Nuclear Power

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Tr. 2129, 2134-2136 (Dieterich). Thus, a "venting back" system could result in a reduction rather than an increase in reactor safety. Tr. 3175, 3176 (Donohew). In view of these unresolved questions, there is no reason to conclude that a "venting-back" system should be implemented at Rancho Seco. Tr. 3173, 3174 (Donohew).

Plants set forth in 10 C.F.R. Part 50, Appendix A (particularly criteria 16 and 50) -- which criteria the Rancho Seco plant undeniably meets.[144] Licensee argued that any modification of the General Design Criteria could only be made by the Commission and would be beyond the power of this Board to direct. The NRC Staff supported Licensee's motion, and CEC opposed it. At the prehearing conference of February 6, 1980, the Board denied Licensee's motion and ruled that consideration of CEC's proposed modification would not constitute a challenge to Commission regulations and was within the scope of the hearing. Tr. 100. On February 19, 1980, Licensee filed a motion for reconsideration of the Board's ruling on this issue, in which it once again argued that CEC Issue 5-2 represents a challenge to the Commission's General Design Criteria. At the hearing, the Board denied Licensee's motion for reconsideration. Tr. 356, 357. While the Board disagreed with Licensee's position, it was and is mindful that the proposed modification constitutes a substantial departure from existing design criteria and philosophy, and therefore requires careful scrutiny.

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144 See, e.g., NRC Staff Testimony of Thomas A. Greene on Containment Overpressurization Protection (CEC Issue 5-2), following Tr. 2783 ("Greene Containment Testimony"), at 7; Licensee's Testimony of Robert A. Dieterich in Response to California Energy Commission Issue 5-2, following Tr. 1988 ("Dieterich Containment Testimony"), at 2, 3.

249. The subject of CEC Issue 5-2 is the containment building at Rancho Seco. Before considering the merits of the overpressurization protection system identified in this issue, the Board will examine the circumstances under which the containment building at Rancho Seco might fail from overpressure, and the forms of failure which might take place. The containment building is a massive structure with a net free internal volume of approximately two million cubic feet. Greene Containment Testimony at 3; Tr. 2617 (Nix). It is a reinforced concrete structure, inside of which is a steel liner that makes the structure leak tight. It has tendons running vertically, horizontally and over the dome for additional strength. Tr. 2213 (Dieterich); Greene Containment Testimony at 3. There are approximately 70 penetrations into the containment. Each penetration contains a line going across the containment boundary, and each line is provided with a redundant set of valves to ensure that the opening can be sealed tightly on demand. Each penetration is sealed to the line it carries with weld material. Those seals are designed to withstand temperatures of at least 286°F. Tr. 2214 (Dieterich).

250. The containment building is designed to hold radioactive materials that may be released during operation of the reactor or in the course of an accident. Therefore, the building is to remain leak tight to prevent such materials from being released to the environment. Tr. 2230 (Dieterich). See

General Design Criteria 16 and 50, Appendix A to 10 C.F.R. Part 50.

251. There are many ways in which the integrity of the containment theoretically can be breached during an accident. For instance, Table 7 in the Prepared Direct Testimony of Daniel Nix Concerning Controlled Filtered Venting (CEC Issue 5-2), following Tr. 2403 ("Nix Testimony") at 15, lists nine categories of PWR accidents that can lead to radioactive releases outside containment.[145] While the nine "release categories" described in that table are intended to represent dominant release sequences for PWRs,[146] the list is not necessarily exhaustive. Tr. 2495 (Nix). All but the last accident sequence included in the table result in "Class 9" accidents, i.e., accidents more severe than the design basis accident for Rancho Seco.[147] Tr. 2494, 2495 (Nix); Nix Testimony at 4, Table 1.

252. Two of the nine release categories in Table 7 -- PWR-2 and PWR-3 -- include failure of the containment from

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145 Table 7 was taken from the "Reactor Safety Study," WASH-1400 (1975), a report assessing the accident risks in U.S. commercial nuclear power plants. The table summarizes 130,000 accident sequences analyzed in the Reactor Safety Study. Tr. 2493 (Nix).

146 The nine PWR release categories set forth in Table 7 of the Nix Testimony are often denoted as "PWR-1", "PWR-2" and so on. That shorthand notation will be used here.

147 Not all "Class 9" accidents, however, involve a melting of the reactor core. Thus, the popular conception that a Class 9 accident is a core melt is erroneous. Tr. 2494, 2495 (Nix).

overpressurization as the mechanism for radioactive releases to the environment.[148] The systems suggested by CEC witness Nix and discussed at the hearing are intended to provide protection only against these two release categories. Tr. 2495 (Nix).

253. In actuality, however, substantial protection against such a failure already exists. The design of the Rancho Seco facility provides two forms of protection against overpressurization. One is an overpressurization protection system consisting of the containment building spray system and the containment building emergency cooling system. Greene Containment Testimony at 2. The containment building spray system features two separate trains of equal capacity which spray water and sodium hydroxide to remove aerosol fission products released to the containment atmosphere. Id. at 2, 5. The containment building emergency cooling system consists of four fan-cooler units and four emergency upper dome circulators. These two systems remove energy from the containment atmosphere following an accident and, if working properly, will prevent the containment from becoming overpressurized.[149] Id. at 2; Tr. 2223-2224, 2264-2265 (Dieterich).

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148 In the PWR-1 sequence, the containment is ruptured by a missile generated by a steam explosion. In the PWR-4, PWR-5 and PWR-8 sequences, there is a failure of containment isolation. In the PWR-6 and PWR-7 sequences, the core melts through the containment building's foundation. Nix Testimony at Table 7.

149 Both the PWR-2 and PWR-3 release categories include failure of the containment spray and heat removal systems as part of the scenario leading to overpressurization. Nix Testimony at 15 and Table 7. If these systems only become operative after the containment has reached high pressure and temperature conditions, they may be unable to control the transient completely; however, (footnote continued next page)

254. The second and principal protection against overpressurization is the design of the containment building itself. The Rancho Seco containment building is designed to withstand a "design basis accident" consisting of the pressure loadings resulting from the double-ended rupture of the largest pipe in the primary system. Dieterich Containment Testimony at 3. See also, General Design Criterion 50, Appendix A to 10 C.F.R. Part 50. The maximum calculated containment pressure produced in the design basis accident is 52 psig,[150] and the Rancho Seco containment design pressure is 59 psig. Greene Containment Testimony at 3; Dieterich Containment Testimony at 3; Tr. 2215 (Dieterich); Tr. 2806 (Greene). This design pressure was obtained by adding to the design basis accident pressure a 12% safety margin and by requiring that the building be able to withstand that internal pressure in the presence of wind and earthquake loadings. Tr. 2215 (Dieterich).

255. Because of the number of very conservative assumptions and safety margins included in the design, the

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to the extent they are operational, they will continue to mitigate, if not control, the pressure and temperature rises produced in an accident such as a core melt. Tr. 2804-2806 (Greene); Tr. 2223, 2224 (Dieterich).

150 This pressure includes the effects of stored energy in the reactor coolant system, decay heat, and energy from other sources such as the secondary system and metal-water reactions. Greene Containment Testimony at 7. Because of the containment building's large volume, an enormous amount of energy has to be released to the containment atmosphere over a short period of time in order for such a pressure increase to be experienced. Id. at 6.

Rancho Seco containment building would be able to withstand pressures well in excess of 59 psig.[151] Greene Containment Testimony at 7; Dieterich Containment Testimony at 3; NRC Staff Testimony of Dr. James F. Meyer on Containment Overpressurization Protection (CEC Issue 5-2), following Tr. 2786 ("Meyer Testimony"), at 4; Tr. 2215 (Dieterich); Tr. 2830-2832 (Meyer). In fact, two analyses performed by the Structural Branch of the NRC Staff and its consultants showed that a large PWR containment such as Rancho Seco's would withstand, without failure, pressures twice as large as the design pressure, i.e., approximately 120 psig.[152] Tr. 2809, 2868-2871 (Greene); Tr. 2865-2866, 2900-2901 (Meyer). And a more recent study by Sandia Laboratories of large PWR containments has produced a family of containment failure pressures, based on particular loading progressions in the containment, which range from 90 psig to 150 psig. Tr. 2866-2867, 2900-2901 (Meyer).

256. All witnesses who addressed the subject stated that there is a great degree of uncertainty on what the actual failure pressure of the Rancho Seco containment would be.

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151 Prior to startup, the Rancho Seco containment was pressurized with air to 115% of its design pressure, i.e., 69 psig. This pressure was maintained for over a day without detrimental consequences. Tr. 2216 (Dieterich); Tr. 2809 (Greene).

152 The more realistic, higher failure pressure is largely obtained by relaxing some of the very stringent and conservative assumptions associated with the design basis accident, such as a prohibition against going beyond the yield stress point in reinforcing rods and similar materials. Tr. 2903, 2904 (Meyer).

There was widespread agreement, however, supported by test data, that the containment would not fail for pressures under 70 psig. Tr. 2688 (Nix); Tr. 2830 (Meyer); Tr. 2215 (Dieterich). As pressure increases, there is an increasing probability that the containment will fail; that probability remains quite low until about 100 psig and then, depending on the containment loading history, it increases dramatically. Tr. 2810, 2811 (Greene); Tr. 2828 (Meyer). No witness was able to predict what the actual containment failure pressure would be. Tr. 2358, 2359 (Dieterich); Tr. 2691, 2707 (Nix); Tr. 2811 (Greene).

257. It appears, moreover, that there is no single failure pressure -- for the containment might fail at different pressures depending on the wind loading and earthquake loading conditions, the loading history of the containment, and the accident sequence. Tr. 2358, 2359 (Dieterich); Tr. 2371, 2872 (Meyer). As Staff witness Greene observed, it is possible to calculate the pressure that a containment can withstand; the converse, i.e., predicting the pressure at which it will fail, is a very difficult, if not impossible, task. Tr. 2871 (Greene).

258. Another source of uncertainty is the form that the containment failure would take. It is possible that, at least for some overpressurization sequences, the containment would not fail catastrophically, but would develop cracks in the concrete that would find their way to the containment

outside surface, open long enough to relieve containment pressure, and then seal back. Tr. 2361 (Dieterich); Tr. 2691, 2706-2707 (Nix); Tr. 2867 (Meyer); Tr. 2872, 2873 (Greene). The releases from such a failure mode would be significantly lower than those generated by a large-scale catastrophic failure (i.e., one resulting in large permanent openings of the containment structure). Tr. 2867 (Meyer).

