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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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BEFORE THE ATOMIC SAFETY AND LICENSING APPEAL BOARD

In the Matter of	)	
	)	
METROPOLITAN EDISON COMPANY	)	Docket No. 50-289
	)	(Restart)
(Three Mile Island Nuclear	)	
Station, Unit No. 1)	)	

LICENSEE'S RESPONSE TO APPEAL BOARD MEMORANDUM AND ORDER OF NOVEMBER 5, 1982

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November 22, 1982

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I. Introduction and Background

On November 5, 1982, the Atomic Safety and Licensing Appeal Board reviewing the exceptions of the parties to the Atomic Safety and Licensing Board's decision on plant design and procedures issues,1/ issued a Memorandum and Order setting forth its ". . . preliminary views and concerns regarding the posture of the record on one of the technical issues before us,

1/ Metropolitan Edison Company (Three Mile Island Nuclear Station, Unit 1), LBP-81-59, 14 N.R.C. 1211 (1981), ¶¶ 589-1225.

i.e., the issue of so-called 'feed and bleed' capability."2/ Memorandum and Order at 1.

The Appeal Board stated that its initial review of the record, although not yet complete, suggests that the Licensing Board's reliance upon feed-and-bleed cooling as a backup for emergency feedwater may have been misplaced. The Appeal Board observed that information submitted in recent Board Notifications tends to support this conclusion. *Id*. at 1, 4-6. The Appeal Board also stated its ". . . tentative view that the ability of the boiler-condenser mode of natural circulation to remove enough decay heat to prevent core damage also has not been adequately demonstrated on the record." *Id*. at 7.

The Appeal Board advanced two suggested changes at TMI-1 which ". . . would ensure core cooling by natural circulation during the interim before the emergency feedwater system is modified to full safety-grade status at the next refueling outage." *Id*. at 9-10. These new measures are: (1) installation of vents in the hot leg high points; and (2) assignment of an individual whose sole function would be to operate the EFW flow control valves manually in the event that the valve control system failed following the onset of an accident. *Id*. at 8-9.

2/ A significant dilemma posed by this Memorandum and Order is that, in fact, it raises, without sufficient explanation, a number of technical issues other than feed and bleed capability -- *e.g.*, the reliability of the TMI-1 emergency feedwater system and the reliability of boiler-condenser cooling.

Because these measures were not fully considered at the hearing, the Appeal Board invited the comments of the parties prior to the issuance of a final decision. The parties are invited particularly to express their views concerning the sufficiency of the Appeal Board's proposed requirements or, in the absence of the proposed changes, the need for reopening the record on feed and bleed cooling.3/ Id. at 10-11.

We appreciate the opportunity to present comments on the Board's preliminary views and concerns. In our comments below, Licensee addresses the two interim measures suggested by the Appeal Board and the alternative of reopening the record. Licensee also must comment, however, on the Appeal Board's potential acceptance of the Licensing Board's conclusion on emergency feedwater reliability, and on the Appeal Board's tentative views on the adequacy of the record to support the reliability of the boiler-condenser and feed-and-bleed cooling modes.

3/ It is not clear why the Appeal Board limited the potential reopening to the feed-and-bleed cooling issue. The adequacy of boiler-condenser cooling, also questioned in the Memorandum and Order, is an entirely separate matter which would not be resolved by an investigation into whether or not feed-and-bleed cooling is a reliable backup to emergency feedwater at TMI-1.

II. Emergency Feedwater Reliability

Feed-and-bleed cooling is not required except in the event of an extended loss of all main and emergency feedwater. Jones (on Board Question 6), ff. Tr. 4588, at 3; Tr. 5201 (Jones). Consequently, a critical threshold question, determinative of the reliance which need be placed on feed-and-bleed cooling, is the reliability of the emergency feedwater system at TMI-1. Licensee therefore begins its response to the Memorandum and Order with comments on the reliability of the emergency feedwater ("EFW") system at TMI-1.

The Appeal Board recites, without comment, the fact that "[a]fter exploring the matter at the restart hearing, the Licensing Board concluded that the emergency feedwater system at TMI-1 was not sufficiently reliable, by itself, to provide adequate protection of the public health and safety." Memorandum and Order at 2.4/ See LBP-81-59, supra, 14 N.R.C. at 1370 (¶ 1050).

Licensee strongly disagrees with this conclusion by the Licensing Board, which we have asked the Appeal Board to

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4/ Later, the Appeal Board does comment that it views the dependency of the emergency feedwater system upon the Integrated Control System to operate the EFW flow control valves to be a "deficiency." Memorandum and Order at 9. This concern, which we address in section III, infra, was not the basis for the Licensing Board's conclusions, however, which rested upon a quantified estimate of overall emergency feedwater system reliability.

modify. See Licensee's Brief in Opposition . . . , May 10, 1982, at 68-84. Licensee's modification request is supported by the NRC Staff. NRC Staff's Brief in Response to the Exceptions of Others . . . , May 20, 1982, at 35. If the Appeal Board modifies the Licensing Board's conclusion on emergency feedwater reliability, as requested by Licensee and the Staff, there would be no need to place reliance on feed-and-bleed cooling as a backup in order to adequately protect the health and safety of the public. See LBP-81-59, supra, 14 N.R.C. at 1372-73 (¶ 1057).

It may well be that at this juncture the Appeal Board has been misled by the Licensing Board's decision on emergency feedwater reliability. That decision did not turn on the question of whether or not the system is safety-grade. Indeed, the Licensing Board found that the TMI-1 EFW system will be safety-grade at restart for small-break loss-of-coolant accidents and for loss-of-main feedwater transients.5/ The Licensing Board, however, ignored this finding in concluding that the system is nevertheless unreliable. Further, the Licensing Board does not find the system adequately reliable

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5/ A fundamental holding by the Licensing Board, which has not been disturbed, established the scope of this proceeding to be limited to matters "having a reasonable nexus to the TMI-2 accident." This standard was proposed by the Staff and several intervenors, including UCS. Metropolitan Edison Company (Three Mile Island Nuclear Station, Unit 1), LBP-81-32, 14 N.R.C. 381, 394 (1981).

following the additional long-term modifications which will make the system safety-grade for all events. See LBP-81-59, supra, 14 N.R.C. at 1372-73 (¶¶ 1056-57). In short, the Licensing Board abandoned the Commission's design criteria in favor of its own risk assessment or probabilistic failure analysis. The Appeal Board, in contrast, appears to be satisfied that at the least the emergency feedwater system will be adequately reliable when modified to full safety-grade status at the next refueling outage.6/ Memorandum and Order at 9-10. Licensee is uncertain, at this point, of the extent to which this reflects Appeal Board disagreement with the basis for the Licensing Board's decision.

Recognizing that the Appeal Board has not yet completed its review of the record on this issue, and that the instant Memorandum and Order reflects only preliminary views not dispositive of a final decision on the matter,7/ Licensee sets forth here, in abbreviated fashion, its major arguments on the subject of emergency feedwater reliability.

It is Licensee's position that the Licensing Board improperly relied solely on a quantitative probabilistic

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6/ Licensee understands this to be the basis for the "interim" nature of the measures tentatively suggested by the Appeal Board.

