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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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USNRC

ATOMIC SAFETY AND LICENSING APPEAL BOARD

'82 DEC 29 P3:27

OFFICE OF SECRETARY
DOCKETING & SERVICE
BRANCH

Administrative Judges:

Gary J. Edles, Chairman
Dr. John H. Buck
Dr. Reginald L. Gotchy

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| In the Matter of<br><br>METROPOLITAN EDISON COMPANY, ET AL.<br><br>(Three Mile Island Nuclear Station, Unit No. 1) | Docket No. 50-289<br><br>(Design Issues) |

MEMORANDUM AND ORDER

December 29, 1982

(ALAB-708)

Introduction

The Licensing Board issued its partial initial decision dealing with various issues of plant design, modifications, and procedures on December 14, 1981. LBP-81-59, 14 NRC 1211. Essentially, the Board concluded that, once various changes were made, TMI-1 could safely be restarted. The Union of Concerned Scientists (UCS) appealed from that decision. Briefs were filed and we heard oral argument on September 1, 1982.

In an unpublished memorandum and order issued on November 5, 1982, we set forth our preliminary views and concerns regarding the evidentiary record on the issues of the capability of the so-called "feed and bleed" and



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"boiler-condenser" processes to remove decay heat from the reactor core in the event of a loss of main feedwater or a small break loss of coolant accident at TMI-1. While acknowledging that our review of the record was not yet complete, we indicated that a reopening of the record might be necessary to resolve our concerns. We noted, however, that a more satisfactory alternative might be available. We then requested the parties' views regarding that alternative and, in the absence of our proposed changes, the need for reopening the record.

Those views are now before us. Briefly, the licensee and the NRC staff argue that the existing evidentiary record is adequate and that neither our proposed conditions nor a reopening of the record is required. [1/] The Union of Concerned Scientists (UCS) is in partial agreement with our analysis but maintains that the record, nevertheless, must be reopened. [2/]

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1/ See Licensee's Response to Appeal Board Memorandum and Order of November 5, 1982 (November 22, 1982) (hereinafter referred to as Licensee Response); NRC Staff Comments in Response to Appeal Board Memorandum and Order of November 5, 1982 (November 22, 1982) (hereinafter referred to as Staff Response).

2/ See UCS Response to Appeal Board Memorandum and Order of November 5, 1982 (November 22, 1982) (hereinafter referred to as UCS Response).

As we explain below, there are substantial inconsistencies in the parties' positions as well as in the testimony presented at the hearing. In addition, the parties' responses raise a number of questions that can not be resolved satisfactorily on the present record. We have concluded, therefore, that a limited reopening of the record is required to facilitate our prompt resolution of these matters.

Background

The TMI-2 accident raised questions about, among other things, the reliability of existing plant systems to provide adequate decay heat removal in the event of a main feedwater transient or certain small break loss of coolant accidents. In its August 9, 1979 Order and Notice of Hearing, the Commission ordered the licensee to take a number of short and long term actions to resolve certain stated concerns and directed the Licensing Board to determine whether those actions were necessary and sufficient to provide adequate protection of the public health and safety. CLI-79-8, 10 NRC 141, 144-46. Our review of the Board's initial decision on these matters requires a consideration of the soundness of the Board's conclusions regarding the sufficiency of the proposed corrective actions.

Before discussing the parties' arguments in detail, we believe that some further explanation of our concerns may be helpful. In the event of an accident involving the reactor or its safety systems, reactor operation automatically ceases. Although the fission process is terminated, heat continues to be produced in the reactor core by the radioactive decay of fission products.[3] As a result, a reliable means of removing this decay heat is required for an extended period after reactor shutdown.

In the event of a small break loss-of-coolant accident or a main feedwater transient, the record suggests essentially two means of reactor core decay heat removal at TMI-1, depending on the conditions that are present.[4] If the emergency feedwater system is available, core cooling may be accomplished by natural circulation of reactor coolant to the steam generators, where heat is transferred to secondary water which converts to steam. Natural circulation is dependent upon the difference in reactor coolant density in the reactor core and the steam generators.

There are two possible types of natural circulation, depending upon the state of the reactor coolant. If the

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3/ The heat rate drops immediately upon shutdown to less than 10 percent of full reactor power, followed by a more gradual decrease.

4/ The reactor coolant pumps and main feedwater system are assumed to be inoperative because they are not safety-grade.

reactor coolant system is relatively free of steam bubbles, liquid (also called single-phase) natural circulation can be maintained. If there is substantial steam formation at the high points of the reactor coolant system, however, cooling would depend on the establishment of a type of two-phase natural circulation referred to as the "boiler-condenser" mode. In this process, core decay heat generates steam, which rises through the hot legs to the steam generators, where it condenses. Water then flows through the cold legs to the core, where the process begins anew. As indicated above, either type of natural circulation is dependent on the operability of the emergency feedwater system.

If emergency feedwater is not available, decay heat must be removed by the so-called "feed and bleed" process, in which cooling water is injected into the reactor vessel by the high pressure injection (HPI) pumps and expelled from the system through the break itself, the power-operated relief valve (PORV), or the safety relief valves. For this process to be successful, flow from the HPI pumps must be sufficient to replace the amount of coolant lost out of the system.

As we noted in our November 5, 1982 memorandum and order (at 2-3), the Licensing Board found that the emergency feedwater system at TMI-1 was not sufficiently reliable, by itself, to provide adequate protection of the public health and safety. This conclusion was based essentially on a quantitative probabilistic analysis of the so-called "failure" on demand of the emergency feedwater system. It also appears to be based, at least in part, upon the Board's observation that the emergency feedwater system will not be fully safety-grade at restart. The Board concluded, as a result, that feed and bleed is needed as a backup. LBP-81-59, supra, 14 NRC at 1370-72 (1981).

As discussed above, natural circulation (either liquid or boiler-condenser mode) must be maintained to transport decay heat from the reactor core to the steam generators to provide adequate core cooling using the emergency feedwater system. The record indicates that liquid natural circulation may be lost during a small break LOCA. See pp. 4-5, supra. Our preliminary view was that the viability of the boiler-condenser or two-phase mode of natural circulation cooling had not been adequately proved on the record. To remove steam and to help reestablish single phase natural circulation cooling, we suggested that the vents in the hot leg high points could be used. We also suggested that an individual be assigned to operate the emergency feedwater flow control valves manually in the event that the Integrated Control System (ICS), which is not safety-grade, failed to operate. We indicated that, with these two modifications in place, we would be prepared to find the emergency feedwater system sufficiently reliable that feed and bleed would not be required. Memorandum and

Order of November 5, 1982 at 9-10. [5] Because these measures were not fully considered at the hearing, we requested, among other things, "the parties' views concerning the sufficiency of our proposed requirements."