259. As will be seen, the pressure and failure modes assumed are very important in determining whether an over-pressurization protection system is desirable or even feasible. Another factor that needs to be considered prior to passing judgment on the desirability of such a system is its risk reduction potential. For, if the system will provide only a small degree of risk reduction, its usefulness may be outweighed by its cost or by negative features that may be associated with its implementation. The potential degree of risk reduction afforded by an overpressurization protection system depends on two factors: the extent of risk associated with containment overpressure, and the effectiveness of the mitigation system in reducing that risk. These factors will be considered separately, followed by an examination of costs and other features which may diminish or outweigh the potential risk reduction benefits of the overpressurization protection system suggested in CEC Issue 5-2.

260. Risk is computed by multiplying the probability of an event taking place by the event's consequences. Tr.

2468, 2471 (Nix). Two kinds of risk computation for PWR accidents are possible: one is the absolute risk posed by a given accident or class of accidents; another, the relative risk, is a comparative ranking of the various accidents in terms of their contribution to total risk. See Tr. 2475-2477 (Nix). If one assumes that the same or similar errors are incurred in computing the absolute risk for the various accident sequences, then the relative risk may be a more reliable indicator to use. Tr. 2478 (Nix). However, there appears to be insufficient knowledge as to the magnitude and direction of the errors made in computing the absolute risk associated with each PWR accident sequence. Therefore, it is impossible to determine whether, for PWR accident analysis, a relative risk method, such as that utilized by CEC witness Nix, gives more reliable results than an absolute risk computation. See Tr. 2478, 2479 (Nix); Nix Testimony at 13, 14, and Table 6.

261. The results of the Reactor Safety Study indicate that, of the PWR release categories listed in Table 7 of the Nix Testimony, the first three -- PWR-1, PWR-2 and PWR-3 -- are dominant contributors to the risk associated with radioactive releases. Nix Testimony at 14 and Table 6; SMUD Ex. 11 at 8-3, 8-11 and 8-14. Understanding of the mechanisms leading to the PWR-1 type of release (rupture of containment by steam explosion-generated missiles), however, is so tenuous that the probability and effect of that class of accident cannot, at this point, be predicted with confidence.[153] Tr. 2487 (Nix); SMUD Ex. 11 at 8-4.

153 Thus, the relative risks presented in Table 6 of the Nix
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262. The current state of knowledge of the relative importance of these and other release categories may improve in the near future. The Interim Reliability Evaluation Program ("IREP") being conducted by the NRC Staff includes a probabilistic analysis of PWRs along the lines of the Reactor Safety Study, and will lead to identification of the dominant failure sequences for various types of PWRs.[154] Tr. 2840 (Meyer); Staff Ex. 4 at 6-1 to 6-4. Until this study is completed, however, there can be no assurance that overpressurization sequences are principal contributors to the overall risk posed by PWR accidents.

263. Assuming, nevertheless, that the PWR-2 and PWR-3 release categories are important contributors to that risk, one must determine the significance of the risk. To answer that question, two areas must be explored: the probability of occurrence of a PWR-2 or PWR-3 sequence, and the consequences of such an accident. With respect to the first of these areas, SMUD Exhibit 11 at 3-4, Table 3-2, gives the 50% probability value for release categories PWR-1 through PWR-7.

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Testimony are not reliable to the extent that they seek to compare the PWR-1 risk to that posed by PWR-2 and PWR-3 sequences.

154 This effort will be completed within the next two or three years, together with the proposed Commission rulemaking on core melt accidents. Tr. 2840 (Meyer). It might prove fruitless to attempt minimizing at this time the risk posed by containment overpressurization if it were determined in two or three years that the overall risk is dominated by another failure sequence such as steam explosions. Compare Tr. 2487-2490 (Nix).

The Table shows that, according to the Reactor Safety Study, the PWR-2 category of accidents has a combined probability of occurrence[155] of 8×10^{-6} accidents per reactor year, or one accident in 125,000 years of reactor operation. SMUD Ex. 11 at 3-4; Tr. 2479-2481 (Nix). For the PWR-3 category, the combined 50% probability of occurrence is 4×10^{-6} accidents per reactor year, or one accident in 250,000 years of reactor operation. Id. CEC witness Nix testified that the absolute probabilities of accident occurrence given in the Reactor Safety Study are based on imperfect knowledge and subject to a large uncertainty. Tr. 2476, 2477 (Nix). Nevertheless, it is not disputed that, as the Reactor Safety Study probabilities suggest, these release categories describe very improbable accidents. Tr. 2461, 2502-2503 (Nix); Tr. 2298, 2299 (Dieterich); SMUD Ex. 18 at II-2, II-11, II-12 and V-5.

264. The other risk element to consider are the potential consequences, in terms of health effects and economic impacts, of one of these very improbable overpressurization accidents. SMUD Exhibits 11 and 18 include the results of a study by a consultant to CEC of the consequences of an extreme reactor accident in the PWR-2 release category. SMUD Ex. 11 at 7-2; Tr. 2462, 2524, 2526-2527 (Nix). The study sought to compute, for four California nuclear power plant locations,

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155 Each release category represents a class of accident sequences having similar characteristics. For instance, the PWR-2 category includes 15 general accident sequences. Tr. 2504 (Nix).

including Rancho Seco, the health effects and property damage associated with such an accident. Tr. 2531, 2532 (Nix). Specifically, the Rancho Seco plant was identified as Site "A" in the results shown in Tables 15 through 33 (pages V-29 through V-51) of SMUD Exhibit 18. Tr. 2566, 2567 (Nix). Therefore, the results for Site "A" given in SMUD Exhibit 18 would be the values directly applicable to Rancho Seco.[156] Id.

265. It is important, however, to understand the assumptions utilized in arriving at the results shown in Tables 15 and 19 of SMUD Exhibit 18. The study from which these results were obtained postulated several extreme core-melt accident sequences, each assumed to lead to loss of containment integrity prior to containment melt-through, so that the amounts of radionuclides available for release to the atmosphere and to the groundwater are maximized.[157] SMUD Ex. 18 at II-11, II-12 and V-5. The fractions of the radionuclides inventory released out of containment are shown in SMUD Exhibit 18, Table 9, at V-10.[158] See also, Nix Testimony, Table 4, at

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156 Thus, the health effects and economic consequences shown as ranges of values on Tables 5 (health effects) and 3 (economic consequences) of the Nix Testimony can be disregarded, for the values within those ranges applicable to Rancho Seco are readily available in, respectively, Tables 15 and 19 at pages V-29 and V-33 of SMUD Exhibit 18.

157 The study considered that only insoluble molecules and the core melt itself would not be available for release. SMUD Ex. 18 at V-5.

158 The fractions shown in Table 9 can be converted to percentages by multiplying times 100. Thus, for the first entry on the table (Xe and Kr), .8 or 80% of the total inventory of these (footnote continued next page)

12. It was assumed that those areas in which the population would receive a dose of 25 rems or more over a period of 30 years would be evacuated, and that all areas on which the total dose over 30 years would be 25 rems or more would remain interdicted, i.e., unavailable for human habitation or other use.[159] SMUD Ex. 18 at V-2, V-52. It was also assumed that evacuation would not take place for a period of 24 hours from the time of containment failure.[160] SMUD Ex. 18 at V-2, V-3 and V-36. Air concentrations of radioactivity at a given point were assumed to reach their average equilibrium value

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radionuclides is assumed released to the environment. Tr. 2659 (Nix).

159 The study assumed that no effort would be made to recover a contaminated area for rehabilitation; only the natural process of weathering and radioactive decay, without human intervention, was assumed to operate to reduce the radiation rate to non-hazardous levels. SMUD Ex. 18 at V-26 to V-28. A decontamination factor of 2 (where a factor of 1 is no decontamination), however, was expected to be "readily attainable over a short period of time with a minimal decontamination effort". If this factor had been used instead of the factor of 1 used in the study, it would have cut in half the interdiction period, and similarly cut in half the total economic consequences presented in Table 19. SMUD Ex. 18 at V-36, V-53.

160 The 24-hour evacuation time was categorized in the study as "an improbable extreme that was included to compare the consequences if emergency evacuation was not undertaken." SMUD Ex. 18 at V-3. The health effect impacts of using different evacuation times are shown on Tables 21 through 29 of SMUD Exhibit 18. For Rancho Seco, those tables indicate that by using a 24-hour evacuation period (Case 4), as the study did, instead of a more reasonable 2 to 3-hour evacuation estimate (Case 2), one increases early deaths from 44 to 66, early illnesses from 50 to 807, late cancer deaths from 0 to 31, thyroid cancers from 6 to 2410, thyroid nodules from 0 to 3169, genetic disorders from 0 to 21, spontaneous abortions from 0 to 7, and temporary sterility cases from 2 to 3057.

instantaneously upon arrival of the radioactive plume, and ground deposition was assumed to reach its average deposition rate instantaneously upon plume arrival. SMUD Ex. 18 at V-13.

266. People downwind from the release point would receive an exposure dose which would depend on a number of factors, such as: (1) the radionuclide concentration in the air; (2) a shielding factor depending on whether or not the people had evacuated when the airborne or ground-deposited radioactivity arrived;[161] and (3) the exposure period (determined by the activity arrival and passage times, and the time for people to evacuate). SMUD Ex. 18 at V-14. Representative population distributions were used to distances slightly in excess of 100 miles from the site; at more distant locations, the average California population density of 130 persons per square mile was used.[162] SMUD Ex. 18 at V-15; SMUD Ex. 11 at 7-5; Tr. 2548, 2549 (Nix).

267. The model used in the study assumed that the radioactive effluent is distributed evenly across a 22.5 degree sector downwind from the release point. SMUD Ex. 18 at V-11. Since the wind direction and stability were determined to be

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161 For people who did not evacuate or were in the process of evacuating when the airborne activity arrived, shielding factors of 0.75 to 1 (essentially no shielding) were used. A shielding factor of 0.5 was used for all external exposures to ground-deposited radioactivity. SMUD Ex. 18 at V-14, V-15.