7/ See, e.g., Memorandum and Order at 1, 3 n.5, 9, 10 n.22, characterizing the Appeal Board's present views as "tentative" and "subject to change in light of our further review."

analysis of the so-called "failure" on demand of the EFW system, while ignoring the reliable and substantial evidence showing that the system: is sufficiently reliable, complies with and in some cases exceeds the Staff's recommended requirements, and compares favorably with those in other operating reactors.8/ Wermeil and Curry, ff. Tr. 16,718, at 12; Tr. 17,017 (Wermeil). While probabilistic risk assessment, if properly performed, may be a valuable tool in comparing designs and identifying major contributors to system unavailability,9/ it should not be used in isolation to determine whether a plant is safe to operate. See LBP-81-59, supra, 14 N.R.C. at 1387 (¶ 1112), 1395-96 (¶ 1138).

Reliability assessments of the TMI-1 EFW system have been performed by Licensee, B&W and the Staff. See, generally, Wermeil and Curry, ff. Tr. 16,718. The analyses performed by Licensee and B&W identified the major contributors to EFW system unavailability (as the system existed in 1979), which led directly to several of the identified restart

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8/ Licensee has previously explained in detail the basis for its objection to the Licensing Board's reliance on this probabilistic analysis; we will not repeat in toto these arguments here, but we urge the Appeal Board to review this matter closely. See, generally, Licensee's Brief in Opposition to the Exceptions of Other Parties . . . (May 10, 1982), at 70-83.

9/ Indeed, Licensee and the Staff employed the technique for these purposes in connection with the TMI-1 EFW system. See Licensee's Brief in Opposition . . . , May 10, 1982, at 71-73, 79.

modifications. Id. at 3-5. The analysis performed by the Staff (directly and solely in response to the Licensing Board's persistent desire to obtain probabilistic reliability figures, Tr. 16,740 (Curry)), contained quantitative estimates of EFW system reliability, as that system existed in 1979, as it will exist at restart, and as it will exist once all of the long-term modifications are complete. The B&W analysis, while not assigning a numerical reliability value for the TMI-1 system, compared the 1979 system with those at Westinghouse and Combustion Engineering plants and found that the TMI-1 system fell in the mid-range. Tr. 5948, 5984-85 (Capodanno); Tr. 6157-59 (Wermeil). Similarly, the Staff's analysis included a comparison of the fully upgraded TMI-1 system with Westinghouse plants, which again showed TMI-1 to rank in the mid-range. Wermeil and Curry, ff. Tr. 16,718, at 35. Further, the Staff testified that, had the Staff's analysis used a more realistic mission success criterion,10/ the upgraded TMI-1 EFW

10/ The Staff's analysis utilized as its "mission success" criterion the ability to deliver EFW flow to the steam generators within five minutes. Wermeil and Curry, ff. Tr. 16,718, at 32-33. This criterion therefore gives virtually no credit for operator actions to correct an EFW system fault and is therefore largely a function of the probability of the system being in the proper configuration on demand. Id. at 33. The Staff believes that consideration of operator recovery actions would improve the reliability. Tr. 17,016 (Curry). Of course, it is recognized that the absence of EFW flow for five minutes does not result in core damage and that, since EFW is sprayed into the steam generator at a very high point, it immediately starts to cool the primary system when it is reestablished. Tr. 16,613-15 (Keaten).

system would have looked more similar to the numbers shown for the Westinghouse plants. Tr. 17,080 (Wermeil), accord, Tr. 17,068, 17,095 (Curry). Finally, the Staff testified that, based upon the reliability estimate of the TMI-1 EFW system combined with the reliability of other plant systems, the probability of core damage at TMI-1 is less than or certainly no greater than in all other operating plants. Tr. 17,089-92 (Curry).

Licensee believes it is important also to review here the improvements which have been made to the TMI-1 EFW system. Following the TMI-2 accident the Commission issued immediately effective orders to all B&W plants requiring completion of certain short-term actions including, inter alia, actions intended to upgrade the timeliness and reliability of the EFW systems and to control EFW independent of the ICS. See, e.g., 44 Fed. Reg. 27776, 27779 (1979) (orders issued to Duke Power Company and the Sacramento Municipal Utility District). No further actions with respect to the EFW systems were required at the other B&W plants before they were permitted to resume operation. Id. The same requirements were subsequently incorporated into the Commission's August 9, 1979 Order and Notice of Hearing for TMI-1 as short-term items 1(a) and 1(b). Metropolitan Edison Company (Three Mile Island Nuclear Station, Unit No. 1), CLI-79-8, 10 N.R.C. 141, 144 (1979). Additional short-term upgrades of the EFW system were required by

short-term order item 8 (i.e., the Category A recommendations set forth in Table B-1 of NUREG-0578). Id. at 145. Each of the modifications encompassed by these two short-term Order items will be fully implemented prior to restart.11/ LBP-81-59, supra, 14 N.R.C. at 1361-1363 (¶¶ 1028-1034); Staff Ex. 1 at C1-1 through C1-12 and C8-34 through C8-40. See also Licensee's Response to the Atomic Safety and Licensing Appeal Board's Order of July 14, 1982 (August 12, 1982) at 9-10. Additionally, a portion of the long-term upgrade (installation of cavitating venturis in each EFW line) has been completed. Id. at 11. Thus, it is clear that Licensee meets and exceeds12/ those short-term EFW requirements which were imposed on other B&W reactors following the TMI-2 accident.

The Licensing Board found that "[t]he EFW system will be safety grade at restart for small break LOCA and MFW transients, the ones of concern to this Board." LBP-81-59, supra, 14 N.R.C. at 1372 (¶ 1057). Consequently, Licensee must raise the question as to why compliance with all of this agency's

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11/ Briefly, these modifications consist of: (1) safety-grade (although only control grade was required in the short term), automatic initiation of the EFW pumps; (2) installation of redundant, safety-grade EFW flow indicators; (3) EFW flow control valves fail open on loss of instrument air; (4) EFW flow control independent of the ICS (see § III, infra); (5) provision of a redundant two hour air supply for the EFW control valves; (6) condensate storage tank low level alarms; and (7) redundant safety-grade steam generator level indication.

12/ See LBP-81-59, supra, 14 N.R.C. at 1363 (¶¶ 1035, 1036).

design requirements for these events is not adequate support for a finding that the TMI-1 EFW system is reliable. The Commission itself has recently stated, in the Proposed Policy Statement on Safety Goals for Nuclear Power Plants, that ". . . because of the present limitations in the state of the art of quantitatively estimating risks, the numerical guidelines are not substitutes for existing regulations." 47 Fed. Reg. at 7024 (1982). The Licensing Board has ignored these agency standards without good cause for doing so.

To summarize, Licensee's position is that the Licensing Board placed excessive reliance on the absolute value of a single numerical estimate based, in part, upon incorrect assumptions,13/ and has failed to take into account the other deterministic criteria long used by the Staff in evaluating the acceptability of the TMI-1 EFW system. Further, it has been shown that the TMI-1 EFW system will meet all current Staff design requirements and has currently exceeded the short-term EFW requirements imposed on other B&W operating reactors.