We also offered our preliminary view that there is insufficient evidence of record to support the Board's finding that feed and bleed is a viable means of decay heat removal at TMI-1. We noted, in addition, that information supplied us by the staff in two recent Board notifications

5/ The licensee challenged as inappropriate the Licensing Board's reliance on quantitative analysis as a basis for concluding that the emergency feedwater system is unreliable. While we have reached no final conclusions with respect to this aspect of the licensee's argument on appeal, we believe that the record is adequate concerning the reliability of the emergency feedwater system in the event of a small break LOCA or a loss of main feedwater at TMI-1.

Very recently, we received two Board Notifications (BN-82-118 and BN-82-118A) which discuss a report by a staff consultant that the emergency feedwater system at TMI-1 may lack the capability to withstand a postulated safe shutdown earthquake. (Although those Board Notifications are dated November 22, 1982 and December 9, 1982, respectively, we did not receive them until December 22, 1982.) The scope of this proceeding does not include seismic qualification of the EFW system. This information does raise the possibility, however, that reliance may have to be placed on other plant systems to provide adequate core cooling. We do not address seismic qualification of the EFW system in this memorandum and order. That matter will be considered by the NRC staff and the Commission outside the adjudicatory process.

tended to undermine the Licensing Board's conclusion. 6/ As we discuss later, the staff's response to our November 5, 1982 order lends support to its position that feed and bleed would provide adequate core cooling at TMI-1.

Analysis

The responses we received raise many questions which we believe must be answered before we can reach a final decision on these matters. There are also a number of inconsistencies in the evidence of record which, in our judgment, must be satisfactorily resolved in order to facilitate our review. Our discussion of them follows.

A. Emergency Feedwater System Reliability

As mentioned previously, the Licensing Board found that the emergency feedwater system, even after it is modified to full safety-grade status, will not be sufficiently reliable to protect the public without feed and bleed as a backup. See pp. 5-6, supra. UCS endorses that finding and argues that our proposed modifications are therefore not sufficient without the availability of feed and bleed. 7/

In contrast, the licensee points out that it has appealed the Licensing Board's decision on emergency feedwater reliability and that the staff has supported that

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6/ See BN-82-93 (Sept. 14, 1982); BN-82-107 (Oct. 22, 1982).

7/ See UCS Response at 3.

appeal. The licensee urges that we modify the Board's decision to hold that the short and long term actions are sufficient to protect the public health and safety. In short, the licensee argues that the emergency feedwater system is sufficiently reliable and that feed and bleed cooling is not necessary. [8/] Although not expressly stated as such, the staff's position appears to be the same for it, too, argues that reliance on feed and bleed is not required. [9/]

It is not our intention to address the entire question of emergency feedwater system reliability now. Nor is it necessary to do so. We shall consider that subject, including the licensee's argument regarding the Board's reliance on quantitative analysis, more fully in our final decision addressing all of the design issues that are before us. At this juncture, it should suffice to note that because of our concerns that steam voids may interrupt liquid natural circulation and that the boiler-condenser process may not be a viable means of decay heat removal (see pp. 15-16, 24-33, _infra_), we are currently unable to determine whether the short term actions to improve emergency feedwater system reliability are sufficient to protect the public.

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8/ See Licensee Response at 4-5, 9-12.
9/ See Staff Response at 8. But see note 5, _supra_.

In our judgment, there are three ways (and perhaps others) in which our concerns might be resolved: (1) the vents to be installed in the hot leg high points could be shown to be useful for successfully removing steam and restoring liquid natural circulation; (2) the boiler-condenser process could be adequately demonstrated as a viable means of decay heat removal at TMI-1; or (3) the viability of feed and bleed as a means of decay heat removal could be sufficiently proven. As we explain in the balance of this memorandum and order, we would need additional evidence before we could accept any one of those propositions in this case. Contrary to the licensee's suggestion (Licensee Response at 5), our conclusion does not depend upon whether or when the emergency feedwater system at TMI-1 will be fully safety-grade. Rather, it stems from our judgment that the problems presented by steam voiding must be adequately resolved for both the short and the long term.

As we mentioned above, the staff and licensee would have us rely upon the emergency feedwater (EFW) system to remove core decay heat in the event of a small break LOCA or a main feedwater transient. See pp. 8-9, <u>supra</u>. See also Tr. 4816-18 (Keaten); Tr. 5016, 5502-03 (Jensen); Tr. 5645-47 (Lanese); Tr. 6146 (Wermiel). We must reiterate that reliance upon the emergency feedwater system necessarily involves reliance upon natural circulation (liquid or boiler-condenser mode) to transport the decay heat from the reactor core to the steam generators. Although the system is undergoing extensive modification, it will not be fully safety-grade at restart. Capodanno et al., fol. Tr. 5642, at 1.

Because the record was unclear regarding the status of the EFW modifications, we requested information on this subject prior to oral argument. 10/ The licensee provided a list of the modifications that will be completed before restart and those to be completed during the next refueling outage. 11/ The staff indicated that the EFW system will be fully safety-grade by the end of the next refueling outage. 12/

One of the near-term modifications which the licensee listed was the provision of operator control of emergency feedwater flow to each steam generator independent of the Integrated Control System (ICS). 13/ In our November 5, 1982 memorandum and order (at 9-10), we discussed our

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10/ See our Order of July 14, 1982 (unpublished) at 3-4.

11/ Licensee's Response to Appeal Board Order of July 14, 1982 (August 12, 1982) at 9-13.

12/ Affidavit of Richard H. Jacobs (Aug. 6, 1982) at 4-5, attached to NRC Staff's Response to Appeal Board's Order of July 14, 1982 (August 9, 1982).

13/ Licensee's Response (August 12, 1982) at 10.

concern for the dependence of the EFW system on the non safety-grade ICS to operate the EFW flow control valves. We noted that the record was unclear as to the safety-grade status of the EFW manual control capability. Id. at 9 n.19. See, e.g., Tr. 5580-81 (Jensen), 5710-11 (Lanese), 7106-07 (Broughton), 7705 (Keaten); Staff Ex. 1 at C1-11. The licensee responds that the manual control stations will be powered from a Class 1E (i.e., high reliability) power supply and a single failure in the manual circuits will not result in a loss of system function. 14/ We interpret this response to mean that the manual control capability will not be fully safety-grade but is considered by the licensee to be highly reliable. The staff, however, asserts that a "safety-grade manual control capability" exists at TMI-1.15/ This apparent inconsistency leads us to wonder whether (1) equipment projected to be safety-grade prior to restart may not actually be so, and (2) equipment that was not intended to be safety-grade by restart may be so. These two questions must be resolved by evidence of record.