162 This simplification assumed that all radioactivity at long distances from the site is deposited in California and none is deposited in a less densely populated state (e.g., Nevada) or in the ocean. SMUD Ex. 18 at V-37.

key factors affecting the numbers of health effects obtained, a weighted average of six wind speed classes and seven stability categories was used for calculation of accident consequences. SMUD Ex. 18 at V-35; SMUD Ex. 11 at 7-5. In addition, a set of "severe" consequences was computed representing "a most adverse and unlikely set of conditions where the release was assumed to occur when the wind was blowing towards the most populated sector" within 50 miles of the Rancho Seco site. SMUD Ex. 11 at 7-5, 7-9; SMUD Ex. 18 at V-35. The results for the most severe case are the maximum values shown in parentheses in Tables 15 through 20 of SMUD Exhibit 18; the smaller results for the weighted average case are shown, not within parentheses, above the maximum values.[163]

268. In computing the weighted average and maximum health effects, the following ground rules were utilized in the study: (1) a linear dose-response health effect relationship for long-term effects, meaning that low doses to large numbers of people generate the same total consequences as will high doses to small numbers of people (since doses and people are multiplied, the health effects may appear significant because low doses have been administered to large numbers of people at

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163 For example, the early deaths for Rancho Seco using the weighted average method would be 32; the maximum value, corresponding to the most severe case, would be 240. SMUD Ex. 18 at V-29, Table 15. The economic consequences shown in Table 19 follow the same format; the total economic cost for Rancho Seco would be $1.3 billion for the weighted average case and $13 billion for the most severe case. SMUD Ex. 18 at V-33, Table 19.

large distances from the reactor), SMUD Ex. 11 at 7-5, 7-6 and 7-8; SMUD Ex. 18 at V-37, V-38; (2) no threshold dose, meaning that there is no minimum dose required in order to experience long-term health effects, Tr. 2569-2572 (Nix); (3) no dose effectiveness factor for long-term effects, meaning that no account is taken of the fact that a low exposure rate may be experienced over long periods of time, SMUD Ex. 18 at V-17, V-28, and V-35; SMUD Ex. 11 at 7-6;[164] (4) a constant rate of long-term cancer incidence due to earlier exposures, extending through the lifetime of the individual rather than through a shorter period,[165] SMUD Ex. 11 at 7-6; and, (5) an "absolute risk" model, i.e, one that assumes that the number of long-term health effects would be proportional to the number of additional cases found among an irradiated population over that found in an unirradiated population.[166] SMUD Ex. 18 at V-37.

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164 As a comparison, the study also computed long-term health effects utilizing a dose effectiveness factor of 0.2 for people exposed to very low exposure rates. SMUD Ex. 18 at V-17. Predictably, less pronounced health effects were obtained using a dose effectiveness factor of 0.2. Compare SMUD Ex. 18 Table 17 at V-31, with Table 15 at V-29.

165 The Reactor Safety Study had used a 30-year risk period for latent cancers resulting from earlier exposures. Use of a lifetime risk period increased the number of long-term cancers by approximately a factor of 1.6 over what they would have been if the Reactor Safety Study cut-off period been utilized. SMUD Ex. 11 at 7-6.

166 The alternative to an absolute risk model would have been to use a "relative risk" model for which long-term health effects are proportional to the spontaneous cancer rates according to age. SMUD Ex. 18 at V-37.

269. It is safe to conclude that the health effects and economic consequences obtained using the above assumptions are conservatively high, and in fact much higher than the results that would be achieved using alternative assumptions such as shorter evacuation times, a threshold dose, and a dose effectiveness factor.[167] Tr. 2599 (Nix). The Board finds this accumulation of extremely conservative assumptions to be unreasonable. Nevertheless, it appears that even these upper bound estimates of the consequences of an extreme and very unlikely accident at Rancho Seco (32 early deaths, 3900 long term cancer deaths over a 30 to 50 year period, $1.3 billion in economic costs over the same period, and so on) are not out of line with other risks, both man-made and natural, deemed acceptable by society although not necessarily by all individuals.[168] CEC Ex. 11 at 7-10, 7-11; Tr. 2539-2543 (Nix).

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167 SMUD Exhibit 18, Table 31 at V-49, shows a comparison of expected long-term cancer deaths computed using some alternative criteria such as a 25 rem dose threshold and a 0.2 dose effectiveness factor. For Rancho Seco, utilization of either of these alternative criteria would result in at least a factor of five reduction in the number of cancer deaths.

168 It is unnecessary to consider whether the effects computed for the most severe case shown in parentheses in Tables 15 and 19 of SMUD Exhibit 18 are socially acceptable. In order to reach those effects, one would have to postulate the occurrence of a PWR-2 accident sequence simultaneous with the wind blowing in the direction of the most populated sector of the Greater Sacramento area. Because 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; the wind blows from Rancho Seco toward Sacramento only 17% of the time. Tr. 2533-2534, 2538 (Nix).

270. An indication of the risk associated with an overpressurization accident is given by Figures 8-1 and 8-3 of SMUD Exhibit 11, which show the probability per reactor year of operation of equaling or exceeding a given number of latent and acute fatalities for the various release categories. These figures, taken from the Reactor Safety Study, indicate that the probability of one or more long-term deaths resulting from a PWR-2 release category accident is about 1 in 100,000 years of reactor operation, and that the probability of one or more early deaths resulting from such an accident is about 1 in 10 million years of reactor operation. SMUD Ex. 11 at 8-11, 8-14; Tr. 2600-2602 (Nix). While the actual probability figures may be inaccurate for the reasons stated previously, they serve to underscore the low risk associated with overpressurization accidents. In summary, the Board finds that the risk associated with an overpressurization accident is uncertain but small. In failing to establish that this risk is significant at Rancho Seco, CEC has failed to meet its burden on its Issue 5-2.

271. Assuming, however, that it is appropriate to investigate mitigating the risk posed by an overpressurization accident, one must determine whether and to what extent the overpressurization protection system identified in CEC Issue 5-2 would be successful in providing such mitigation, the costs, and any new risks created by its incorporation into an existing facility such as Rancho Seco.

272. The theory behind a controlled, filtered venting system ("CFVS") is that it is preferable to release deliberately to the environment, in a controlled manner, the contents of a containment building that otherwise is going to fail from overpressure. Nix Testimony at 8; Meyer Testimony at 2. The conceptual advantage of a CFVS is that the release takes place through filtering and energy absorbing devices intended to mitigate the health and economic consequences of a catastrophic, uncontrolled release. Id. In order for a CFVS to be effective, therefore, it must: (a) operate if, and only if, a catastrophic containment failure due to overpressurization is inevitable;[169] (b) effectively reduce radionuclide releases to the atmosphere; and (c) avoid interfering with other plant safety systems in their mitigation of the transient. If, indeed, a CFVS could be designed that met these conditions, it would offer a large reduction of the risk associated with overpressurization accidents,[170] and thus could provide a large benefit to the health and safety of the public relative to an uncontrolled, unfiltered release situation.[171] Tr. 2838 (Meyer); Meyer Testimony at 2.

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169 As noted earlier, a CFVS will be useful only in mitigating overpressurization failures of the containment (release categories PWR-2 and PWR-3); it will not mitigate any other type of accident leading to releases from containment. Tr. 2483-2485 (Nix).

170 As just discussed, the risk associated with PWR-2 and PWR-3 release category accidents is very low; such accidents are significant contributors to the risk posed by PWR accients only in the relative sense of perhaps posing a higher risk than other release category accidents. Tr. 2700 (Nix).

171 One of the main benefits of a CFVS would be that it could (footnote continued next page)

273. The conditions for effective CFVS operation are, however, not easily met. The CFVS proposed by CEC and described in its Underground Siting Study (SMUD Ex. 11) is a passive system in which the containment atmosphere is discharged through a number of access points or "ports"[172] which, up to the time of system activation, are sealed by metallic discs designed to rupture at a predetermined pressure.[173] Tr. 2614-2616 (Nix); Nix Testimony at 8, 11. In order to maintain reliability and retain the system's passive nature, which is one of the main advantages perceived by its proponents (<u>see</u>, <u>e.g.</u>, Nix Testimony at 10, 11), the discs would have to be

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provide a substantial benefit in evacuation time. Tr. 2813, 2888-2889 (Meyer). Such benefit would be most valuable for plants located near large population centers, since the vast majority of early fatalities are confined to a range of 10 to 15 miles from the plant. SMUD Ex. 11 at 8-4; Tr. 2591, 2592 (Nix). Since the population density at this 10 to 15-mile distance around Rancho Seco is very low, this benefit of a CFVS would be of limited value. SMUD Ex. 11 at 8-13, Fig. 8-2; Tr. 2409-2410, 2595-2599 (Nix); Tr. 2889 (Meyer).

172 The conceptual CFVS design presented in SMUD Exhibit 11 envisioned 24 ports, each one foot in diameter and containing a vent pipe. Tr. 2616, 2620-2621 (Nix). Under one of the arrangements conceptualized in SMUD Exhibit 11, there would be an annular concrete enclosure surrounding the outside of the containment building and connected to the containment atmosphere by 24 vent pipes, each sealed off with a rupture disc. <u>Id.</u>

173 A number of discs could be placed in series at each access point to increase the reliability of the system against premature failure, <u>i.e.</u>, failure at a lower pressure than desired. Tr. 2615, 2616 (Nix); Nix Testimony at 11. By doing this, however, one increases the probability that one or more of the discs would fail to rupture at the pressure setpoint and defeat the operation of the system. Tr. 2286 (Dieterich). Since similarly designed discs could also be subject to common mode failures, placing them in series might not improve their reliability. Tr. 2383, 2384 (Dieterich).

designed to rupture at or very near one single prescribed pressure.[174] Nix Testimony at 11. Choosing the proper rupture pressure for the discs would be quite important, for if the discs ruptured at a pressure much below the containment building's catastrophic failure pressure[175] there could be an unnecessary release of radioactivity.[176] Dieterich Containment Testimony at 6; Meyer Testimony at 3; Tr. 2232, 2353-2354 (Dieterich). On the other hand, if the rupture pressure setpoint was too high the system would be ineffectual, for the containment would fail before the discs ruptured. Even if the CFVS was activated prior to containment failure, if its rupture setpoint was too high pressure relief might not come fast enough to stop the pressure increase before failure of the

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174 The discs could also be made temperature-sensitive, so that they would rupture when the containment temperature rose to a certain setpoint. The object of this feature would be to relieve the containment when the containment seal integrity was threatened by high temperatures. Tr. 2623 (Nix); Nix Testimony at 11. Since the same disc would be subject to rupture on two separate conditions, however, there would be at least two different malfunction modes for each disc. Tr. 2624 (Nix).

175 As previously discussed, analyses conducted by the Staff suggest that an overpressurization failure of the containment may take the form of self-sealing cracks that close once the internal pressure decreases. Pressure relief by this mechanism would be preferable to venting the entire containment atmosphere through a CFVS. Tr. 2825, 2826 (Meyer).