13/ The Licensing Board adopted a demand frequency of 0.3 per year due to loss of main feedwater. LBP-81-59, *supra*, 14 N.R.C. at 1370 (¶ 1050). This figure, however, is not based upon TMI-1 operational history, but that of five other B&W plants. As Licensee has previously pointed out, this factor is very plant specific; TMI-1's operating history shows that it has never experienced a loss of main feedwater or a total loss of feedwater. Keaten, ff. Tr. 16,612, at 9; Tr. 6175-76, 6219 (Wermeil). *See* Licensee's Brief in Opposition to Exceptions of Other Parties . . . (May 10, 1982), at 77-78.

Licensee therefore believes that the record adequately supports a finding that the TMI-1 EFW system is sufficiently reliable to allow restart.

III. Proposed Dedicated Operator for Manual Operation of EFW Control Valves

In order to overcome perceived deficiencies in the methods of controlling EFW flow to the steam generators, the Appeal Board has proposed that an individual be stationed at the EFW flow control valves in the event of system failure following an accident.14/ Memorandum and Order at 9. Licensee submits that the current provisions, described below, for assuring delivery of sufficient EFW flow are adequate pending completion of the long-term upgrade of the EFW system and, therefore, that the action recommended by the Appeal Board is not necessary.

The Appeal Board has noted correctly that the ICS is the normal method of controlling feedwater flow to the steam generators. However, in the event of an ICS failure to control EFW flow properly in the automatic mode, the following alternative methods are available.15/ If the control valves (EF-V30A,

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14/ Licensee understands the Appeal Board's recommendation as requiring an individual to be stationed at <u>these valves</u> at <u>all times</u> while the plant is operational, regardless of the then present operating mode (*i.e.*, during normal and transient conditions).

15/ Redundant, safety-grade EFW flow indicators for each steam generator have been installed at TMI-1. Licensee Ex. 15 at 6; *see also* Licensee's Response to the Atomic Safety and Licensing

(Continued Next Page)

B) have failed to open while the ICS is in the "AUTO" station, the control room operator is directed to attempt to control the valves through the ICS, but with the ICS in the "HAND" mode. Tr. 7104-05 (Broughton); Licensee Ex. 49, Att. I, at 6.0. Should this action fail to open the control valves, the ICS control can be completely overridden and the operator can manipulate the flow control valves from the new manual EFW control stations which have been installed in the control room. Tr. 7105 (Broughton); Licensee Ex. 49, Att. I, at 6.0; see also LBP-81-59, supra, 14 N.R.C. at 1362 (¶ 1031).

The Appeal Board has expressed its concern with the safety classification of the new manual control station. Memorandum and Order at 9. The manual control stations, which are totally separate from the ICS, will be powered from a class 1E power supply which receives multiple power feeds, and a single failure in the manual circuits will not result in the loss of a system function.16/ Licensee Ex. 1, §§ 2.1.1.7.3, 2.1.1.7.6; Licensee Ex. 15 at 7; see also LBP-81-59, supra, 14 N.R.C. at 1362 (¶ 1031).

(Continued)

Appeal Board's Order of July 14, 1982 (August 12, 1982), at 9. Loss of EFW flow can therefore be readily determined by the control room operators.

16/ Even if the power source were to fail, the control valves would still be capable of providing flow, as the valves would fail half open. See pp. 14-15, infra.

While Licensee believes that the above-described provisions for manually controlling EFW flow are more than sufficient, additional assurance is provided by the required actions in the station emergency operating procedures: i.e., it is required that the control room operator dispatch an auxiliary operator to the flow control valves for any EFW pump auto-start condition.17/ See, e.g., Licensee Ex. 49 at 2.0 (Step B.3), 6.0; Licensee Ex. 48 at 10.0 (step 14), 30.0. Thus, in the unlikely event that the EFW control valves cannot be opened from the control room, an auxiliary operator, who is in communication with the control room operators, will be in position to open the valves manually by taking local handwheel control of the valves.18/ Id.; see also Tr. 7105 (Broughton).

Further, it must be noted, with respect to the ability to deliver EFW flow to the steam generators, that neither a loss of control power nor a loss of instrument air will totally

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17/ All such auto-start conditions are annunciated in the control room. See LBP-81-59, supra, 14 N.R.C. at 1361 (¶ 1028)

18/ In its July 14, 1982 Order, the Appeal Board expressed its concern with the time available to initiate EFW or HPI following an RCP suction break. Slip op. at 10. As the Staff has explained (Jensen August 6, 1982 Affidavit at 2-5; App. Tr. 303-304 (Jensen)), HPI initiation for this break within 20 minutes would provide sufficient core cooling, with somehwat less time available if only emergency feedwater is to be relied upon. This time period is not of concern here because, upon information and belief, an auxiliary operator can reach the control valves within 5 minutes after leaving the control room.

defeat the flow control system. The failure mode of the control valves has been modified such that, upon a loss of instrument air, the valves will fail open. A loss of control power will result in the valves failing half open. Tr. 5675 (Capodanno); Licensee Ex. 15 at 6; LBP-81-59, supra, 14 N.R.C. at 1362 (¶ 1030). Thus, at least partial flow would be provided to the steam generators until repairs could be effected.

In sum, based on the foregoing, Licensee contends that sufficient assurance presently exists that manual control of EFW flow can be taken in a timely manner and, therefore, that there is no need to station a dedicated operator at the flow control valves on a full-time basis.

IV. Boiler-Condenser Cooling

Presumably addressing a small-break LOCA event,19/ the Appeal Board observes that

> If there is substantial steam voiding at the high point vents of the reactor coolant system, however, cooling would depend on the establishment of a type of natural circulation referred to as the boiler-condenser mode. It is our tentative view that the

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19/ The discussion begins with a reference to a main feedwater transient. Memorandum and Order at 6. Voiding would not occur, however, except in the event of a loss of all feedwater, or in the event of a small break LOCA of the appropriate size. The boiler-condenser process, of course, requires the availability of feedwater. See Keaten and Jones, ff. Tr. 4588, at 7.

> ability of the boiler-condenser mode of natural circulation to remove enough decay heat to prevent core damage also has not been adequately demomstrated on the record. Therefore, at the present time, we do not consider the boiler-condenser mode a viable method of removing decay heat.

Memorandum and Order at 7 (footnote omitted).

In the boiler-condenser process, which assumes continued availability of main or emergency feedwater, steam generated by core decay heat rises through the hot leg and is condensed in the steam generator. The condensed primary coolant then returns to the core by gravity flow through the cold legs to provide further heat removal. Keaten and Jones, ff. Tr. 4588, at 7; Tr. 4852-54 (Jones); Jensen (Natural Circulation), ff. Tr. 4913, at 6.

The Licensing Board correctly found that reactor coolant system breaks of 0.005 ft^2 or less do not involve voiding. For breaks larger than 0.02 ft^2, secondary heat removal is not required since the energy discharged through the break is sufficient to prevent a pressure increase, whether or not forced or natural circulation occurs. LBP-81-59, supra, 14 N.R.C. at 1227 (¶ 607). It is for breaks in between these sizes, then, that boiler-condenser cooling may be required.

The basis for the Appeal Board's tentative disagreement with the Licensing Board's findings on boiler-condenser cooling is expressed in footnote 15 of the Memorandum and Order. There the Appeal Board cites testimony by a Licensee witness for the

proposition that this cooling mode has been predicted by computer modeling, but that no tests had been performed to demonstrate its viability.