In our November 5, 1982 memorandum and order (at 9), we proposed the assignment of an individual whose sole function would be to operate the flow control valves manually

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14/ Licensee Response at 13.

15/ Staff Response at 3.

following the onset of an accident.16/ We indicated that this assignment would resolve our concern for the dependence of the emergency feedwater system on the non safety-grade ICS. The licensee referred us to plant procedures that require the control room operator to dispatch an auxiliary operator to the flow control valves for any EFW pump auto-start condition. See Lic. Ex. 49 at 2.0, 6.0; Lic. Ex. 48 at 10.0, 30.0.17/ If the emergency feedwater flow were not achieved by the control room operator, the auxiliary operator would take manual control of the flow control valves.18/ We are satisfied with the plant procedures for manual control of the EFW flow control valves. Provided that they are retained for use by TMI-1 operators, we consider our concern regarding the capability for manual control of emergency feedwater to be resolved.

UCS argues that the emergency feedwater control capability is not safety-grade because there is only one

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16/ The licensee appears to have interpreted this proposal to mean the stationing of an operator at the valves on a full-time basis. See Licensee Response at 12 n.14. However, our intent was the assignment of this duty to an individual only if an accident should occur.

17/ *Id.* at 14.

18/ *Id.*

flow control valve for each steam generator. 19/ It claims that a break in one of the steam generators would cause isolation of that steam generator, with the result that a single failure of the flow control valve to the other steam generator would cause a total loss of feedwater. UCS asserts that this possibility would exist regardless of whether emergency feedwater control is manual or automatic.

We disagree. As explained above, we are satisfied with the licensee's procedures for manual control of the valves as a short-term measure before the emergency feedwater system is fully safety-grade. A single electrical failure of a flow control valve could be overcome by manual control of the valve handwheel. A single mechanical failure of the flow control valve would not affect the operability of the entire EFW system, which should provide adequate core cooling. 20/ In addition, the licensee is modifying the

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19/ UCS Response at 2. One of the long-term modifications to achieve a fully safety-grade EFW system is the provision for parallel EFW flow control valves to each steam generator. See Wermiel and Curry, fol. Tr. 16,718, at 25, 30.

20/ General Design Criteria 34 (Residual heat removal) and 35 (Emergency core cooling) of Appendix A to 10 CFR Part 50 require that adequate core cooling be available in the event of a "single failure." A single failure is defined as "an occurrence which results in the loss of capability of a component to perform its intended safety functions. Multiple failures resulting from a single occurrence are considered to be a single failure." 10 CFR Part 50, Appendix A, Definitions and Explanations. Staff witness Jensen testified that two HPI pumps would provide adequate core cooling even if emergency feedwater were not available. Tr. 5588-89. See also our discussion of feed and bleed (pp. 33-42, *infra*).

flow control valves prior to restart to provide backup instrument air supplies with provisions for the valves to move to the open position upon loss of instrument air. See Lic. Ex. 1 at 2.1-25-26; Lic. Ex. 15 at 6-7. As a result, we consider the manual control capability together with the licensee's short-term modifications to make the EFW flow control valves sufficiently reliable until the emergency feedwater system is modified to full safety-grade status. We shall address the long-term modifications in our final decision. 21/

B. Liquid Natural Circulation

As discussed earlier, natural circulation (either liquid or boiler-condenser mode) must transport decay heat from the reactor core to the steam generators for the core to be adequately cooled using the emergency feedwater system. In this section, we discuss maintenance of liquid natural circulation and the possible use of the vents. Our concerns for the viability of the boiler-condenser mode are discussed in the following section.

Analyses indicate that liquid natural circulation would be interrupted by steam formation for any break in the reactor coolant system larger than about .005 $ft^2$ if only

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21/ At that time, we shall also address UCS' argument on appeal that the Licensing Board improperly delegated its decisionmaking authority to the staff to provide a long-term solution to the steam generator bypass logic problem. See UCS Brief on Exceptions to the Partial Initial Decision of December 14, 1981 (March 12, 1982) (hereinafter referred to as UCS Brief) at 58.

one HPI pump were operating and about .01 $ft^2$ if two HPI pumps were operating. Tr. 4683-84 (Jones). 22/ Steam bubbles would collect at the high points of the primary system. It may be possible to remove this steam by use of the reactor coolant pumps or by ejection from high point vents. Tr. 4617, 4623-24 (Jones). The reactor coolant pumps are not safety-grade and, as a result, cannot be relied upon to perform this function. Therefore, we concentrate our discussion on the vents to be installed in the hot leg high points.

The parties are in agreement that the capability of the hot leg vents to remove steam from the high points of the hot legs sufficiently to re-establish natural circulation is not demonstrated on the record. In its response to our November 5, 1982 memorandum and order, the licensee goes further to state that "the record at best casts doubt on the

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22/ The location of the break can significantly affect the ability of emergency core cooling systems to safely mitigate an accident. B&W analyses indicate that the reactor coolant pump discharge is the worst location for a small break because substantial loss of HPI flow out the break will occur. Lic. Ex. 5 at Section 6.2.1.3.2. Where witnesses have not specified the break location, we have assumed it to be the reactor coolant pump discharge.

utility of these vents to remove steam and re-establish natural circulation." 23/

The licensee and UCS cite staff statements at oral argument to the effect that calculations performed at Los Alamos National Laboratory indicate that the vents may not be useful in restoring natural circulation. 24/ See App. Tr. 291-92 (Sheron). We note, however, that those calculations assumed a vent of approximately 1 centimeter (0.394 in.) in diameter, whereas the vents to be installed at TMI-1 were reported to be 0.8 inches in diameter. 25/ The flow rates associated with these different vent sizes may have a significant effect on the potential for successful use of the vents to promote natural circulation.

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23/ Licensee Response at 39. The licensee argues that its witness Jones was referring only to the TMI-2 accident in discussing the use of the vents to restore natural circulation. *Id.* at 40. See Tr. 4617, 4623-24. While we agree that Mr. Jones initially addressed the circumstances of the TMI-2 accident, his testimony can be fairly read to include the general use of the vents to promote liquid natural circulation at TMI-1. See Tr. 4623-24. Later, Mr. Jones also discussed the use of the vents to assist in refilling the primary system and restoring natural circulation. Tr. 10,778.

24/ Licensee Response at 40; UCS Response at 4.

25/ See Board Notification BN-82-65 (July 9, 1982), Enclosure 1 at 27, 40-41. See also Tr. 4865 (Jones). For perspective, the size of the PORV is 1.05 $in^2$ (*i.e.*, about 1.15 inches diameter). Tr. 5090 (Jones).

In order to confirm or reject the capability of the vents, additional tests with more realistic plant characteristics would be necessary.