176 For instance, it is possible that in a LOCA the spray system may be temporarily inoperative, causing a containment overpressurization. In the absence of CFVS activation, actuation of the spray system may occur in time to terminate the transient without serious consequences. Operation of the CFVS under those circumstances would cause unnecessary radioactive releases. Dieterich Containment Testimony at 6.

containment. Meyer Testimony at 3; Tr. 2232-2235, 2353-2354 (Dieterich); Tr. 2872 (Meyer).

274. The preceding discussion illuminates a fundamental difficulty with designing a passive CFVS such as that described in SMUD Exhibit 11 and testified to by CEC witness Nix. The fairly precise single pressure setpoint that must be established for rupture of the discs[177] is inconsistent with the fact that the containment is likely to fail at various, currently unknown, pressure levels depending on the accident sequence,[178] wind and earthquake loadings, and the rate of containment pressure buildup.[179] Tr. 2871, 2872 (Meyer). Because of this multiplicity of required setpoints, it would be impossible to select a single disc rupture pressure that could accommodate all accident conditions without leading

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177 There would always be error bands in the rupture pressure of the discs. Tr. 2284, 2285 (Dieterich).

178 As noted earlier (<u>see</u> n.155, <u>supra</u>), the PWR-2 and PWR-3 release categories actually comprise a large number of accident sequences, each resulting in different containment temperature and pressure loadings and necessitating a different CFVS activation pressure. Tr. 2883 (Meyer).

179 A containment overpressurization situation is a dynamic environment in which rapid pressure surges are possible. In order to operate successfully and relieve pressure before containment fails, the CFVS might have to operate in some sequences at pressures well below the actual failure point of the containment in order just to "catch up" with the transient. Tr. 2232-2234, 2359-2360, 2368 (Dieterich). Thus, in some accident sequences identified by the Staff, a large (120 psig) pressure spike is experienced at the time when the molten core comes in contact with accumulator water. The pressure spike occurs so rapidly that the CFVS rupture pressure would have to be set quite low to accommodate it. Tr. 2828, 2829 (Meyer).

to unnecessary radioactive releases to the environment. Tr. 2359 (Dieterich); see also, Tr. 2826-2829 (Meyer).

275. The difficulties in determining the proper rupture pressure for the discs could be partially eliminated by replacing the discs with valves activated by the operators from the control room. However, such a solution would negate the main advantage of the system, i.e., its passive nature, and would introduce a host of possible failure modes -- such as unavailability of power sources to operate the valves, valve malfunction and human error -- that could decrease the reliability of the system. Tr. 2644-2647 (Nix); Tr. 2836 (Meyer). In failing to show that a workable setpoint can be established for actuation of a CFVS, the Board finds that CEC has failed to meet its burden of showing that the system suggested in its Issue 5-2 is effective and capable of implementation at Rancho Seco.

276. Assuming, nevertheless, in the interest of a thorough exploration of this issue, that the proper pressure setpoint for the CFVS could be achieved, one still has to determine whether the system would indeed be effective in minimizing radionuclide releases. Once the discs rupture, there is a sudden and potentially massive dynamic pressure surge through the pipes into the filtering and venting portion of the system.[180] Tr. 2664-2665, 2688-2689 (Nix). The release

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180 No studies have been conducted of the capability of the piping, filtering and venting elements of the CFVS to with-
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rate and the loading placed on the filtering and venting system would depend on the volume that was sought to be vented,[181] and on the size and number of the venting ports, discs and pipes that would need to be provided.[182] Tr. 2665 (Nix).

277. The filtering and venting scheme suggested by CEC witness Nix would have the piping coming out of containment lead to a "pressure relief volume" consisting of a concrete box buried underground and filled with sand and gravel. Tr. 2625, 2641 (Nix); Nix Testimony at 9, 11. The pressure relief volume would serve the multiple functions of heat sink, pressure mitigation volume, and radioactive material absorber. Id. Its dimensions would be on the order of 100 feet by 150 feet by 20 feet, for a volume of 300,000 cubic feet.[183] Tr. 2641, 2715

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stand a high pressure surge, nor of the possibility that the filtering material will be seriously disarranged by the pressure pulse as it travels across the filter. Tr. 2688-2689, 2716-2717 (Nix).

181 For some accident sequences (such as those leading to the 120 psig pressure surge described earlier) a very large volume would need to be discharged very quickly, necessitating a large penetration on the order of 20 feet in diameter; for other sequences, a 2 to 3-feet diameter penetration would suffice. Tr. 2829, 2878-2879 (Meyer); Tr. 2227 (Dieterich).

182 When Rancho Seco's containment was tested at 69 psig for overpressure, a 12-inch diameter line was used to relieve the pressure at the end of the test; it took several days to reduce the pressure back to atmospheric levels. Tr. 2227, 2228 (Dieterich). This experience suggests that fairly sizable openings would have to be utilized to provide a sufficient rate of pressure relief.

183 It appears that the dimensions suggested for the pressure relief volume are significantly smaller than would be required to perform its intended functions. The relief volume is only
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(Nix). The pressure relief volume would lead to a venting stack, perhaps filled with charcoal,[184] which would discharge directly into the atmosphere. Tr. 2655, 2656 (Nix).

278. A number of potential drawbacks of sand and gravel filters have been identified. Perhaps the most serious question which has been raised concerns the effectiveness of such filters. Staff witness Meyer testified that Swedish studies have reported discouraging decontamination factors for these filters. Tr. 2881 (Meyer). Moreover, such filters would not absorb radioactive iodine[185] or noble gases.[186] Tr. 2882

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300,000 cubic feet and is full of filtering material. The containment volume it has to accommodate is 2 million cubic feet at a pressure in excess of 70 psig. At the very least, the hot radioactive gases released at high pressure into the relatively small volume would move quickly through the filter and may not be held up long enough to be absorbed or decay (this is particularly true of noble gases). In addition, the motion of gas at high pressure through the filter material might tend to disarray it. See Tr. 2715-2719 (Nix).

184 The system originally proposed in SMUD Exhibit 11 did not include charcoal filters, but it was later discovered that charcoal might mitigate iodine releases. Nix Testimony at 11. As will be seen presently, charcoal filters appear to have substantial drawbacks that may outweigh their usefulness. See paragraph 280, infra.

185 Some types of gravel filters may suppress elemental iodine, but they are ineffective against organic iodine. Tr. 2882 (Meyer).

186 It was suggested that a sand and gravel filter might hold up noble gases for considerable periods of time. Tr. 2661-2663 (Nix). Such an assumption is questionable in view of the limited size of the pressure relief volume. Tr. 2718, 2719 (Nix). And, if the noble gases were indeed held up in the filter, they might present an even more serious problem because they would slowly leak out to the surface. Consequently, no effective measures could be taken to protect the public against them. Tr. 2663, 2664 (Nix).

(Meyer); Tr. 2659, 2660 (Nix). The latter is a very significant concern, for 80% of the krypton and xenon available in the core is released in a PWR-2 accident, representing millions of curies of radioactive matter which would not be absorbed or mitigated by the filter.[187] Tr. 2659-2662 (Nix).

279. Even if the sand and gravel filters were found to be effective absorbers of radioactive materials, they would still be subject to a number of potential operational problems. See Tr. 2641-2654 (Nix). The main problem identified is that the filters tend to become "plugged up", i.e., the pressure drop across the filter becomes excessive due to high humidity produced by steam condensation. When this occurs, there may be a backflow of the releases into the containment with the attendant risk of hydrogen deflagration.[188] Tr. 2643, 2644 (Nix).

280. The charcoal filters that might be relied upon for removing radioactive iodine also have a considerable number

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187 The health effect figures shown in SMUD Exhibits 11 and 18 and in the Nix Testimony for uncontrolled PWR-2 releases include the large noble gas releases as part of the dose to the population. The health effect figures provided in those documents for the filtered releases assumed that the noble gases and all other radioactive materials were discharged into the soil fifty feet beneath the surface and held there. Tr. 2663, 2686, 2718-2719 (Nix).

188 Solving this problem may require periodic replacement of filter materials. If, as assumed in the Nix Testimony and SMUD Exhibit 11, the pressure relief volume is buried underground, any filter replacement, maintenance or other corrective action would require digging the filter out and burying it back. Tr. 2647-2649 (Nix).

of potential problems. They have no resistance to fire, and radionuclides absorbed by the filter would be released in the event of a fire. Tr. 2655 (Nix). Moreover, the filtering capability of charcoal decreases with time, so that even in a standby mode they have to be replaced every three years. Id. See also, Tr. 2714, 2715 (Nix). These problems could be remedied by utilizing silver zeolite instead of charcoal to retain iodine, for silver zeolite does not suffer from these shortcomings. However, silver zeolite appears to be very costly.[189] See Tr. 2656, 2657 (Nix).

281. The Underground Siting Study (SMUD Ex. 11) and its supporting study (SMUD Ex. 18) report removal rates of essentially 100% for most radioisotopes by means of a CFVS. Tr. 2696-2697, 2711 (Nix); Nix Testimony at 13 and Table 5. However, the filtering method assumed to achieve these figures called for venting the containment atmosphere directly into the soil beneath an underground plant, 50 feet under the surface. Tr. 2663, 2683-2686, 2718-2719 (Nix). By doing so, the study took credit for the filtering and holdup capabilities of the soil, and thus assumed a filter essentially infinite in two of its three dimensions -- a much larger filter indeed than that available to surface facilities. Tr. 2712, 2718-2719, 2735-2736 (Nix). Since the filtering method assumed in SMUD Exhibit

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189 The cost of a CFVS computed in SMUD Exhibit 11 did not include either charcoal or silver zeolite filters, since none were included in the conceptual design. Tr. 2656 (Nix).

11 is not available for retrofit to surface facilities such as Rancho Seco (Tr. 2688 (Nix)), it is not clear that the very large reductions in health effects indicated (for example, in Table 5 of the Nix Testimony) could indeed be achieved in a surface facility, particularly in view of the uncertainty discussed above as to the effectiveness of sand, gravel and charcoal filters.