First we note that the boiler-condenser cooing mode was endorsed not only by witnesses fom Licensee, but also by the NRC Staff's witness. See, e.g., Jensen, supra. In fact, no witness presented testimony questioning the efficacy of boiler-condenser cooling at TMI-1.

Second, there is no basis for the Appeal Board's apparent unwillingness to rely upon the Babcock & Wilcox computer codes used to model small-break loss-of-coolant accidents. Computer modeling is an accepted and required technique for demonstrating compliance with NRC regulations governing plant design. See 10 C.F.R. § 50.46 and Appendix K to 10 C.F.R. Part 50.

It is undisputed that the B&W ECCS evaluation model is an NRC approved code under Appendix K. It is this code, with additional noding and with modified assumptions in order to produce a best estimate prediction for the development of operator guidelines, which was used to predict the boiler-condenser cooling mode. LBP-81-59, supra, 14 N.R.C. at 1329 (¶ 925), 1332 (¶ 937); see generally, Jones and Broughton, ff. Tr. 5038. Similarly, it is undisputed that the results of the B&W Appendix K small break analyses meet the requirements of 10 C.F.R. § 50.46, and that multiple failures must occur before the 10 C.F.R. § 50.46 limits are challenged in the

post-accident analyses. LBP-81-59, <u>supra</u>, 14 N.R.C. at 1332-33 (¶ 941), 1335 (¶ 952), <u>as modified by</u> Licensing Board Memorandum and Order Modifying Partial Initial Decision of December 14, 1981 (January 26, 1982). The Staff confirmed this to the Appeal Board once again at oral argument. App. Tr. 284 (Sheron). Commission approval of the B&W model presumably includes the provision in section II.4 of Appendix K, that "[t]o the extent practicable, predictions of the evaluation model, or portions thereof, shall be compared with applicable experimental information."

This Licensee, like other licensees with operating reactors of B&W design, is in compliance with Commission regulations with respect to the design of emergency systems required to cool the core in the face of loss-of-coolant accidents, and with respect to the evaluation model used to demonstrate the adequacy of the performance of those systems. We therefore question that the Appeal Board may disregard the B&W model as an inadequate demonstration of the viability of boiler-condenser cooling in B&W plants such as TMI-1.20/ <u>See, e.g.</u>, <u>Union of Concerned Scientists v. A.E.C.</u>, 499 F.2d 1069, 1088-89 (D.C. Cir. 1974) (holding that an attempt to

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20/ While additional small-break LOCA analyses have been performed at NRC direction following the TMI-2 accident, these analyses were for the purpose of developing improved operating guidelines, and not to confirm the adequacy of the plant design. <u>See</u> LBP-81-59, <u>supra</u>, 14 N.R.C. at 1329-30 (¶ 926).

demonstrate the invalidity of the Commission approved evaluation model constituted an impermissible attack on the acceptance criteria themselves); Commonwealth Edison Company (Zion Station, Units 1 and 2), ALAB-226, 8 A.E.C. 381, 403-404 (1974).

Intervenor UCS long ago abandoned whatever interest it expressed at the hearing in questioning the adequacy of the B&W modeling of small-break LOCAs. UCS filed no proposed findings of fact with the Licensing Board, and no exceptions with this Board, on the adequacy of Licensee's small-break LOCA analyses and the operator guidance which is based on those analyses.21/ See generally LBP-81-59, supra, 14 N.R.C. at 1328-1340 ("Additional LOCA Analysis"), as modified.

Nevertheless, the record compiled in response to the Licensing Board's interest in the small break LOCA analyses is extensive. See, e.g., Licensee Exs. 3 to 13 and Board Ex. 4. Since the parties were not called upon, by exceptions, to brief this record to the Appeal Board, we call attention to Licensee Proposed Findings of Fact 333-359, June 12, 1981. When the Appeal Board completes its review of this lengthy record, we are confident that it will find overwhelming support for the Licensing Board's reliance on the B&W analyses.

21/ See Licensee's Brief in Opposition . . . , May 10, 1982, at 61-62, with respect to the UCS complaint that there will not be an adequate condensing surface for boiler-condenser cooling.

For example, on the specific question of the boiler-condenser process which follows the interruption of single-phase natural circulation for some breaks, Licensee witness Jones described model improvements, in the form of noding changes for some analyses (for the 180 degree bend in the top of the hot leg and down into the upper plenum of the steam generator), made after the TMI-2 accident in order to predict more accurately the potential for interruption of natural circulation. Tr. 5161-62 (Jones). See also Tr. 5545 (Jensen); Licensee Ex. 5, § 6.2.4.2. These analyses demonstrate the viability of boiler-condenser cooling. See Tr. 5094-97 (Jones) and Licensee Ex. 5.

While it is true, as the Appeal Board observed,22/ that tests have not been performed physically to demonstrate the boiler-condenser cooling mode in a B&W plant, there is evidence from the TMI-2 accident itself that steam condensation and heat removal via the steam generators occurred. Tr. 4627-30, 4685-86 (Jones). In addition, the Staff testified that the basic phenomenon has been tested in the "U-tube" steam generator configuration:

> To date, experimental evidence has been obtained that two-phase and reflux-boiling modes of natural circulation can adequately remove decay heat from the core. This data has been obtained at both the Semiscale and LOFT facilities in the U.S. and in the PKL

22/ Memorandum and Order at 7, n.15.

> facility in Germany. For example, test results from LOFT test L3-2 (conducted in February 1980) indicated that the steam generator transitioned from liquid natural circulation to two-phase natural circulation, and possibly to reflux-boiling and then back again to liquid natural circulation with no evidence of instability. During another LOFT test conducted in September 1980 (L3-5) the pressure vessel liquid level was lowered far below the hot leg and flow conditions in the hot leg were monitored to further study the question of reflux-boiling. All flow measurements showed that even for this case, flow continued in the positive direction and adequately provided core cooling. For these tests, staff codes predicted the major phenomena in the proper sequence. Each of the facilities mentioned above have recirculation ("U"-tube) type steam generators. An integral systems test facility utilizing a OTSG does not presently exist.

Ross and Capra, ff. Tr. 15,806, at 34-35. See also Tr. 5223-24 (Jones).

Furthermore, the Licensing Board specifically inquired into the potential need for additional experimental verification to confirm the analyses of boiler-condenser cooling. Licensee witness Jones explained that such experiments would not be needed to confirm that the basic phenomenon works, but rather that experiments might be used to confirm the accuracy of the code in predicting the amount of heat transfer for a given system heat condition. On that score, the model compares well with the classical heat transfer models, and for the small breaks where boiler-condenser cooling comes into play, there are large margins to core uncovery, so that any uncertainties

in the heat transfer would not be expected to result in core uncovery. Mr. Jones testified that, in fact, the model-predicted heat transfer capability could be wrong by a factor of ten without reaching an uncovered core situation. Tr. 5293-95 (Jones). <u>See also</u> Jones and Broughton (Board Question on UCS 8), ff. Tr. 5039, at 16-17.

Citing correspondence from this year, the Appeal Board noted ". . . that the Advisory Committee on Reactor Safeguards and the staff have subsequently expressed concern for the modeling of the dynamic thermal hydraulic behavior of B&W plants during small break loss of coolant accidents." Memorandum and Order at 7, n.15. It is not necessary to go outside the record, however, to ascertain the NRC Staff's position on these matters, which has been clearly and unequivocally expressed to the Licensing Board and to this Board.