UCS suggests that opening the vents, with the resultant loss of pressure, might cause more water to flash to steam if there is inadequate margin to saturation. 26/ The staff also argues that the vents would be "both unnecessary and ineffective" in re-establishing liquid natural circulation. 27/ The staff then indicates, however, that the vents may be beneficial in recovering liquid natural circulation "from a condition of prior operation in feed and bleed or boiler-condenser natural circulation." 28/ Although the staff's argument is not entirely clear, we understand it to be similar to that advanced by UCS -- i.e., that the vents would not be useful when the primary coolant is saturated because coolant would flash to steam as a result of depressurization when the vents were opened.

The staff also discusses the possible use of the vents to perform the "bleed" function during feed and bleed

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26/ UCS Response at 4-5.

27/ Affidavit of Walton L. Jensen, Jr. (Nov. 22, 1982) at 3, attached to Staff Response.

28/ Id.

cooling. [29/] Staff calculations indicate that the vents would be too small to provide adequate steam relief for a significant period after reactor shutdown. [30/] Similarly, UCS suggests that "some of the same difficulties with feed and bleed demonstrated by the Semiscale tests S-SR-1 and S-SR-2 might also be encountered in attempting to 'bleed' the steam accumulated in the hot leg through the vents."[31/] UCS argues that, depending on the conditions present, flow through the vents could be two-phase or liquid with a potential net loss in reactor coolant system inventory.

It is possible that, during saturated conditions in the hot legs, the vents might not be useful in removing sufficient excess steam to restore natural circulation. It is also possible that the vents might not be of use for feed and bleed immediately after reactor shutdown. These matters must be explored further before any firm conclusions can be drawn.

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29/ *Id.* at 4-7.

30/ *Id.* at 4. We note that the vent size (0.5 inches diameter) specified by staff witness Jensen is significantly smaller than that (0.8 inches) indicated by the licensee in its testimony. See Tr. 4865 (Jones).

31/ UCS Response at 4.

The licensee asserts that the Commission has established the purpose of the vents and the schedule for their installation in connection with its hydrogen control rulemaking. 32/ The staff also observes that the vents are designed to remove noncondensible gases in accordance with 10 CFR §50.44. 33/ While it is true that the Commission has required the installation of high point vents in connection with hydrogen control, it is not at all clear to us that the only permissible use for the vents is the removal of noncondensible gases. 34/ The licensee itself has indicated that the vents could also provide an alternate means of reactor coolant removal when release outside the containment building is not permitted because of high radioactivity in the reactor coolant. See Lic. Ex. 1 at 2.1-38e.

We fully appreciate the Commission's admonition -- recently reaffirmed in CLI-82-32, 16 NRC ___ (Oct. 22, 1982) -- that the issue of whether the licensee has satisfactorily

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32/ See Licensee Response at 40-42.

33/ Staff Response at 4.

34/ We note, for example, that in an enclosure (at 1) to a letter from NRC Chairman Palladino to the Honorable Morris K. Udall (July 30, 1981) discussing the formation of a steam bubble at TMI-2 in September 1977 during hot functional testing, it was stated that the "ability to cope with incidents involving gases or vapor in the system is now being provided through installation of high point vents."

completed necessary short-term or long-term items shall be determined by the NRC staff and the Commission outside the adjudicatory process. We have no intention of altering any schedules the staff or the Commission might establish for the completion of required items or deciding whether various required steps have been completed. 35/ Our responsibility, however, as the Commission specifically pointed out in CLI-82-32, 16 NRC at ___ (slip opinion at 1-2), is to determine "what short-term or long-term actions are necessary and sufficient to adequately protect the public health and safety." Consistent with that mandate, we believe we have the authority to determine (should the evidence support such determination) that the installation of high point vents prior to restart as a means of removing excess steam to assure restoration of natural circulation is a necessary short-term action which must be taken before we can find that the public health and safety is adequately protected.

As UCS correctly points out, significant questions remain regarding the adequacy of operator training and

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35/ The Commission, for example, has decided on a timetable for the installation of high point vents as a means of removing noncondensible gases; such vents may be installed no later than the first refueling outage after restart. In such circumstances, we may not require, as a condition of restart, that the removal of noncondensible gases by means of high point vents be available.

emergency procedures for use of the high point vents. 36/ The licensee states that the vents are intended to be used during inadequate core cooling only to remove noncondensible gases. 37/ In addition, the licensee asserts that its operators will not be trained to use the high point vents to remove steam. 38/ This is inconsistent with the staff position stated in a March 25, 1982 letter from the Director of the Division of Licensing, Office of Nuclear Reactor Regulation to the Babcock & Wilcox (B&W) Owners Group that was the result of a staff meeting with the Owners Group. 39/ Thus, we find the licensee's assertion unsettling. In contrast, the owner of another B&W plant, Rancho Seco, has provided information to the staff discussing the possible use of the hot leg vents to remove steam during "normal"

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36/ See UCS Response at 5.

37/ See Licensee Response at 43.

38/ Id. at 43 n.34.

39/ The letter states that, in the staff's understanding, "operators will be trained to use the high point vents to remove any steam bubbles." Letter from Darrell G. Eisenhut to J.J. Mattimoe, Enclosure at 3-4. In this connection, we note that the release of non-condensible gases is likely to be accompanied by the formation and release of steam.

(i.e., adequate core cooling) small break LOCAs. 40/

Finally, the licensee indicates that there is not sufficient time to construct and install the hot leg high point vent system prior to restart. 41/ The licensee explains that major and essential pieces of equipment will have been received by the end of this year but that the detailed engineering is not yet complete. Construction and installation would then take some four to six months. 42/

There is conflicting evidence concerning whether the vents might be useful in removing steam voids from the high points of the primary system and in restoring liquid natural circulation. Such a procedure might be useful, for example, if steam voids are produced during a small break LOCA after the HPI pumps have refilled the primary system or during

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40/ See letter from J.J. Mattimoe to Director of Nuclear Reactor Regulation (July 1, 1981) "Position Paper on Reactor Vessel Head Vents" at Section 4.1.2; letter from W. Walbridge to Director of Nuclear Reactor Regulation (March 4, 1982), Enclosure at 8. Both letters are part of the record in the Rancho Seco special proceeding (Docket No. 50-312), which is now undergoing Appeal Board review. See, e.g., Sacramento Municipal Utility District (Rancho Seco Nuclear Generating Station), ALAB-703, 16 NRC ___ (Nov. 23, 1982).

41/ Licensee Response at 44.

42/ Id. We note that this statement appears to be inconsistent with that made to Commissioner Gilinsky during a recent site visit. See Memorandum to File from Edward Abbott (Nov. 5, 1982) at 3, which states that "[m]uch of the electrical work for the vent modification is complete and the hardware is on-site."

plant cooldown. 43/ As the foregoing makes clear, however, many open questions remain and some further analysis on the record is required.