282. An alternative to the sand-gravel-charcoal approach utilized in SMUD Exhibit 11 could be a conceptual filtration system developed by Sandia Laboratories on an emergency basis for possible use during the TMI-2 accident and never implemented. Tr. 2834, 2835 (Meyer); Nix Testimony at 16. The Sandia TMI-2 concept utilized water treated with an iodine-capturing additive as the main filtering element, with possible sand and gravel filters to enhance radionuclide entrapment. Tr. 2669-2670, 2713 (Nix). Even with those refinements, not available in the Underground Siting Study approach, the Sandia TMI-2 concept only claimed removal of 90% of the radioactive iodine.[190] Tr. 2714, 2715 (Nix). It is not possible to assess the effectiveness of the Sandia TMI-2 retrofit approach or its applicability to Rancho Seco since the design was never implemented,[191] and was not recommended by

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190 The Sandia TMI-2 approach considered using charcoal as an agent to remove the remaining 10% of the iodine, but expressed reservations as to the ability of charcoal filters to remove even that small amount of iodine. Tr. 2714 (Nix).

191 CEC witness Nix recommended in his written testimony that (footnote continued next page)

Sandia for utilization elsewhere due to its hasty genesis and lack of detailed engineering design.[192] See Tr. 2667, 2668 (Nix). Again, in failing to identify a filtering system whose effectiveness has been reasonably established, the Board finds that CEC has failed to meet its burden of showing that the system suggested in CEC Issue 5-2 is workable.

283. In addition to these questions about the effectiveness of a CFVS in reducing radionuclide releases, additional unresolved issues have been identified regarding: (1) the interactions between a CFVS and the plant's engineered safety features; (2) the possibility of hydrogen ignition due to the system's operation; and (3) the potential adverse impact of a controlled filtered venting discharge because of the temperature reduction caused by the filtration system, which might make the resulting plume less buoyant than an unfiltered one. Meyer Testimony at 6; Dieterich Containment Testimony at 5, 6; Tr. 2250-2280, 2297 (Dieterich); Tr. 2691-2692, 2719-2725 (Nix); Tr. 2821-2825, 2835 (Meyer); Tr. 2835, 2836 (Greene).

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SMUD "use the Sandia-developed TMI retrofit concept as a beginning design concept" for application to Rancho Seco. Nix Testimony at 17. At the hearing, however, Mr. Nix stated that he had not performed any analyses that could lead him to form an opinion as to the effectiveness of the Sandia TMI-2 approach. Tr. 2671 (Nix).

192 An apparently more refined filtering system conceptualized by Sandia in a recent study, still based on a radionuclide suppression water pool enclosing a gravel filter, is claimed to produce attenuation factors for particulates and elemental iodine greater than 98%. Tr. 2905, 2906 (Meyer); Meyer Testimony at 2.

These potential problems, which the Board addresses below, could in fact exacerbate overpressurization accidents or increase their risk.

284. There are accident scenarios in which venting through a CFVS would take place and yet the containment would not have otherwise failed. Tr. 2825-2827, 2830 (Meyer). For those accidents, in addition to the unnecessary radioactive releases produced by venting, several engineered safety features could be compromised by the depressurization of the containment.[193] Thus, in primary system large break accidents, containment overpressurization could interfere with subsequent reflooding of a dried out core by failing to provide back pressure in the containment. Tr. 2821-2824 (Meyer); Tr. 2253-2260 (Dieterich); Dieterich Containment Testimony at 5. Depressurization could also cause flashing of the containment sump water leading to cavitation and disabling of the reactor building spray pumps and the low pressure injection pumps. Tr. 2260-2266 (Dieterich); Tr. 2824 (Meyer); Dieterich Containment Testimony at 5. Depressurization followed by actuation of the reactor building spray system could create a vacuum leading to emptying of the containment sump. Tr. 2825 (Meyer); Dieterich Containment Testimony at 6.

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193 For those accidents in which the containment is inevitably going to fail, these problems are of no special consequence, since they would arise whether or not venting took place. Tr. 2825 (Meyer). However, if there is premature or spurious actuation of the CFVS, these problems would be of concern. Id.

285. Another area of concern is the possibility that venting may exhaust much of the air in the containment and, if the core melt subsequently reacts with water entrapped in the concrete base mat to produce hydrogen, the hydrogen would burn or explode as it is emitted because of its high concentration relative to the containment atmosphere. Tr. 2273-2276 (Dieterich); Tr. 2723 (Nix); Dieterich Containment Testimony at 6. A hydrogen fire or explosion could also be produced in the vent line itself.[194] Tr. 2274-2278 (Dieterich); Tr. 2724 (Nix).

286. There is also a potential adverse effect from the temperature drop in the gases emitted from containment as they move through the filter and are discharged by the venting stack. The resulting plume would be at a lower temperature, and thus less buoyant, than the plume produced by an unfiltered release. As a result, the plume emitted by the CFVS would disperse less, and could give a higher dose[195] to a smaller area, than an unfiltered plume. Tr. 2278-2283 (Dieterich); Dieterich Containment Testimony at 6.

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194 As noted earlier, backflow from the filtering section of the CFVS can also cause a hydrogen fire in the containment. See paragraph 279, supra.

195 While some radionuclides might have been removed by the filtering process, the plume from a CFVS would contain large amounts of noble gases and possibly organic iodine. Tr. 2280, 2281 (Dieterich). The question, which remains unresolved, is whether decontamination through filtering can make up for a more concentrated dose due to a less buoyant plume. Tr. 2724, 2725 (Nix).

287. The design and performance uncertainties and potential problems previously discussed are reflected in an even greater uncertainty as to the cost of implementing a CFVS. See Nix Testimony at 17. The Underground Siting Study estimated a cost of 14 million 1977 dollars for implementing such a system in a new facility. Tr. 2491, 2640 (Nix); Nix Testimony at 17. The costs of retrofitting an existing facility such as Rancho Seco with a CFVS could include, among other things: the cost of creating a number of new large containment penetrations[196] (Nix Testimony at 17); the cost of building the CFVS to seismic-1 standards[197] (Tr. 2638-2640 (Nix)); the cost of adding expensive filtering materials to absorb radionuclides such as iodine and noble gases (Tr. 2656-2657, 2664 (Nix); Tr. 2879, 2880 (Meyer)); the cost of ensuring reliability if the system is active (as opposed to a passive design), and especially if it is manually operated (Tr. 2816 (Meyer)); the cost of developing and licensing the system (Tr. 2287 (Dieterich); Tr. 2679, 2680 (Nix)); the cost of down time during installation (Nix Testimony at 17); and the cost of procuring, installing and maintaining the system.[198] These

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196 Licensee witness Dieterich testified that all containment penetrations at Rancho Seco are committed to other uses and there are no available penetrations large enough to be useful for venting. Tr. 2388, 2389 (Dieterich). Large new penetrations would have to be made leak tight, resulting in very costly plant modifications. Tr. 2384 (Dieterich).

197 CEC witness Nix estimated that a CFVS designed to seismic-1 standards would cost about 50% more than if such standards did not have to be met. Tr. 2639, 2640 (Nix).

198 Staff witness Meyer testified that a preliminary estimate
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large cost uncertainties make it impossible for the Board to determine whether any risk reduction benefits that might be achieved by use of a CFVS would be outweighed by the costs of its development, licensing, installation and maintenance. CEC, then, has failed again to meet its evidentiary burden in order to have a CFVS considered for implementation at Rancho Seco.

288. In conclusion, CEC has not met the burden on its Issue 5-2 set by the Board above in paragraph 247. Many of the open questions regarding the feasibility, effectiveness and risk reduction potential of CFVS, however, may be resolved in the next two or three years. The Commission is conducting an extensive, high priority analysis and design program to address these areas of uncertainty in containment overpressurization protection, and the utilities will undertake simultaneously a parallel program on several of these questions. Meyer Testimony at 6, 7; Tr. 2839, 2840 (Meyer). The TMI-2 Lessons Learned Task Force and the NRC Staff's TMI-2 Action Plan have recommended that the Commission conduct a rulemaking proceeding on methods for mitiging the consequences of core melt accidents, including CFVS. Dieterich Containment Testimony at 7, 8; Meyer Testimony at 7. The rulemaking will cover a very broad spectrum of questions regarding core melt and core

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(continued)
of the cost of retrofitting the Indian Point facility with a CFVS is from $15 to $50 million. Tr. 2815-2820 (Meyer). This cost assumed use of an existing 3 foot-diameter penetration for the CFVS. Tr. 2820 (Meyer).

degradation accidents, and may result in guidelines, design bases and requirements to be imposed on all operating licensees. Tr. 2841, 2893-2896 (Meyer). The specific analyses of plant requirements will be undertaken by means of the IREP, which as noted above (paragraph 262, supra) includes a probabilistic analysis of all reactors along the lines of the Reactor Safety Study to identify dominant accident sequences[199] and to provide the opportunity for development of any revised design bases and requirements. Tr. 2840-2841, 2893-2894 (Meyer).

289. Simultaneously with the rulemaking and the analysis and development efforts, the NRC Staff is conducting a study of implementing a CFVS at the Indian Point 3 and Zion plants. Tr. 2888, 2897 (Meyer). These plants were selected for separate study because they are located in very high population density areas near New York City and Chicago. Id.; Tr. 2246 (Dieterich).[200]

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199 While the dominant accident sequences may be found to vary from plant to plant, it is expected that the number of such sequences identified as a result of the rulemaking will be small and relatively insensitive to reactor characteristics. Tr. 2814, 2840, 2891-2892 (Meyer). IREP is completing a dominant accident sequence for the Crystal River 3 plant and will next undertake such a study of the Indian Point 3 and Zion plants. Tr. 2876, 2877 (Meyer).

200 Health effects and economic impacts in the event of an uncontrolled release from a PWR-2 or PWR-3 accident sequence would be higher for those plants than for Rancho Seco because of the lower population distribution in the vicinity of the Rancho Seco plant. See Tr. 2593-2599 (Nix).

290. This significant effort by the Staff suggests that clarification of unanswered questions in this novel area of reactor safety[201] may be forthcoming. Meanwhile, there remain the host of questions, discussed above, which would have to be answered before a conclusion could be made on whether the risk to society would be substantially reduced if a CFVS were installed at Rancho Seco. Tr. 2838, 2878-2879, 2891 (Meyer). The appropriate arena for consideration of CFVS feasibility is the rulemaking proceeding soon to be instituted. The Board has been provided no basis upon which the Rancho Seco plant should be singled out for consideration of a CFVS at this time;[202] to

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201 No commercial light water reactor in the United States has ever utilized a CFVS. Tr. 2372 (Nix). Applications of the concept in this country and abroad have been to liquid metal fast breeder reactor facilities using a small volume of liquid sodium as primary coolant instead of large volumes of water as in PWRs and boiling water reactors. Tr. 2239-2241 (Dieterich). The experience gained at those facilities is not generally transferable to commercial reactors because the overpressurization and containment failure sequences and scenarios would be totally unlike those for light water reactors. Tr. 2360, 2361 (Dieterich).