The Staff has found that TMI-1 continues to be in compliance with 10 C.F.R. § 50.46. Jensen, ff. Tr. 15,808, at 3; Tr. 5023 (Jensen). A principal finding of the Staff's generic review of the B&W analyses after the TMI-2 accident is that the original LOCA analyses for TMI-1 remain valid. Staff Ex. 1 at C1-13. The Staff provided for the record documentation on the results of its review of the B&W small-break LOCA analyses performed in response to NRC direction and requests following the TMI-2 accident, including the Staff's own audit calculations used in the review. <u>See</u> Board Ex. 4 (NUREG-0565). The Staff's main 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.

Board Ex. 4 at 4-25.

The Staff's safety evaluations for TMI-1, prepared specifically in response to the recommendations embodied in the Commission's Order and Notice of Hearing, document the Staff's conclusion that Licensee has satisfactorily completed all short-term requirements, and made reasonable progress toward the completion of all long-term requirements, associated with analyses of small breaks and the development of operator guidance. See Staff Ex. 1 at C1-12 to C1-16, C2-16, C8-48 and D2-1; Staff Ex. 14 at 43-44, 50.

The Staff has recommended that B&W licensees should demonstrate experimentally the various modes of two-phase natural circulation, and should provide verification of their analysis models to predict two-phase natural circulation by comparison of the analytical model results to appropriate integral systems tests.23/ This recommendation is being

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23/ The models have been compared against separate effects tests. App. Tr. 292-93 (Sheron).

pursued as a part of the TMI Action Plan. Nowhere, however, is there testimony by any Staff witness, or by any one else for that matter, suggesting that the performance of such tests would be necessary to provide reasonable assurance that TMI-1 can be operated safely in the short term. See, e.g., Ross and Capra, ff. Tr. 15,806, at 34-35. The fact that the Staff finds it desirable to seek, in the long term, additional confirmation of the results of its review of the B&W analyses does not undermine the fact that the Staff has confidence in the B&W model (as well as its own) and has identified no immediate safety concerns. See LBP-81-59, supra, 14 N.R.C. at 1337-38 (¶ 959); Tr. 5584 (Jensen).

The Staff's position was stated again at oral argument before this Board:

> MR. SHERON: My name is Brian Sheron. I'm Chief of the Reactor Systems Branch.
>
> The reason we are requesting the confirmatory experimental data is basically one that we have looked at the models, we do believe that we find the plant in continued conformance with the Regulations 50.46 and Appendix K.
>
> JUDGE BUCK: Excuse me. Do you have any problems with the models themselves? Do you think they're satisfactory? Do they need correcting or anything of that nature?
>
> MR. SHERON: We've looked at the models, we've looked at the verification that has been provided to date by the Licensees, and based on that information provided we have sufficient assurance that the plant can be operated safely. However, there is longer-term confirmation that we believe is

> needed in order to, as I would say, confirm this assurance that we have right now.

App. Tr. 284. Dr. Sheron went on to state that adequacy of decay heat removal is not the item of concern, id. at 288, and that the interruption [in natural circulation] itself is not going to lead directly to an unacceptable situation. Id. at 292. Dr. Sheron also reported that the Staff has considered and attempted to bound the consequences of being wrong about the B&W analyses, and has concluded that there are no unacceptable results from the standpoint of providing adequate core cooling. Id. at 298.

Consequently, there is no basis for the Appeal Board's tentative view that the boiler-condenser mode is not a viable method of removing decay heat. This cooling mode has been shown to work in U-tube configurations, the basic phenomena are well understood and appear to have occurred with the B&W design during the TMI-2 accident, and the mode has been predicted by NRC approved evaluation models. The modeling concerns raised by the Staff and ACRS are not of immediate concern and do not require resolution in order to provide reasonable assurance that TMI-1 can be operated safely.24/

24/ The Staff's proposals for integral systems testing would involve a several year test program. NRC Staff's Response to Appeal Board's Order of July 14, 1982 (August 9, 1982), Jensen Affidavit at 12.

V. Feed-and-Bleed Cooling

The Appeal Board has voiced its present view that the viability of feed and bleed has been called into question by the tests recently performed at the Semiscale facility.25/ Memorandum and Order at 4, 6. Apart from the tests, however, the Appeal Board stated that it was ". . . inclined toward the view that there is insufficient evidence of record to support the Licensing Board's conclusion that feed and bleed is a viable means of removing decay heat from the reactor core at TMI-1." Id. at 6 (footnote omitted). Licensee will address first the Appeal Board's concern with the adequacy of the record. We will then comment on the Semiscale tests and their implications for TMI-1.

A. Adequacy of the Record

The Licensing Board found, with respect to feed-and-bleed cooling, that "[s]ince only safety-grade equipment is involved and procedures and training have been directed to the TMI-2 accident, a high degree of reliability is expected." LBP-81-59, supra, 14 N.R.C. at 1370-71 (¶ 1051). Licensee

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25/ See Board Notifications BN-82-93 (Sept. 14, 1982) and BN-82-107 (Oct. 22, 1982). At the time Licensee filed its reply (Oct. 24, 1982) to UCS on BN-82-93, Licensee had not reviewed BN-82-107. Consequently, we were unaware that tests were performed with other than low head HPI pumps.

expects that the Appeal Board, upon completion of its review of the record, will find reliable evidence to support such a finding.

Initiation of the feed-and-bleed cooling mode is a very simple operation. If neither main nor emergency feedwater is available, the operator will initiate and maintain full high pressure injection until feedwater is restored. The operator can open the PORV and its block valve, or allow the code safety valves to open to provide a flow path. Keaten et al., ff. Tr. 16,552, at 10.

Once initiated, the feed-and-bleed cooling mode will automatically continue without need for additional short-term operator actions. In the long term, the operator must transfer the suction of the high pressure injection pumps from the borated water storage tank to the containment building sump via the low pressure injection pumps. If the Engineered Safety Features Actuation System ("ESFAS") has automatically initiated, this transfer requires opening four valves and closing four valves, all of which can be done at the main control console. If ESFAS has not automatically initiated, the Low Pressure Injection ("LPI") pumps must be started manually, but this also can be accomplished from the main control console. Keaten et al., ff. Tr. 16,552, at 10.

Termination of the feed-and-bleed cooling mode is also very simple. Once the appropriate criteria, established in

station procedures (Licensee Exs. 48 and 51), are met, the HPI discharge valves are throttled and eventually the HPI pumps are turned off. These actions are also performed from the main control console. Such throttling and/or termination of high pressure injection, however, is only permissible when specific criteria regarding reactor coolant system conditions are met. Keaten et al., ff. Tr. 16,552, at 11.

It should be noted that the simple actions associated with initiation, continuation and termination of feed-and-bleed cooling would be performed by an operator assigned to this portion of the control panel. Any parallel actions being taken in an attempt to restore main or emergency feedwater would be taken by a different operator assigned to the feedwater control panel. The TMI-1 Technical Specifications require that two licensed reactor operators be in the control room during startup, shutdown, and recovery from a reactor trip. The normal control room practice is that immediately upon reactor trip one operator goes to the portion of the console from which HPI and LPI are controlled, and the other operator goes to the feedwater control portion of the panel. This allows actions to be carried out in parallel under the supervision of the senior watchstanders. Keaten et al., ff. Tr. 16,552, at 11.