C. Two-Phase Natural Circulation (Boiler-Condenser Process)

In our November 5, 1982 memorandum and order, we indicated our tentative view that the ability of the boiler-condenser mode of natural circulation to remove enough decay heat to prevent core damage had not been adequately demonstrated on the record. 44/ UCS apparently shares that conclusion but does not comment on it in detail. 45/ The licensee and the staff, however, argue that

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43/ On June 11, 1980, a steam bubble formed in the vessel head during a natural circulation cooldown at St. Lucie. See IE Circular No. 80-15 (June 20, 1980). Also, IE Circular No. 81-10 (July 2, 1981) discusses steam voiding in the reactor coolant system during decay heat removal cooldown.

44/ See our Memorandum and Order of November 5, 1982 at 7 n.15, referencing testimony by licensee witness Jones that this mode had been predicted by computer modeling but no tests had been performed to demonstrate its viability. See Tr. 4687-88, 4691, 4702; Jones and Broughton (Board Question on UCS Contention 8), fol. Tr. 5038, at 16-17. We also 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 Babcock & Wilcox (B&W) plants during small break loss of coolant accidents. See, e.g., letter from P. Shewmon to William J. Dircks (October 13, 1982); letter from Darrell G. Eisenhut to J.J. Mattimoe (March 25, 1982).

45/ See UCS Response at 1.

there is no basis for our view.[46]

The licensee argues that the process was endorsed by witnesses for both the staff and the licensee, and that no witness presented testimony questioning the efficacy of that process.[47] Licensee witness Jones testified, however, that there have been no tests of this method of decay heat removal at TMI-1 and that the licensee does not intend to conduct any because there is insufficient instrumentation to control the process. Tr. 4687-88. In addition, there has been no experience with the boiler-condenser process as a stable cooling mode. Tr. 4685-87 (Jones). In our judgment, this testimony raises doubts about whether the process can be relied upon to provide adequate protection of the public health and safety in the event of an accident.

The licensee also asserts that UCS has abandoned its interest in questioning the adequacy of the licensee's small-break LOCA analysis.[48] That argument is somewhat misleading, for UCS filed and briefed several exceptions concerning the boiler-condenser mode.[49] UCS would have us

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46/ See Licensee Response at 25; Staff Response at 2, 6.

47/ Licensee Response at 17.

48/ Id. at 19.

49/ See UCS Brief at 3, 5, 8-9, 15.

reject the Licensing Board's conclusion that the TMI-2 accident did not reveal a problem with reliance on natural circulation. That conclusion, UCS asserts, was based in part upon the incorrect premise that the boiler-condenser mode will be established and will remove sufficient core decay heat. 50/ In addition, UCS takes exception to the Board's finding that the boiler-condenser mode meets the requirements of General Design Criteria 34 and 35. 51/ See note 20, <u>supra</u>. UCS charges that the Board failed to confront evidence demonstrating that the boiler-condenser mode is not sufficiently reliable because (1) there is no instrumentation to determine primary water level in the steam generators; 52/ (2) emergency procedures require refilling of the primary system, which will prevent the establishment of the boiler-condenser mode; 53/ and (3) the

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50/ *Id.* at 2-3.

51/ *Id.* at 8-9. See LBP-81-59, *supra*, 14 NRC at 1230.

52/ This issue will be addressed in our final decision on design issues.

53/ UCS explains that refilling the primary system, as the operators are directed to do following a LOCA, would block the steam condensing surface in the steam generators and preclude boiler-condenser cooling. UCS Brief at 8. We agree that, if the primary system could be refilled, this would preclude the boiler-condenser mode until the primary level dropped sufficiently to expose a condensing surface. However, if the primary system can be kept full, the boiler-condenser mode would not be needed.

effectiveness of that process has not been tested. 54/ Finally, UCS argues that the boiler-condenser mode is not sufficiently reliable because of its dependence on the emergency feedwater system. 55/

The licensee maintains that the B&W emergency core cooling system (ECCS) evaluation model is an NRC-approved computer code under Appendix K to 10 CFR Part 50, and therefore is not open to challenge in this proceeding. 56/ The B&W ECCS evaluation model was approved in September 1978 and no changes have been made since then for demonstrating compliance with 10 CFR §50.46. Tr. 5159 (Jones). Accident analyses performed prior to the TMI-2 accident did not include breaks smaller than .04 $ft^2$. Tr. 4691-92 (Jones); Tr. 5505-06 (Jensen). In those analyses, reliance on the boiler-condenser process was unnecessary because the break was sufficiently large to permit adequate removal of decay heat through the break itself. Tr. 4691-92 (Jones). Following the TMI-2 accident, new analyses were performed, primarily to provide guidance for the preparation of operator procedures. Jones and Broughton (Board Question on UCS

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54/ Id. at 8-9.

55/ Id. at 9, 15. Unlike that of UCS, our concern for the viability of the boiler-condenser mode is not related to the reliability of the emergency feedwater system.

56/ Licensee Response at 17-19.

Contention 8), fol. Tr. 5038, at 4-5; Tr. 5517-18 (Jensen). In addition, the staff group responsible for review of the B&W small break LOCA analyses, the Bulletins and Orders (B&O) Task Force, did not review the adequacy of the Appendix K model. Tr. 5544-46 (Jensen). 57/ Thus, it is not altogether clear to us that a challenge to the ability of the model to predict correctly boiler-condenser flow can be considered an impermissible attack on the Commission's regulations.

Staff witness Jensen testified that questions had been raised by other members of the B&O Task Force with regard to the degree to which data predicted by the models had been compared with experimental data in the small break range. Tr. 5583-84. The staff's generic small-break LOCA analysis for B&W reactors states that the "methods must be revised and verified before they can be considered for NRC approval under 10 CFR 50.46." Board Exh. 4 at 2-3. Staff witness Jensen appeared to interpret this recommendation to mean that the models will be reviewed by the staff as additional experimental data become available. Tr. 5021-24. Licensee witness Jones disagreed with staff recommendations

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57/ The staff provided the results of its review of the B&W small-break LOCA analyses in NUREG-0565, Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox Designed 177-FA Operating Plants (January 1980). NUREG-0565 is included in the record as Board Exhibit 4.

concerning the need for experimental verification of the B&W analyses. See generally Tr. 5221-30.