202 The Commission's Director, Office of Nuclear Reactor Regulation ("NRR"), recently denied a petition by FOE under 10 C.F.R. § 2.206 to require the NRC to prepare a supplemental environmental impact statement on Class 9 accidents at Rancho Seco and two other plants. Arizona Public Service Company (Palo Verde Nuclear Generating Station, Units 1-3), et al., DD-80-22, 11 N.R.C. ___ (June 19, 1980). In denying FOE's petition with respect to Rancho Seco, NRR found that the Rancho Seco design was not novel "but rather typical for a land-based pressurized water reactor"; that the population density around Rancho Seco would still remain "well within the [Regulatory Guide 4.7 and 10 C.F.R. Part 100] guidelines" by the year 2000; and that "[u]sing conservative assumptions, the Staff estimates that it would take tens of years for contaminated groundwater to migrate to the nearest well which is located at the site boundary." Slip opinion at 13-14. Based on these considerations, NRR concluded that "[t]here are no special or unusual circumstances surrounding
(footnote continued next page)

do so would require going beyond or duplicating ongoing Staff and Commission efforts. Tr. 2890 (Meyer). This would be an unnecessary expenditure of time, effort and resources[203] in view of the fact that the Rancho Seco containment meets all licensing requirements, criteria and regulations currently in place, can withstand safely a severe design basis over-pressurization accident,[204] and will not fail due to over-pressurization from a feedwater transient unless multiple failures and extremely improbable conditions occur. Tr. 2238, 2239 (Dieterich); Tr. 2808, 2809 (Greene); Meyer Testimony at 7. Therefore, the Board finds that the Rancho Seco facility is safe to operate without a CFVS and the answer to CEC Issue 5-2 is that the containment building at Rancho Seco should not be modified to provide overpressurization protection by means of such a system. See Meyer Testimony at 7; Tr. 2837 (Meyer); Dieterich Containment Testimony at 9.

(continued)
the Rancho Seco Station which would warrant re-opening environmental proceedings on the facility." Id. at 14. These findings confirm that there is no feature of the Rancho Seco design or site that would make it advisable to consider the CFVS issue outside the Commission rulemaking proceeding.

203 The Underground Siting Study alone, which produced a conceptual design of a CFVS without any specific implementation details, cost $1.3 million to complete. Tr. 2718 (Nix). It did, however, comprise other research areas in addition to the CFVS concept.

204 The TMI-2 accident, considered by many analysts as rather severe, resulted in a 28 psig pressure pulse that did not threaten the integrity of the containment despite large amounts of radioactive releases to the containment. Meyer Testimony at 7; Dieterich Containment Testimony at 7, 8.

P. Concluding Findings of Fact

291. As we stated at the outset, the Board views its charge from the Commission to be to determine whether the actions and modifications required by the Commission in its Order of May 7, 1979, provide reasonable assurance that the Rancho Seco facility will respond safely to feedwater transients. No participant in this hearing contended that the May 7 Order is inadequate. In the foregoing findings of fact, however, the Board has carefully examined the lengthy evidentiary record compiled in response to questions raised by the Board and the California Energy Commission on the adequacy of that order. We have made findings on each of the 29 questions raised (including 4 contentions of a withdrawn intervenor).

292. The Board has assessed the adequacy of the requirements of the May 7 Order against the factual background in which they were imposed and implemented -- including a recognition of other changes made at Rancho Seco since the Three Mile Island accident. For example, the short-term actions which were completed prior to the restart of the facility on July 5, 1979, were viewed with the knowledge that changes had already been made in April, 1979, in response to bulletins issued by the NRC's Office of Inspection and Enforcement. The long-term modifications, similarly, are not the only changes which have been made and are being made to the facility since it restarted.

293. Neither Licensee nor any other participant in the hearing formally challenged the necessity of the short-term actions required by the Commission's Order of May 7, 1979, to provide reasonable assurance that the facility will respond safely to feedwater transients pending completion of the long-term modifications in the order. The actions were completed prior to the restart of the facility. Likewise, no one questioned whether Licensee should be required to accomplish, as promptly as practicable, the long-term modifications set forth in the May 7 Order. The record shows that Licensee has accomplished some of these modifications already, and is actively implementing the remainder.

294. The questions raised and probed by the Board did go to the sufficiency of the short-term actions pending completion of the long-term modifications. The record shows, and we have found, that the prompt changes accomplished after the Three Mile Island accident and prior to the restart of Rancho Seco on July 5, 1979, provided added reliability to the reactor system and the operators at the facility to respond safely to feedwater transients. The long-term modifications are intended to enhance this reliability even further. The Board does not find, however, that additional short-term actions were <u>necessary</u> to provide reasonable assurance that the facility will respond safely to feedwater transients pending completion of the long-term modifications.

295. The suffic'ency of the long-term modifications to provide continued reasonable assurance that the facility will respond safely to feedwater transients was also raised in the questions explored in this proceeding. Since May 7, 1979, of course, the Commission has imposed numerous additional requirements upon this and other facility operating licensees in response to the investigations of, and lessons learned from, the accident at Three Mile Island. The record compiled here has left the Board well informed on the extent and implications of these additional requirements. While some witnesses (but no parties) have suggested even further modifications, the Board does not recommend the need for any amendment to the Commissi n's Order of May 7, 1979. The suggestions advanced are not supported by a record which warrants their adoption at this time, although some may deserve additional study by the NRC and industry on a generic basis. None of them are required, however, to provide reasonable assurance that the facility will respond safely to feedwater transients. The Board finds that the long-term modifications directed by the Commission in its Order of May 7, 1979, along with the other changes, are sufficient to provide such assurance.

III. CONCLUSIONS OF LAW

296. The Board has considered all documentary and oral evidence presented by the parties and by the witnesses for the interested state representative on the questions raised by the Board and former parties and on the issues raised by the interested state representative. Based upon a review of the entire record in this proceeding and the foregoing findings of fact, the Board enters the following conclusions of law.

297. The actions required by subparagraphs (a) through (e) of Section IV of the Commission's Order of May 7, 1979, 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 of the May 7 Order.

298. Licensee should be required to accomplish, as promptly as practicable, the long-term modifications set forth in Section II of the Commission's Order of May 7, 1979.

299. These long-term modifications, coupled with the additional changes completed and being undertaken at the facility, are sufficient to provide continued reasonable assurance that the facility will respond safely to feedwater transients.

IV. ORDER

300. 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 Commissior 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 th Initial Decision or designated portions thereof must be fil thin ten (10) days after service of the decision. A n 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 exceptions.

IT IS SO ORDERED.

Respectfully submitted,

SHAW, PITTMAN, POTTS & TROWBRIDGE

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Thomas A. Baxter

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(202) 331-4100

Dated: July 11, 1980

Appendix A

WRITTEN TESTIMONY RECEIVED INTO EVIDENCE

WITNESS	Following Transcript Page
Allenspach, Frederick R. "Testimony of Frederick R. Allenspach Relating to Management and Technical Competence (FOE III(d) and Board Question 32)"	3920
Bridenbaugh, Dale G., and Gregory C. Minor "Prepared Direct Testimony of Dale G. Bridenbaugh and Gregory C. Minor Concerning Operator Training and Human Factors Engineering"	3496
Canter, Harvey L. "NRC Staff Testimony of Harvey L. Canter Relative to the Competence of SMUD to Operate the Rancho Seco Facility (FOE Contention III(d) and Board Question 32)"	3920
Capra, Robert A. "Testimony of Robert A. Capra on Implementation of Long-Term Modifications Established in the Commission Order of May 7, 1979 (FOE Contention III(c))"	1163

WITNESS	Following Transcript Page
Diet rich, Robert A.	
"Licensee's Testimony of Robert A. Dieterich in Response to Licensing Board Question CEC 1-6; California Energy Commission Issue 5-1; Board Questions H-C 9, 20; Friends of the Earth Contention III(c); and, Additional Board Question 1"	1988
"Licensee's Testimony of Robert A. Dieterich in Response to California Energy Commission Issue 5-2"	1988
"Licensee's Supplemental Testimony of Robert A. Dieterich in Response to Board Question H-C 20"	1988
Donohew, Jack N.	
"NRC Staff Testimony of Jack N. Donohew on Changing the Systems Outside Containment to Vent into Containment Building (CEC Issue 5-1)"	3168
Gagliardo, James E., and Darrell G. Hinckley	
"Supplemental Testimony of NRC Performance Appraisal Branch Regarding SMUD Management Controls"	4233
Greene, Thomas A.	
"NRC Staff Testimony of Thomas A. Greene on Containment Overpressurization Protection (CEC Issue 5-2)"	2783
"NRC Staff Testimony of Thomas A. Greene on Hydrogen Recombiner (Board Question 20)"	2783
Hinckley, Darrell G.	
(see Gagliardo, *supra*)	4233

WITNESS	Following Transcript Page
Johnson, Allen D. "NRC Staff Testimony of Allen D. Johnson Relative to the Competency of SMUD to Operate the Rancho Seco Facility (FOE Contention III(d) and Board Question 32)"	3920
Jones, Robert C., and Bruce A. Karrasch "Licensee's Testimony of Bruce A. Karrasch and Robert C. Jones in Response to Licensing Board Questions CEC 1-2, 1-4, 1-7, 1-10; California Energy Commission Issues 1-1, 1-12; Licensing Board Questions H-C 16, 21, 24; Friends of the Earth Contention III(a); and, Additional Board Questions 1, 2 and 3"	535
"Licensee's Supplemental Testimony of Bruce A. Karrasch and Robert C. Jones in Response to Licensing Board Question H-C 21"	535
Karrasch, Bruce A. (see Jones, *supra*)	535
Lewis, Harold W. "Prepared Direct Testimony of Dr. Harold W. Lewis Concerning Natural Circulation Cooling"	477
Mann, Bruce J. "Prepared Direct Testimony of Bruce J. Mann Concerning Release of Radioactivity from Containment (CEC Issue 5-1)"	2926
Matthews, Philip R. "NRC Staff Testimony of Philip R. Matthews, Adequacy of the Pressurizer and Pressurizer Relief Tank Size (Board Question 21)	1163
"NRC Staff Testimony of Philip R. Matthews on Reliability and Timeliness of the Emergency Feedwater System (Board Question CEC 1-6)"	1163