While the PORV may be used as the fluid discharge path from the reactor coolant system, feed and bleed can be accomplished with only safety-grade systems and components -- i.e.,

the pressurizer safety valve(s) in conjunction with the borated water storage tank, high pressure injection, containment and low pressure injection. Keaten and Jones, ff. Tr. 4588, at 12.26/ The only action required of the PORV and safety valves in feed-and-bleed cooling is that one or more of these valves open to provide a fluid discharge path.27/ Jones (on Board Question 6), ff. Tr. 4588, at 1-2. Further, based on the results of the EPRI Safety and Relief Valve Test Program, and in particular of steam and water tests for safety valves of the model present at TMI-1, the inlet piping configuration at TMI-1 is being modified prior to restart to the configuration for which the safety valves exhibited stable performance for all fluid inlet conditions (including two-phase) during the EPRI tests. See Licensee's Response to the Atomic Safety and Licensing Appeal Board's Order of July 14, 1982 (August 12, 1982), at 7-8.

There is also some experience which shows that feed-and-bleed operation can provide adequate core cooling. During the

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26/ The individual systems and components required for feed-and-bleed cooling (e.g., HPI, LPI and the safety valves) are routinely operated and/or tested to assure their functionability. Jones (on Board Question 6), ff. Tr. 4588, at 4; Tr. 4886-87 (Jones).

27/ The effect in this case is to create a loss of coolant accident in the range where energy removal through the break matches decay heat. Events in this range have been fully analyzed. See generally, Jones and Broughton, ff. Tr. 5038.

February 26, 1980 event at Crystal River 3, the HPI system injected water into the primary system and fluid was discharged initially by the PORV and then by a safety valve. Therefore, the incident was a demonstration of the operability of feed-and-bleed cooling. It should also be noted that during a portion of the Crystal River transient, secondary side cooling was significantly reduced or non-existent. Throughout the scenario, however, the core was adequately cooled. Jones, ff. Tr. 4588, at 3, 4. See also Jensen (Natural Circulation), ff. Tr. 4913, at 9, 10.

Finally, the record includes the B&W analysis, performed with its NRC approved evaluation model, of a loss of all feedwater without a LOCA. The analysis further assumed, inter alia, that the PORV does not open and that the pressurizer safety valves open for decay heat removal. The analysis shows that establishment of emergency feedwater or initiation of high pressure injection within twenty minutes assures adequate core cooling. See Jones and Broughton, ff. Tr. 5038, at 5 and 13 (Table 2); Licensee Ex. 9 and oral summary at Tr. 5064-73 (Jones). This evaluation, which we discuss further below in comments on the Semiscale tests, is essentially a confirmation or demonstration that feed-and-bleed cooling is a viable mode of core cooling at TMI-1, if it is ever needed. Tr. 5066 (Jones). For the reasons set forth in our comments above on boiler-condenser cooling, we submit that the Appeal Board

should rely on the results of the B&W analyses, without any additional testing at this time.

B. The Semiscale Tests

While acknowledging the tests' limitations and the arguments on inapplicability advanced by Licensee and the Staff, the Appeal Board nevertheless has expressed concern that the viability of feed-and-bleed cooling has been called into question by tests recently performed at the Semiscale facility. Memorandum and Order at 4-6. The tests' results have been documented in a report entitled "Analysis of Primary Feed and Bleed Cooling in PWR Systems," EGG-SEMI-6022, September 1982 ("EGG"). Licensee believes that the tests done at Semiscale do not call into question the viability of feed-and-bleed cooling at TMI-1. To understand this position, Licensee includes in these comments the following examination of the EG&G test report.

The approach taken by EG&G to examine feed-and-bleed cooling was to first identify the key parameters which influence feed-and-bleed performance. These were identified to be core decay heat, HPI injection capacity, PORV energy removal rate, and PORV mass removal rate. EGG at iv. Next they defined operating maps which represented anticipated regions of acceptable feed-and-bleed operation. From these operating maps, test boundary conditions were developed for the

experimental facility at Semiscale which were expected to result in a stable system operating condition. Two experiments were then run to develop data for subsequent computer code assessment. EGG at iv.

EG&G formulated upper and lower bound functional conditions which defined the boundaries of the region of stable feed-and-bleed operation at a given decay heat level. The lower bound was determined by identifying the pressure at which sufficient fluid mass and enthalpy are relieved from the system to remove the core decay heat. The upper bound was determined by identifying the pressure at which the cooling water injection capacity is equal to the average coolant mass removal rate. At pressures above the upper limit a coolant mass deficit exists. EGG at 3.

The purpose of the first test (S-SR-1) was to "determine the feasibility of operating within the predicted [feed-and-bleed] operating band and to examine the thermal-hydraulic effects accompanying rapid depressurizations to lower operating pressures when significant primary mass depletion had occurred." EGG at 22. Unfortunately, "operational problems with uncontrolled coolant leakage from the system precluded the use of results from test S-SR-1 for direct interpretation as to the viability of feed and bleed cooling." Id. The test did, however, provide data that could assist in the identification of the phenomena that influence feed-and-bleed operation.

Test S-SR-1 attempted to substantiate the viability of feed-and-bleed cooling by establishing a system pressure which would correspond to the upper bound point on the feed-and-bleed operating map. This pressure was determined from the operating map to be 15.17 MPa. The PORV was thus cycled for most of the test as required to maintain a pressure of 15.17 MPa. EGG at 26. This pressure resulted in a steady HPI injection rate and an average PORV mass flow as defined on the operating map. At this condition a system mass balance condition should have existed and the PORV energy removal rate would have exceeded the core power generation. In this way the core would be cooled and system inventory would be maintained. EGG at 26.

The desired test condition was not met, however, because of a large uncontrolled leakage which developed from another part of the test system. This leakage altered the balance of masses. Since the HPI mass flow was equal to the PORV flow at this pressure, the additional leakage mass flow invalidated the operating map conditions and resulted in the depletion of vessel inventory. This was a failure of the experiment, not a failure of feed-and-bleed cooling. UCS ignores this fact.

It also precluded any possibility of meeting the first test objective of demonstrating operation within the predicted operating band, since the test system was never actually in the operating band. Later in the test, when pressure was reduced, the HPI flow increased sufficiently to recover the deficit

mass, but not before core uncovery had already taken place. If the pressure had been reduced sooner, core uncovery would not have occurred at all because the HPI flow at reduced pressure was more than sufficient to balance the flow from both the PORV and the uncontrolled system leakage. EGG at Fig. 12. This was the second test objective. It showed that core uncovery cannot occur if the HPI pumps have a sufficient flow capability relative to pressurizer valve outlet mass flow rates. There were no previously unrecognized thermal-hydraulic phenomena which prevented this from happening. As soon as the HPI injection system was capable of delivering enough mass to exceed that which was being lost through the PORV and the leakage, the system inventory decrease ceased and the vessel began to refill rapidly. This would have occurred sooner if pressure had been reduced sooner, and the core would then not have uncovered.