Staff witness Jensen believed that the smallest break that must be analyzed for the purpose of verifying compliance with Appendix K or the limits of 10 CFR §50.46 are breaks slightly smaller than the most severe in order to show that the most severe has been identified. Tr. 5527.58/ The smallest break that was reviewed for the purpose of conformance with Appendix K was .04 $ft^2$. Tr. 5538. Mr. Jensen also indicated that the analysis of a .005 $ft^2$ break was performed for the purpose of providing guidance for operator actions in the event of a small break LOCA. Tr. 5527. We do not understand the basis for staff's position that breaks of approximately .07 $ft^2$ are the only ones that must be analyzed in order to demonstrate compliance with the regulations. As the licensee acknowledges, the boiler-condenser mode may be needed for breaks smaller than approximately .02 $ft^2$ to help provide core cooling if liquid natural circulation is lost. 59/ Therefore, it would appear

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58/ The most "severe" break (i.e., that break producing the highest peak cladding temperature) has been identified by analysis to be .07 $ft^2$ at the reactor coolant pump discharge. Jensen, fol. Tr. 5496, at 5-6; Lic. Ex. 5 at Section 6.2.1.3.3.

59/ Licensee Response at 16.

that analyses must be performed to demonstrate that the boiler-condenser mode is adequate to prevent the limits of 10 CFR §50.46 from being exceeded during these small break accidents.

The licensee cites testimony that experimental tests of the boiler-condenser mode have been performed for primary systems with U-tube steam generators. 60/ See Ross and Capra, fol. Tr. 15,806, at 34-35; Tr. 5223-24 (Jones). The staff also responds that tests involving U-tube steam generators demonstrate the effectiveness of the boiler-condenser mode for TMI-1 because the same basic heat transfer mechanisms would occur. 61/ While these tests confirm the effectiveness of the boiler-condenser mode for plants with U-tube steam generators, we are not convinced that they establish the viability of this mode for plants like TMI that have a different primary system piping configuration and straight-through steam generators. 62/

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60/ *Id.* at 20-21.

61/ Affidavit of Walton L. Jensen, Jr. at 2-3, attached to Staff Response.

62/ In this regard we note that the absence of a test facility that conforms to the TMI-1 design is one of the concerns discussed in recent ACRS and staff correspondence. See letter from P. Shewmon to William J. Dircks (October 13, 1982); letter from Darrell G. Eisenhut to J.J. Mattimoe (March 25, 1982).

In its response, the staff explains that its need for additional experimental data does not contradict its original conclusion on the efficacy of the boiler-condenser mode. 63/ The licensee makes a similar argument, quoting staff statements made at oral argument concerning the need for long-term model confirmation. 64/ See App. Tr. 284 (Sheron). At oral argument, the staff indicated that it did not have confirmation of the process of trapping a steam bubble in the hot legs and that the re-establishment of natural circulation had not been demonstrated experimentally. App. Tr. 287 (Sheron).

The licensee asserted below that the boiler-condenser mode occurred during the TMI-2 accident. 65/ See Tr. 4627-30, 4685-86 (Jones). But its witness Jones conceded that the first time at which it can be documented that adequate core cooling was established at TMI-2 was at 16 hours after the onset of the accident, when the reactor coolant pumps were started. Tr. 4655. Therefore, we do not believe that the boiler-condenser mode can be considered viable on the basis of the TMI-2 accident experience alone.

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63/ Affidavit of Walton L. Jensen, Jr. at 3, attached to Staff Response.

64/ Licensee Response at 24-25.

65/ Id. at 20.

Our concern is not with the mechanics of the boiler-condenser process but rather with the ability of this mode to remove sufficient decay heat to adequately provide core cooling. The licensee relies on testimony to the effect that tests are not needed to confirm that the basic phenomenon works but may be used to confirm the accuracy of the code in predicting the amount of heat transfer for a given system heat condition. 66/ See Jones and Broughton (Board Question on UCS Contention 8), fol. Tr. 5038 at 16-17. As mentioned earlier, the licensee does not plan to conduct any such tests. See p. 25, supra.

From the record, it appears that the boiler-condenser mode may be needed only for a limited time period during certain small break LOCAs. 67/ Once the core decay heat rate has dropped sufficiently, one HPI pump could supply adequate flow to provide core cooling without the aid of natural circulation. 68/ For example, analyses indicate

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66/ Id. at 21-22. Licensee witness Jones claimed, without substantiation, that there may be significant conservatism in the model. Tr. 5293-95.

67/ Natural circulation would not be needed for breaks larger than approximately .01 ft² because the break could adequately remove core decay heat. Jensen (UCS Contention 1) fol. Tr. 4913, at 5; Tr. 4930-31 (Jensen); Tr. 4852-54 (Jones).

68/ Analyses indicate that two HPI pumps would provide adequate core cooling for any small break LOCA even if the EFW system were not available. Tr. 5588-89 (Jensen). However, this would not meet the Commission's regulations concerning the assumption of a single failure. See generally 10 CFR Part 50, Appendix A.

that one HPI pump could match core decay heat after about one hour for a .005 $ft^2$ break with EFW available. Tr. 5549-53 (Jensen). See also Lic. Ex. 5 at Section 6.2.4.3.3. It is for the time period before the available HPI flow could match the boil-off rate of core decay heat that we believe additional analysis is needed in order to confirm that the boiler-condenser mode can adequately remove core decay heat.

D. Feed and Bleed

As mentioned previously, the Licensing Board relied on feed and bleed as a backup to the emergency feedwater system, which it considered not sufficiently reliable. Based on the testimony of several staff and licensee witnesses, 69/ the Licensing Board found that, in the event of a failure of the emergency feedwater system, the core could be adequately cooled using feed and bleed while repairs to the emergency feedwater system were being made. LBP-81-59, *supra*, 14 NRC at 1370. We believe that there is insufficient evidence of record at the present time to support the Licensing Board's conclusion. We reiterate that our interest in feed and bleed as a backup is not based upon the Board's conclusions regarding emergency feedwater reliability. Rather, it stems from our judgment that the

69/ See, *e.g.*, Jones, fol. Tr. 4589, at 1-4; Tr. 5586-89 (Jensen); Capodanno *et al.*, fol. Tr. 5642, at 1-3, 11; Tr. 6200-01, 16,734-36, 16,846-47, 16,893-94 (Wermiel); Tr. 7704-09, 7806 (Keaten).

boiler-condenser mode of core cooling has not been adequately demonstrated.