WITNESS	Following Transcript Page
Novak, Thomas M.	
"NRC Staff Testimony of Thomas M. Novak Regarding Reconsideration of the Requirements for Automatic and Manual Safety Actions (CEC Issue 5-3a)"	1163
Novak, Thomas M., and Mark P. Rubin	
"NRC Staff Testimony of Mark P. Rubin and Thomas M. Novak Regarding the Sensitivity of the Once-Through Steam Generator Design (Additional Board Question 3)"	1163
"NRC Staff Testimony of Mark P. Rubin and Thomas M. Novak Regarding the Acceptability of Feedwater Transients Referenced in NUREG-0560 (FOE Contention IIIa)"	1163
"NRC Staff Testimony of Mark P. Rubin and Thomas M. Novak Regarding the Design Basis for Rancho Seco Safety Systems (CEC Contentions 1-1 and 1-12)"	1163
Performance Appraisal Branch	
(see Gagliardo, supra)	4233
Rodriguez, Ronald J.	
"Licensee's Testimony of Ronald J. Rodriguez in Response to Licensing Board Questions CEC 1-2, 1-6, 1-7, 5-3a; California Energy Commission Issues 1-1, 1-12, 3-1, 3-2, 3-3; Friends of the Earth Contentions III(d), III(e); and, Additional Board Questions 2 and 3"	2948
Rubin, Mark P.	
(see Novak and Rubin, supra)	1163

WITNESS	Following Transcript Page
Thatcher, Dale F.	
"NRC Staff Testimony of Dale F. Thatcher Relative to Direct Initiation of Reactor Trip Upon the Occurrence of Off-Normal Conditions in the Feedwater System (Board Question 9 and Additional Board Question 1)"	1163
"NRC Staff Testimony of Dale F. Thatcher Relative to the Integrated Control System (Board Question 16)"	1163
Webb, Clifford M.	
"Prepared Direct Testimony of Clifford M. Webb Concerning Design Sensitivities of the Babcox and Wilcox Nuclear Steam Supply System"	1801
Wilson, Bruce A.	
"NRC Staff Testimony of Bruce A. Wilson on Operator Training and Competence (Board-CEC Question 1-7, CEC Issue 3-1, CEC Issue 3-2, CEC Issue 3-3, Board Question 32 and FOE Contention III(e))"	3788
"NRC Staff Testimony of Bruce A. Wilson on Control Room Design (Board Question 31)"	3788
"NRC Staff Testimony of Bruce A. Wilson on Instrumentation for Diagnosis and Control of Off-Normal Conditions (CEC Issue 5-3a)"	3788

NOTE: NRC Staff's Evaluation of Licensee's Compliance with the NRC Order Dated May 7, 1979, follows transcript page 362.

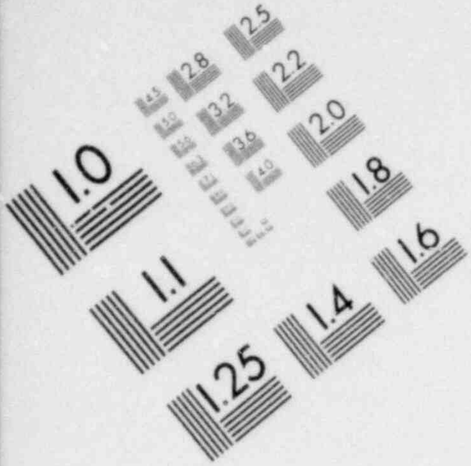
Appendix B

EXHIBITS

Exhibit Number	Description	Identified at Transcript Page	Admitted at Transcript Page
Board Ex. 1	"Report Review: Integrated Control System Reliability Analysis," prepared by the Instrumentation and Controls Division, Oak Ridge National Laboratory	[Identified as as CEC Ex. 4 at Tr. 649]	1352
SMUD Ex. 1	Crystal River Loss of NNI Power Event - 2/26/80; Sequence of Events prepared by Bruce A. Karrasch	435	438
SMUD Ex. 2	Letter dated January 29, 1980 from John J. Mattimoe to Robert W. Reid attaching report entitled "Potential Reactor System Voiding During Anticipated Transients"	1708	1711
SMUD Ex. 3	Statement of Affiliations and Qualifications of Prospective Witnesses, Clifford M. Webb	1802	

Exhibit Number	Description	Identified at Transcript Page	Admitted at Transcript Page
SMUD Ex. 4	Affidavit of Joseph McCarthy dated April 3, 1980	1812	
SMUD Ex. 5	Affidavit of William E. Kessler dated April 3, 1980	1832	
SMUD Ex. 6	Letter dated March 31, 1980, from D.H. Nelson to Thomas A. Baxter	1832	
SMUD Ex. 7	California Energy Commission Responses to Licensee's First Set of Interrogatories, dated December 24, 1979	1852	
SMUD Ex. 8	Letter dated December 12, 1979 from Lester Rubenstein to James H. Taylor enclosing "Safety Evaluation, Atypical Weld Metal"	1883	
SMUD Ex. 9	California Energy Commission's Responses To First Set of NRC Staff Interrogatories, dated December 5, 1979	2404	2612
SMUD Ex. 10	California Energy Commission's Responses To the Licensee's Second Set of Interrogatories, dated January 17, 1980	2404	2613

Exhibit Number	Description	Identified at Transcript Page	Admitted at Transcript Page
SMUD Ex. 11	California Energy Commission Staff Report (Draft), "Underground Siting of Nuclear Power Reactors: An Option for California," dated June, 1978	2423	2614
SMUD Ex. 12	Excerpt of transcript a Hearing before a Committee of the [California] State Energy Resources Conservation and Development Commission, <u>In the Matter of: Notice of Intention for Sundesert Nuclear Project</u>, Docket No. 76-NOI-2 (Testimony of Daniel Nix), dated August 5, 1977	2452	
SMUD Ex. 13	"The Safety of Fission Reactors," by Harold W. Lewis, <u>Scientific American</u>, Vol. 242, No. 3 (March 1980), pp. 53-65	2462	
SMUD Ex. 14	Excerpts from "Evaluation of the Feasibility, Economic Impact, and Effectiveness of Underground Nuclear Power Plants; Final Technical Report," May 1978, prepared by The Aerospace Corporation	2470	



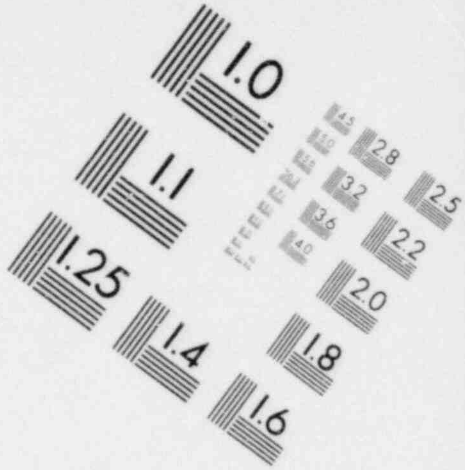
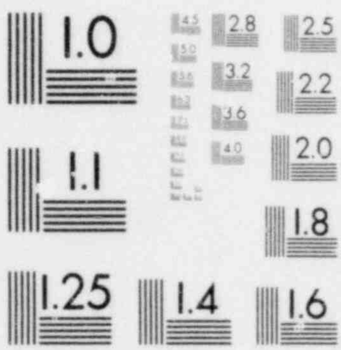


IMAGE EVALUATION
TEST TARGET (MT-3)



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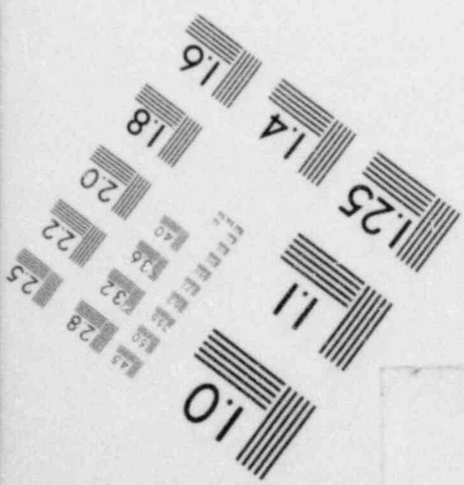
MICROCOPY RESOLUTION TEST CHART



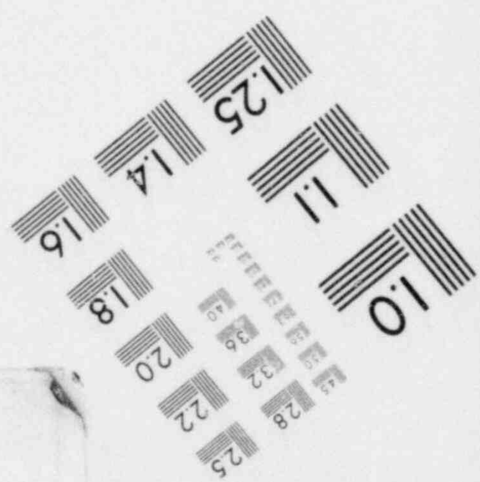


IMAGE EVALUATION
TEST TARGET (MT-3)

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MICROCOPY RESOLUTION TEST CHART

Exhibit Number	Description	Identified at Transcript Page	Admitted at Transcript Page
SMUD Ex. 15	Excerpts from the Report of The President's Commission on The Accident at Three Mile Island, "The Need for Change: The Legacy of TMI," October 1979	2509	
SMUD Ex. 16	Excerpts from "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," Main Report, October 1975 by the U.S. Nuclear Regulatory Commission.	2510	
SMUD Ex. 17	Excerpts from "Analysis of Public Consequences from Postulated Severe Accident Sequences in Underground Nuclear Power Plants," by Advanced Research and Applications Corporation	2566	
SMUD Ex. 18	"Analysis of Public Consequences from Postulated Severe Accident Sequences in Underground Nuclear Power Plants," by Advanced Research and Applications Corporation	2611	2611
SMUD Ex. 19	Letter dated May 8, 1979 from Allan S. Benjamin to Raymond di Salvo attaching "Contingency Vent-Filter for the Three Mile Island Unit II Reactor"	2666	

Exhibit Number	Description	Identified at Transcript Page	Admitted at Transcript Page
SMUD Ex. 20	NRC Staff Response to Questions 25 & 26, Aamodt's Sixth Set of Interrogatories, dated March 31, 1980 (Metropolitan Edison Company, Three Mile Island, Unit 1, Docket No. 50-289)	3480	
SMUD Ex. 21	Resume of Gregory C. Minor	3498	
Staff Ex. 1	Crystal River Sequence of Events, dated February 28, 1980 by John A. Olshinski, NRC	469	470
Staff Ex. 2	"Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox Designed 177-FA Operating Plants," NUREG-0565, January 1980 by the U.S. Nuclear Regulatory Commission	1161	1162
Staff Ex. 3	Draft, "Transient Response of of Babcock & Wilcox Designed Reactors," NUREG-0667, April 1980 by the U.S. Nuclear Regulatory Commission	1230	1230
Staff Ex. 4	Memorandum dated May 1, 1980 from Robert L. Tedesco to Harold R. Denton with attached final version of "Transient Response of Babcock & Wilcox Designed Reactors," NUREG-0667	3652	3662