The second test objective was met successfully, however. The statement made by UCS that "the PORV setpoint was adjusted so that it opened at the safety valve setpoint" is not correct.28/ The setpoint of 15.17 MPa (about 2200 psig) was chosen because it happened to be the operating map upper bound pressure setpoint for test S-SR-1. EGG at 26. UCS also

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28/ See Union of Concerned Scientists' Reply to Appeal Board Order of October 15, 1982 (October 29, 1982), at 3, n.2.

appears to have omitted any statement that the core uncovery was caused by the uncontrolled system leakage, and implies that it was caused by a failure of feed and bleed. This also is incorrect. The test showed that feed and bleed will prevent core uncovery if sufficient mass is injected relative to that which is leaving the vessel.

Semiscale test S-SR-2 was performed with low head HPI injection. The objectives of the test were again to "determine the feasibility of operating within the predicted operating band" and to "determine the feasibility of initiating a steady-state feed and bleed operation by depressurizing from a representative operating pressure." EGG at 34. The test attempted to achieve a stable feed-and-bleed condition based upon a predicted steam performance for the Semiscale PORV. This predicted performance was a calculation of the mass flow rate out of the PORV based upon steam conditions. This formed the basis for the operating map. Once again they set up the test at the upper bound condition. Id. During the test they were unable to achieve stable feed and bleed at this condition because the PORV mass discharge rate was greater than what was predicted. EGG at 38. During the test, two phase flow conditions were present in the pressurizer and at the PORV. The prediction was based upon pure steam. This resulted in a higher mass flow out of the PORV. Since the operating map condition was no longer being met, there was a mass deficit and

a resultant vessel inventory decrease. Later in the test, the PORV was latched open and system pressure decreased to saturation conditions. EGG at 44. At this new pressure HPI flow would have balanced PORV flow if the HPI injection in the test had sufficient head capacity or if the PORV flow were slightly less. The S-SR-2 HPI injection system did not have quite enough flow capability at this pressure. The core was being cooled, but a small mass deficit still existed and the core eventually uncovered in the test. EGG at 48. Small differences in the Semiscale PORV or HPI simulation could have resulted in a favorable mass balance. EG&G noted in its report that this "deficit in the mass balance was small enough that it was well within the range of experimental uncertainties and therefore makes no direct statement as to the viability of primary feed and bleed." EGG at v-vi. There are no negative implications for TMI-1 in this test because of the high capacity, high head pump capability at TMI-1. The Semiscale tests have established that if a mass imbalance is present, the system inventory will change in such a way that it will either increase or decrease depending upon the direction of imbalance. No new thermal-hydraulic phenomena were discovered which would preclude this from happening. It can thus be said with confidence that if a higher capacity HPI system capability had been modeled in S-SR-2, the core uncovery would not have taken place.

The importance of the S-SR-1 and S-SR-2 test series was in the fact that no thermal-hydraulic phenomena were discovered which would <u>prevent</u> viable feed-and-bleed operation, and that existing reactor systems can be evaluated on a plant specific basis using presently available computer codes to demonstrate whether or not feed and bleed is a viable cooling mode for a given plant under a given set of conditions. To determine whether feed-and-bleed would succeed on a Babcock & Wilcox plant, plant specific analysis was recommended by EG&G.

Licensee Exhibit 9, "System Response to Total Loss of Heat Sink," represents a plant specific analysis of an event similar to the full scale plant analysis in the EG&G report. <u>Compare</u> Licensee Ex. 9 <u>and</u> EGG, Section 6. Plant specific values of parameters such as HPI discharge pressure and flow data, safety valve capacity and decay heat levels were used. For example, no credit for PORV actuation is taken in Licensee Exhibit 9, whereas full credit was taken for the RESAR plant PORVs in order to reduce system pressure below the shut off head of the RESAR High Pressure Injection system.

A careful comparison of each analysis will reveal that equivalent system responses were achieved for each scenario. In the period prior to the time when injection flow rate equaled core boil off rate, a mass deficit existed between system relief rate and injection rate, while system heat rejection equaled core boil off. In each case, at the moment

when core boil off equaled system injection sufficient inventory existed covering the core such that excessive heatup had not occurred. Beyond that point, recovery begins. With continually decreasing decay heat levels, the subsequent injection rate would exceed core boil off and system inventory would be restored.29/

Thus the results of experiments performed at Semiscale, although not representative of TMI-1 plant specific parameters, do not identify new phenomena relative to the viability of feed and bleed in large PWRs generally, or TMI-1 specifically. Rather, the results at Semiscale demonstrate that the operational conditions necessary for feed and bleed to succeed are understood and, with the aid of full scale plant simulation, satisfactory assessments of plant performance can be made.

The "sensitivity of feed and bleed viability to a number of different, plant specific variables" which is discussed by UCS30/ is not new information. Semiscale results support the conclusion that feed and bleed is a mass and energy balance condition. The balance is "sensitive" to changes in its key parameters, which are mass and enthalpy in and out of the

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29/ "Subsequent to the Semiscale test, the Staff has conducted calculations for TMI-1 using the RELAP-5 code which demonstrate that the core would remain cool and covered." NRC Staff Response to Appeal Board Order of October 15, 1982 (October 25, 1982), at 8.

30/ UCS Reply, supra, at 2.

system and decay heat production. Semiscale could not achieve this balance for the various reasons cited earlier.

VI. Proposed Addition of Hot Leg High-Point Vents

The Appeal Board has suggested that the vents ultimately planned for the high point of the reactor coolant system hot legs should be installed prior to restart at TMI-1 in order to remove steam and to help reestablish natural circulation for those small breaks which result in the interruption of liquid natural circulation due to collection of steam bubbles in the U-bend of the hot legs. Memorandum and Order at 7-8.

It is beyond dispute that vents will be installed at the hot leg high points at TMI-1. The Appeal Board's proposal raises two questions, however: (1) the schedule for installing the vents; and (2) the purpose to be served by the vents once installed.

First, it is Licensee's position that there will not be a need to vent trapped steam in the hot legs because the boiler-condenser process, discussed above (IV, supra), will provide decay heat removal while effective core cooling is maintained by the HPI-injected fluid covering the core. See Licensee's Brief in Opposition . . . , May 10, 1982, at 59-60.

Second, the record at best casts doubt on the utility of these vents to remove steam and re-establish natural circulation.31/ In the cestimony of Licensee witness Jones

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31/ We note that the Licensing Board relied upon testimony by Staff witness Jensen for the proposition that high point vents

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cited by the Appeal Board (Memorandum and Order at 8), Mr. Jones was responding to cross-examination on the sequence of events during the TMI-2 accident and on the operators' purported use of reactor coolant pumps to establish natural circulation. Responding to questions on other means to restore natural circulation in that situation, Mr. Jones stated that the hot leg high point vents, to be installed at TMI-1, could be used. Mr. Jones was addressing the TMI-2 accident, however, and he assumed the presence of hydrogen as well as steam. See Tr. 4617, 4623-24 (Jones). The Staff reported at oral argument that calculations performed at Los Alamos indicate that the vents may not help restore natural circulation interrupted by steam voids. App. Tr. 291-92 (Sheron).

The Appeal Board implies that the schedule and purpose for these vents are of Licensee's own making. See Memorandum and Order at 8, n.16. To the contrary, Licensee simply reported the schedule and purpose for these vents established by the Commission itself.