Our primary concern with the viability of feed and bleed does not involve the reliability of the operators or plant equipment. The record appears to contain sufficient evidence to support a conclusion that the operations associated with feed and bleed are relatively simple and employ, for the most part, safety-grade systems. See, e.g., Keaten and Jones, fol. Tr. 4588, at 12; Tr. 4734-35, 4777-830 (Keaten and Jones); Wermiel et al., fol. Tr. 6035, at 5-7; Keaten et al., fol. Tr. 16,552, at 10-11. See also Licensee Response at 27-29. 70/

Nevertheless, we are still somewhat troubled by the lack of experimental verification of the process predicted by computer models. Both the staff and the licensee argue that computer analyses predict the capability of feed and bleed to adequately provide core cooling in the event of various small breaks. 71/ See, e.g., Jones, fol. Tr. 4589, at 1-2; Jones and Broughton (UCS Contention 8 and ECNP Contention 1(e)), fol. Tr. 5038, at 4-8; Jensen (UCS Contention 1), fol. Tr. 4913, at 9. See generally Lic. Exs. 3-9 and 13. No experimental verification of these analyses

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70/ These matters will be discussed further in our final decision on the technical issues in this proceeding.

71/ Staff Response at 3-4; Licensee Response at 30, 37-39.

has been introduced into the record. We identified our interest in such experimental verification in questions posed prior to and at oral argument, in which we made specific reference to the loss-of-fluid test (LOFT) facility. 72/ The staff construed our requests to be limited to LOFT tests and failed to mention the Semiscale test facility. 73/

On September 14, 1982, two weeks after oral argument, we received Board Notification BN-82-93, which provided information on recent experimental testing of feed and bleed at the Semiscale facility. The preliminary report from EG&G attached to BN-82-93 described a test that led to an uncovering of the core. It concluded that the results "tend to support a concern about the relative tenuousness of the process." 74/ Also included was a staff memorandum that briefly discussed the test results. It stated: "Although neither the staff nor the licensees or applicants have ever relied upon feed and bleed in order to meet the Commission's

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72/ See, e.g., our Order of July 14, 1982 at 14; App. Tr. 206-12, 292-96. See generally App. Tr. 282-98.

73/ See Affidavit of Walton L. Jensen, Jr. (Aug. 6, 1982) at 10, attached to NRC Staff Response to Appeal Board's Order of July 14, 1982 (August 9, 1982).

74/ Letter from P. North, Manager of Water Reactor Research Test Facilities Division, EG&G, to R.E. Tiller, Director of Reactor Operations and Programs Division, Idaho Operations Office, Department of Energy (Aug. 6, 1982) at 9, attached to BN-82-93, note 6, supra (hereinafter referred to as EG&G letter). EG&G is a research organization that is conducting core cooling tests for the NRC at the Semiscale facility.

regulations, and although the staff has never concluded that all plants with installed HPI and safety-relief systems can successfully 'feed and bleed,' we believe that there is an inherent margin of safety attributable to a feed and bleed capability." 75/

This statement appears to be inconsistent with the testimony of staff and licensee witnesses that feed and bleed is needed in certain situations. 76/ While in general

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75/ Memorandum from Roger J. Mattson to Darrell Eisenhut (Aug. 30, 1982) at 1, attached to BN-82-93, note 6, supra.

76/ The following are examples of testimony by staff and licensee witnesses that implies dependence upon feed and bleed in the event of a main feedwater transient or a small break loss of coolant accident:

> Staff witness Jensen agreed that, assuming no emergency feedwater, there are certain scenarios in which feed and bleed is relied on in order to meet 10 CFR §50.46. Tr. 5587.
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> Licensee witness Keaten testified that "in a supplement to the FSAR there is a specific discussion of the fact that if the emergency feedwater system is not available, that the core can be adequately cooled by the feed and bleed cooling mode." Tr. 7806.
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> Staff witness Curry indicated that the probability of core damage must take into consideration the reliability of both the emergency feedwater system and the feed and bleed option. Tr. 16,723-24.
>
> Staff witness Wermiel testified that "when we look at the emergency feedwater system for mitigating feedwater transients and the scenarios that could get you to core melt, we recognized that there is a feed and bleed backup capability to the system." Tr. 16,734.

(FOOTNOTE CONTINUED ON NEXT PAGE)

the staff and licensees may not rely upon feed and bleed to meet the regulations, the effectiveness of feed and bleed is of special significance in this proceeding, because of the testimony presented and the Licensing Board's findings.

On October 22, 1982, the staff provided us with a second EG&G report of two Semiscale tests of feed and bleed and the staff's analysis of the results in Board Notification BN-82-107. The first test, S-SR-1, was performed using "high head" HPI pumps similar to those at TMI-1. This test was terminated as a result of "operational problems with uncontrolled coolant leakage."77/ Semiscale

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76/ (FOOTNOTE CONTINUED FROM PREVIOUS PAGE)

> Staff witness Wermiel stated that feed and bleed was part of the backup in the interim to compensate for the lack of safety-grade emergency feedwater automatic initiation. Tr. 16,846-47, 16,869-70. We understand that the staff considers automatic initiation to include control of the EFW flow. See Tr. 17,014-15 (Wermiel).

The staff also appears to rely upon feed and bleed in the event of a main steam line break:

> Staff witness Wermiel testified that "in the case of the steam line break, for example, we do have our feed and bleed backup." Tr. 6126.
>
> Staff witness Wermiel agreed that the staff is relying on feed and bleed to cool the core in the event of a main steam line break in the interim until the emergency feedwater system is fully safety-grade. Tr. 6200-01.

77/ EGG-SEMI-6022, "Analysis of Primary Feed and Bleed Cooling in PWR Systems" (September 1982) at 20, 22, attached to BN-82-107, note 6, supra (hereinafter referred to as EG&G Report).

test S-SR-2, which used "low head" HPI pumps, resulted in excessive heating of the core simulator. The report concluded that feed and bleed appears feasible "but its viability depends on plant-specific characteristics and postulated scenarios."78/ As we indicated in our November 5, 1982 memorandum and order (at 6), however, we believe that these tests raise questions about the viability of the feed and bleed option at TMI-1.

In its response to our order, UCS indicates its agreement with that view but provides no comments beyond those it already made in response to the Board Notifications and in reply to the other parties' response. 79/ In its response to Board Notification BN-82-93, UCS noted that one conclusion of the EG&G letter is that feed and bleed is theoretically possible only within a certain band of primary system pressure. 80/ UCS asserts that the record contains no evidence that an analysis was performed to demonstrate that such a pressure band exists for TMI-1. 81/ The

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78/ Id. at 111.

79/ See UCS Response at 1. See generally UCS Response to Board Notification BN-82-93 (October 7, 1982); UCS Reply to our Order of October 15, 1982 (October 29, 1982).

80/ UCS Response to Board Notification BN-82-93 at 7. See EG&G letter at 2-3.