Exhibit Number	Description	Identified at Transcript Page	Admitted at Transcript Page
Staff Ex. 5	Memorandum dated May 9, 1980 from Rodney M. Satterfield to Paul S. Check with attached "Assessment of B&W Report BAW-1564, 'Integrated Control System Reliability Analysis'"	4137	4137
CEC Ex. 1	Licensee's Answers to California Energy Commission Requests for Admissions to Sacramento Municipal Utility District, dated January 16, 1980	471	472
CEC Ex. 2	NRC Staff Response to California Energy Commission Requests for Admissions to Nuclear Regulatory Commission, dated January 17, 1980	473 [as corrected]	473
CEC Ex. 3	"Integrated Control System Reliability Analysis," Report No. BAW-1564, August 1979 by Babcock & Wilcox	629	1042
CEC Ex. 4	"Report Review: Integrated Control System Reliability Analysis," prepared by the Instrumentation and Controls Division, Oak Ridge National Laboratory	649	[admitted as Board Ex. 1 at Tr. 1352]

Exhibit Number	Description	Identified at Transcript Page	Admitted at Transcript Page
CEC Ex. 5	Letter dated November 16, 1979 from W.P. Gammill to all Babcock & Wilcox Plants with an Operating License attaching October 25, 1979 letter from H.R. Denton to Babcock & Wilcox Construction Permit Holders	660	1239
CEC Ex. 6	Brookhaven National Laboratory Memorandum dated January 10, 1980 from C.J. Hsu, "B&W Overfeed Transient Analysis Using the IRT Code"	727	
CEC Ex. 7	Licensee's Answers (Set No. 1) to the First Set of Interrogatories of the California Energy Commission, dated December 4, 1979	727	
CEC Ex. 8	Licensee's Answers (Set No. 2) to the California Energy Commissions First Set of Interrogatories, dated December 4, 1979	727	
CEC Ex. 9	Licensee's Answers (Set No. 3) to the First Set of Interrogatories of the California Energy Commission, dated December 1, 1979	727	

Exhibit Number	Description	Identified at Transcript Page	Admitted at Transcript Page
CEC Ex. 10	Licensee's Answers (Set No. 1) to California Energy Commission's Second Set of Interrogatories to the Sacramento Municipal Utility District, dated January 17, 1980	727	
CEC Ex. 11	Licensee's Answers (Set No. 2) to California Energy Commission's Second Set of Interrogatories to the Sacramento Municipal Utility District, dated January 17, 1980	727	
CEC Ex. 12	Licensee's Answers (Set No. 3) to California Energy Commission's Second Set of Interrogatories to the Sacramento Municipal Utility District, dated January 16, 1980	727	
CEC Ex. 13	NRC Staff Responses to California Energy Commission's First Set of Interrogatories to the Nuclear Regulatory Commission, dated December 11, 1979	727	
CEC Ex. 14	NRC Staff's Responses to California Energy Commission's Second Set of Interrogatories to the Nuclear Regulatory Commission, dated January 17, 1980	727	

Exhibit Number	Description	Identified at Transcript Page	Admitted at Transcript Page
CEC Ex. 15	Letter dated December 4, 1979 from S.H. Howell to H.R. Denton attaching Revision 1 to response to October 25, 1979 request regarding B&W System Sensitivity	907	1043
CEC Ex. 16	Table attachments to CEC Ex. 15	907	1043
CEC Ex. 17	Letter dated August 29, 1979 from J.T. Janis to R.J. Rodriguez with attached analysis of Reactor Coolant Pump Trip for Non-LOCA Cases	966	
CEC Ex. 18	Letter dated September 5, 1979 from J.T. Janis to R.J. Rodriguez, with attached revised Parts I and II of the Small Break Operating Guidelines	1026	
CEC Ex. 19	Letter dated October 24, 1979 from J.J. Mattimoe to R.H. Engelken with attached revised Section III of "Analysis Summary in Support of an Early RC Pump Trip"	1100	1111

Exhibit Number	Description	Identified at Transcript Page	Admitted at Transcript Page
CEC Ex. 20	Letter dated December 17, 1979 from J.J. Mattimoe to R.W. Reid with attached "Auxiliary Feedwater System Reliability Analysis for the Rancho Seco Nuclear Generating Station Unit No. 1" and listing of Auxiliary Feedwater System "Outstanding NUREG-0578 Items"	1502	2081
CEC Ex. 21	Letter dated February 26, 1980 from R.W. Reid to J.J. Mattimoe with attached NRC Staff review of documents attached to CEC Ex. 20 and implementation schedule	1541	1656
CEC Ex. 22	Letter dated March 18, 1980 from J. J. Mattimoe, to R.W. Reid with attached response to February 26, 1980 NRC letter (CEC Ex. 21)	1624	2097
CEC Ex. 23	Letter dated January 9, 1980 from R.W. Reid to all licensees of Babcock & Wilcox plants, "Concern for Voiding during Transients on B&W Plants"	1668	1676
CEC Ex. 24	Excerpts from NRC Staff slide presentation at January 8, 1980 Advisory Committee on Reactor Safeguards meeting	1774	

Exhibit Number	Description	Identified at Transcript Page	Admitted at Transcript Page
CEC Ex. 25	Letter dated April 27, 1979 from J.J. Mattimoe to H.R. Denton committing to shutdown of Rancho Seco and completion of specified actions	2026	2076
CEC Ex. 26	"NRR Status Report on Feedwater Transients in B&W Plants," dated April 25, 1979	2027	3743
CEC Ex. 27	Letter dated April 28, 1979 from J.H. Taylor to H.R. Denton with attached "Scope and Schedule for a Reliability Analysis of the Integrated Control System (ICS)"	2036	
CEC Ex. 28	"Subcommittee on Systems and Equipment Design Criteria Recommendations to the AIF Policy Committee on Follow-Up to the Three Mile Island Accident," dated September 12, 1979.	2132	
CEC Ex. 29	Rancho Seco Final Safety Analysis Report Sections 5.2.3 and 5.2.4, Table 5.2-2 and Figure 5.2-9	2137	2297

Exhibit Number	Description	Identified at Transcript Page	Admitted at Transcript Page
CEC Ex. 30	Letter dated January 7, 1980 from J.J. Mattimoe to R.W. Reid with attached status of NUREG-0578 actions	2140	
CEC Ex. 31	Record of telephone conversation dated April 24, 1979 between Dennis Keyes, Bob Stein and Lee Keilman regarding Palo Verde #1 hydrogen recombiners	2154	
CEC Ex. 32	"Program Plan for the Investigation of Vent-Filtered Containment Conceptual Designs for Light Water Reactors," by Allan S. Benjamin, Sandia Laboratories, NUREG/CR-1029, SAND 79-1088, October 1979	2247	
CEC Ex. 33	"Human Factors Review of Nuclear Power Plant Control Room Design," EPRI NP-309, November 1976. Prepared by Lockheed Missiles & Space Company, Inc. for Electric Power Research Institute	2967	
CEC Ex. 34	WITHDRAWN [Diagram of Rancho Seco Control Room]		

Exhibit Number	Description	Identified at Transcript Page	Admitted at Transcript Page
CEC Ex. 35	SMUD/Rancho Seco Administrative Procedure 25, "Licensed NRC Operator Retraining"	3082	3106
CEC Ex. 36	Deposition of Dennis E. Tipton taken January 25, 1980	3107	3107
CEC Ex. 37	Deposition of Daniel E. Comstock taken January 24, 1980	3107	3107
CEC Ex. 38	Deposition of Wayne S. Morisawa taken January 24, 1980	3108	3108
CEC Ex. 39	Letter dated August 1, 1979 from J.L. Crews to J.J. Mattimoe with attached "U.S. Nuclear Regulatory Commission Office of Inspection and Enforcement Region V Report No. 50-312/79-14"	3124	4188
CEC Ex. 40	Twenty "Abnormal Occurrence"/ "Reportable Occurrence" reports pertaining to the Rancho Seco Nuclear Generating Station	3140	3215

Exhibit Number	Description	Identified at Transcript Page	Admitted at Transcript Page
CEC Ex. 41	Letter dated May 1, 1980 from R.W. Reid to J.J. Mattimoe with attached "Evaluation of Licensee's Compliance with Category "A" Items of NRC Recommendations Resulting from TMI-2 Lessons Learned"	3179	
CEC Ex. 42	Excerpts from the Rancho Seco training records for M.A. Carter	3270	
CEC Ex. 43	Rancho Seco Nuclear Generating Station Unit No. 1 Procedure Change Approval Form for Procedure D.5, Rev. 14 dated September 5, 1979 with attached Procedure D.5, "Loss of Reactor Coolant/Reactor System Pressure"	3280	3421
CEC Ex. 44	Letter dated October 17, 1979 from D.G. Eisenhut, NRC, to All Operating Nuclear Power Plants with attached "North Anna Unit 1 Radioactivity Release Pathway"	3296	
CEC Ex. 45	Letter dated April 3, 1980 from D.G. Eisenhut to J.J. Mattimoe regarding the March 20, 1978 Rancho Seco transient with attached memorandum, "Single Failure Potentially Leading to Core Damage"	3303	

Exhibit Number	Description	Identified at Transcript Page	Admitted at Transcript Page
CEC Ex. 46	Rancho Seco Procedure D.5, "Loss of Reactor Coolant/Reactor Coolant System Pressure," Revision 15, dated March 4, 1980	3421	3421
CEC Ex. 47	Excerpts from transcripts of Nuclear Regulatory Commission Office of Inspection and Enforcement Public Hearing held on May 2, 1980	3490	
CEC Ex. 48	Sacramento Municipal Utility District Post TMI-2 Training Quiz	3796	
CEC Ex. 49	Letter dated March 29, 1980 from H.R. Denton to All Power Reactor Applicants and Licensees with attached criteria for reactor operator qualification and training	3818	3820
CEC Ex. 50	Eleven "Abnormal Occurrence"/"Reportable Occurrence" reports pertaining to the Rancho Seco diesel generators	4001	