The Commission initially set out requirements for the installation of high-point vents in its Proposed Rule, "Interim

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would be useful in reestablishing natural circulation following a small break LOCA. See LBP-81-59, supra, 14 N.R.C. at 1230 (¶ 620), citing Jensen (Natural Circulation), ff. Tr. 4913, at 10. Mr. Jensen was addressing the venting of non-condensible gases, however, and not steam alone.

Requirements Related to Hydrogen Control and Certain Degraded Core Considerations," 45 Fed. Reg. 65466 (1980), as corrected by 45 Fed. Reg. 70473 (1980). In discussing this proposal, the Commission stated

> There is a concern that the accumulation of pockets of noncondensible gases in the primary system of a reactor may interfere with . . . natural circulation . . . [or] may interfere with pump operation. It has therefore been concluded that under certain circumstances it would be desirable to have provisions for venting noncondensible gases from high points in the primary system.

Id. at 65468 (emphasis added).

This same requirement was included in the proposed "TMI lessons learned" rule for operating license applicants. *See* Proposed Rule, "Licensing Requirements for Pending Operating License Applications," 46 Fed. Reg. 26491, 26497 (1981). In promulgating its final rule, "Interim Requirements Related to Hydrogen Control," the Commission noted that the high-point vent provision had been included in the proposed OL rule, but concluded that the vent requirement "was felt to be primarily hydrogen related and thus more appropriately included in this Interim Rule."32/ 46 Fed. Reg. 58484 (1981) (emphasis added).

32/ Similarly, the NUREG-0737 Item II.B.1 clarification of this issue states that "[t]he important safety function enhanced by this venting capability is core cooling. For events beyond the present design basis, this venting capability will substantially increase the plant's ability to deal with large quantities of noncondensible gas which could interfere with core cooling." NUREG-0737 at 3-56 (emphasis added). The use of the vents for design basis events is recognized; howe-

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Nowhere in the proposed or final rules is the use of vents for relieving steam discussed.

The Commission's hydrogen control regulations also dictate the schedule for installing the high-point vents at all light water reactors. The regulation, 10 C.F.R. § 50.44(c)(3)(iii), requires installation of these vents "by the end of the first scheduled outage <u>beginning</u> <u>after</u> July 1, 1982 and of sufficient duration to permit required modifications . . ." (emphasis added). In conformance with this requirement, Licensee will install the hot leg and vessel head vents during the next refueling outage after restart.33/

The Commission has spoken through its adoption of 10 C.F.R. § 50.44(c)(3)(iii), which applies to all reactors. The Commission has not required the installation of these vents as a condition of continued operation at any other plant, and did not require it as a short-term condition for the resumption of operation at the other B&W plants shut down following the accident at TMI-2. The Appeal Board, Licensee submits, is

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ver, such usage "must not result im a violation of 10 CFR 50.44 or 10 CFR 50.46." In this regard, NUREG-0737 urges that, where practical, the RCS vents should be kept smaller than the size corresponding to the definition of a LOCA. <u>Id</u>.

33/ The pressurizer high point vent will be installed and be operable at restart. <u>See</u> Licensee's Response to the Atomic Safety and Licensing Appeal Board's Order of July 14, 1982 (August 12, 1982), at 14.

bound by this Commission regulation. Licensee contends that, absent a showing that TMI-1 is unique in its needs for these vents, installation should proceed in accord with the schedule required of all other operating reactors (including other B&W reactors). The Commission's schedule, reflecting its studied conclusion on the priorities for implementing improvements at operating reactors following the TMI-2 accident, no doubt reflects the fact that these vents are solely a back-up to mitigate a beyond design basis event -- the generation of noncondensible gases. See Tr. 4991-93 (Jensen).

We should also note that when the hot leg high point vents are installed at TMI-1, Licensee does not intend to use them to vent steam during design basis small break LOCAs which rely in part upon the boiler-condenser process. While implementing plant procedures for the use of this vent system have not yet been prepared, the generic operator guidelines have been developed. The guidelines call for utilizing the vent system only during an inadequate core cooling situation, when non-condensible gases would be present.34/ See Wallace Affidavit (attached). The operator will not be using the inadequate core cooling procedures for small breaks in which

34/ The Staff statement to the contrary, on pages 3 and 4 of the Enclosure to the March 25, 1982 letter from Eisenhut to Mattimoe ("Presently, we understand operators will be trained to use the high point vents to remove any steam bubbles"), simply is in error.

the boiler-condenser process is involved, unless and until the situation degrades, for some reason, beyond the design basis. See, e.g., Licensee Exs. 48 and 51 (plant emergency procedures).

Finally, it may be helpful for Licensee to explain briefly the impact of any order to install the hot leg high point vents prior to the restart of TMI-1. The attached Affidavit of Edward G. Wallace outlines the time and effort which would yet be required to install the vents, and the relationship of that effort to the ongoing steam generator repair program and the return of TMI-1 to service.

The major and essential pieces of equipment will have been received by the end of 1982, and the detailed engineering required is expected to be completed in January, 1983. Construction and installation of the system would then require 4 to 6 months, depending upon whether one or two shifts are utilized. Because construction must take place in containment and in the control room, it would have to be completed prior to the beginning of hot functional testing -- which is scheduled to begin in March, 1983. Consequently, there is not sufficient time to construct and install the hot leg high point vent system prior to the restart of TMI-1. See Wallace Affidavit.

VII. Proposed Reopening of the Proceeding

The Appeal Board stated its tentative view that, in the absence of the changes suggested in its opinion, it would need more evidence, possibly in the form of additional test results, before it would be able to conclude on this record that there is reasonable assurance that the plant can be operated without endangering the public. Memorandum and Order at 10.

In our comments above, Licensee has explained why the Licensing Board's concern with emergency feedwater reliability is without merit, and why a completion of its review of the record should cause the Appeal Board to alter its tentative views on the viability and reliability of the boiler-condenser and feed-and-bleed cooling modes. Consequently, it is Licensee's position that there is no need to reopen the proceeding. The record on the subjects raised by the Appeal Board in its Memorandum and Order is, we believe, complete. Licensee is not aware of additional relevant test results, which could be provided for the record and which would be dispositive. See Pacific Gas and Electric Company (Diablo Canyon Nuclear Power Plant, Units 1 and 2), CLI-81-5, 13 N.R.C. 361, 362-363 (1981). Absent some other unarticulated concern with the record, Licensee therefore submits that a reopening is not likely to be productive. To repeat, we also believe that a reopening is totally unwarranted.

VIII. Conclusion

For all of the foregoing reasons, it is Licensee's position that upon completion of its review of the evidentiary record the Appeal Board should modify the Licensing Board's conclusion and find that the TMI-1 emergency feedwater system is adequately reliable to permit restart of the plant, and should alter its own tentative and preliminary views on the viability of the boiler-condenser and feed-and-bleed cooling modes at TMI-1. The evidentiary record on emergency feedwater reliability and on the boiler-condenser and feed-and-bleed cooling modes is complete, and the uncontradicted, reliable and substantial evidence supports the positions of Licensee and the NRC Staff on the adequacy of the plant design. Nothing in the recent Board Notifications on the Semiscale tests contradicts that record. Consequently, there is no reason to reopen the proceeding, and the interim corrective measures suggested by the Appeal Board are not necessary.

Respectfully submitted,

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