81/ UCS Response to Board Notification BN-82-93 at 7-8.

licensee, in its reply to the UCS response, explains that there is not a concern at TMI-1 for maneuvering the plant into a certain pressure band because the high head HPI pumps can provide cooling flow up to the safety relief valve setpoints. 82/ We agree that the existence of high head HPI pumps at TMI-1 appears to remove the concern for a feasible feed and bleed pressure band. We nevertheless believe that a plant-specific analysis of feed and bleed must be provided. Such an analysis should address the possibility noted by UCS that two-phase flow through the safety relief valves might affect the ability to feed and bleed successfully. 83/

UCS also filed and briefed several exceptions concerning the feed and bleed mode of decay heat removal. 84/ Only some of those arguments are of concern to us now; the rest will be discussed in detail in our final decision on design issues.

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82/ Licensee's Reply to UCS Response to Board Notification BN-82-93 (October 25, 1982) at 3-5.

83/ See UCS Response to Board Notification BN-82-93 at 8.

84/ See UCS Brief at 2-3, 9-13, 15, 18-19, 21-24, 41, 44, 103-04, 106-08.

UCS asserts that feed and bleed "is an untested, unverified cooling mode which depends on operator action and a complex decision process." 85/ UCS also maintains that the Licensing Board misplaced the burden of proof by finding that it "has not been shown to be an unacceptable way of cooling the core." LBP-81-59, supra, 14 NRC at 1269-70. 86/ Finally, UCS argues that the safety relief valves are not qualified to perform the "bleed" function during feed and bleed and that the power operated relief valve (PORV) would be needed to lower primary system pressure during a steam generator tube break accident. 87/

The licensee and staff maintain that the record is sufficient to demonstrate feed and bleed capability at TMI-1. They also argue that the recent Semiscale tests do not challenge the viability of that process. 88/

The licensee asserts that an event which occurred on February 26, 1980 at the Crystal River facility demonstrated the operability of feed and bleed. 89/ See Jones, fol. Tr.

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85/ Id. at 3.

86/ Id. at 9.

87/ Id. at 21-24.

88/ See Licensee Response at 26-27, 31; Staff Response at 9.

89/ See Licensee Response at 29-30.

4589, at 3-4; Jensen (UCS Contention 1), fol. Tr. 4913, at 9-10. The record indicates, however, that this event was not a demonstration of feed and bleed over an extended period because emergency feedwater was restored within 20 minutes. Tr. 5011-12 (Jensen).

As part of its effort to investigate feed and bleed, EG&G performed an analysis of the Semiscale test S-SR-2 using the "RELAP5" computer code to determine whether the code could predict the test phenomena. [90/] In response to our November 5, 1982 memorandum and order, the staff discusses the discrepancies that were found between the code and the test for the primary coolant inventory. [91/] The staff indicated that EG&G will perform the calculations with corrected HPI flow characteristics and expects this change to provide better agreement between the code and test results. [92/] The staff also described a feed and bleed analysis using the RELAP5 code for the Midland plant. [93/] With only one HPI pump available and the safety relief valves performing the "bleed" function, the analysis

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90/ See EG&G report at Section 5.

91/ Affidavit of Brian W. Sheron (Nov. 22, 1982) at ¶¶15-17, attached to Staff Response.

92/ *Id.* at ¶15.

93/ *Id.* at ¶18.

predicted that the core would be adequately cooled. 94/ This sort of demonstration might also be possible for TMI-1. 95/ We would be prepared to conclude that feed and bleed has been adequately demonstrated for TMI-1, if (1) the re-analysis of the S-SR-2 test demonstrates the capability of the RELAP5 computer code to predict the feed and bleed phenomenon, and (2) the code predicts that feed and bleed will successfully provide core cooling using actual TMI-1 plant parameters.

Conclusion

A. Information

As we indicated in the foregoing analysis, we believe that the existing record is unclear as to whether adequate core decay heat removal can be assured for TMI-1 in the event of a loss of main feedwater or a small break loss of coolant accident. Therefore, a limited reopening of the record is necessary to clarify this matter. We have determined that supplemental testimony is required in the following areas:

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94/ Id.

95/ The staff indicated that the Midland plant is designed with a core power level that is five percent lower than that for TMI-1. The licensee's computer analyses have indicated that omission of the American Nuclear Society's factor of 1.2 for core decay heat would result in the need for only one HPI pump to provide adequate core cooling. See generally Lic. Ex. 9. Therefore, we are concerned that the five percent difference in power level might affect the success of feed and bleed at TMI-1.

1. The exact size and flow rate of the vents to be installed in the hot legs (from the licensee).
2. When and under what conditions such size vents would or would not be useful to promote liquid natural circulation, including reasons for the conclusions reached (from the staff).
3. The current status of the hot leg vent installation (from the licensee).
4. Whether the modified B&W ECCS evaluation model for small breaks that predicts the boiler-condenser process is an NRC approved code under Appendix K to 10 CFR Part 50 (from the staff).
5. Whether the staff has reviewed the B&W Appendix K model to determine the ability of the code to calculate the effects of small breaks, including reliance upon boiler-condenser circulation (from the staff).
6. Whether only breaks slightly smaller than 0.07 $ft^2$ must be analyzed (from the staff).
7. Confirmation (such as by means of detailed computational analysis or experimental testing) that boiler-condenser circulation flow will transport sufficient core decay heat to the steam generators to prevent core damage (from the licensee and the staff).
8. Clarification of the apparent inconsistencies and confusion concerning the safety-grade status of

components in the EFW system (from the licensee and the staff).

9. Whether and under what circumstances reliance on feed and bleed is necessary at TMI-1 (from the licensee and the staff).

10. Results of the effort by EG&G to demonstrate the ability of the RELAP5 computer code to predict the results of Semiscale test S-SR-2 (from the staff).

11. Results of a RELAP5-type analysis to determine whether feed and bleed will successfully provide core cooling at TMI-1 (from the staff).

Although we direct the presentation of testimony by only the licensee and the staff on selected issues as indicated above, any party may offer testimony on any of the matters listed. (UCS may file written testimony in accordance with the schedule below if it wishes to present its own witnesses rather than rely upon cross-examination.)

B. Procedure

We intend to proceed promptly to supplement the record and to complete the appellate process in this phase of the case. All supplemental written testimony shall be in our hands and in the hands of other parties no later than the close of business, Wednesday, January 26, 1983.

The evidentiary hearing will be held in the NRC Public Hearing Room, Fifth Floor, East-West Towers Building, 4350

East-West Highway, Bethesda, Maryland, at 9:00 a.m. on Tuesday, February 8, 1983. We expect to complete the hearing within a day or two. Parties will be afforded an opportunity to file briefs, which shall include any proposed findings of fact or conclusions of law that they wish us to make. Briefs shall be in our hands by no later than the close of business Monday, February 28, 1983.

It is so ORDERED.

FOR THE APPEAL BOARD

Barbara A. Tompkins

Barbara A. Tompkins
Secretary to the
Appeal Board