UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

DOCKETED

BEFORE THE ATOMIC SAFETY AND LICENSING APPEAL BOARD

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

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CERTIFICATE OF SERVICE

I hereby certify that copies of "Licensee's Brief on the Reopened Hearing (Design Issues)" were served this 12th day of April, 1983, by hand delivery upon the parties identified by an asterisk and by deposit in the U.S. mail, first class postage prepaid, to the other parties on the attached Service List.

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

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LICENSEE'S BRIEF ON THE REOPENED HEARING (DESIGN ISSUES)

SHAW, PITTMAN, POTTS & TROWBRIDGE

George F. Trowbridge, P.C.
Thomas A. Baxter, P.C.

Counsel for Licensee

April 12, 1983

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LICENSEE'S BRIEF ON THE REOPENED HEARING (DESIGN ISSUES)

I. INTRODUCTION

On December 29, 1982, the Atomic Safety and Licensing Appeal Board reviewing the Atomic Safety and Licensing Board's decision on plant design and procedures issues in this proceeding,1/ issued a Memorandum and Order directing a limited reopening of the record to receive additional evidence on whether adequate core decay heat removal can be assured for

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

TMI-1 in the event of a loss of main feedwater or a small break loss of coolant accident (small break LOCA). Metropolitan Edison Company, et al. (Three Mile Island Nuclear Station, Unit No. 1), ALAB-708, 16 N.R.C. ___ (Dec. 29, 1982). The Appeal Board provided that the parties2/ would be afforded an opportunity to file briefs on a single date after the completion of the reopened hearing, which briefs shall include any proposed findings of fact or conclusions of law that the parties wish the Appeal Board to make. Id., slip op. at 45.

Licensee herein submits its brief in accordance with the schedule established by the Appeal Board at the hearing. Tr. 790. While this brief is not cast precisely in the form of proposed findings of fact and conclusions of law, Licensee requests that the Appeal Board decide the material issues of fact and law as discussed herein.3/ In an effort not to burden this particular portion of the appellate review process, however, Licensee has attempted to avoid here wholesale

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2/ The parties to the appeals on plant design and procedures issues are Licensee, the NRC Staff, and intervenor Union of Concerned Scientists ("UCS"). The Appeal Board also granted the unopposed request of the Commonwealth of Pennsylvania, which participated in the hearings before the Licensing Board under the provisions of 10 C.F.R. § 2.715(c), to participate in the reopened hearings. Tr. 4.

3/ Licensee's brief includes exact citations to the transcript of record and exhibits in support of its discussion of the evidence, as contemplated by 10 C.F.R. § 2.754(c). We simply have not employed numbered paragraphs or cast the brief in the language of the trier of the issues.

repetition of matters which have been well covered in previous briefs on design issues and which, based upon the discussion in ALAB-708, have been addressed adequately for the Appeal Board's review.

Before addressing the evidence presented on the specific issues raised in ALAB-708, Licensee will discuss generallly the scope of the reopened hearing, which was the subject of dispute among the parties both before and during the hearing.

In its 45-page Memorandum and Order of December 29, 1982 (ALAB-708), the Appeal Board identified the bases for its belief that the record compiled before the Licensing Board 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. The Appeal Board ". . . concluded, therefore, that a limited reopening of the record is required to facilitate our prompt resolution of these matters." ALAB-708, supra, slip op. at 3; see also id. at 42. Eleven requests for supplemental testimony were delineated by the Appeal Board to address its concerns with the record before the Licensing Board. Id. at 42-44. The scope of the reopened proceeding is limited, then, to the scope of the direct testimony requested by the Appeal Board.4/

4/ The eleven areas for which the Appeal Board determined that supplemental testimony is required are identified in

(Continued Next Page)

On numerous occasions before and during the reopened hearing, UCS attempted to add new issues to the reopened proceeding. In a motion filed on January 19, 1983, UCS asked the Appeal Board, among other things, to expand the scope of the reopened proceeding and to seek certain additional information from the Staff and Licensee. The Appeal Board found no need to expand the issues because its existing questions were adequate to elicit the information the Appeal Board needed. In short, the Appeal Board found, in its Order of January 26, 1983, that no evidence beyond that called for in ALAB-708 need be submitted.

In a subsequent ruling on a UCS motion for prehearing discovery, the Appeal Board observed that "[t]he purpose of the reopened hearing is to receive evidence in response to Board questions," and that the proceeding was ". . . reopened primarily to clarify issues already litigated." Order (unpublished), at 2, 4 (Feb. 7, 1983).

(Continued)

ALAB-708, supra, slip op. at 43, 44. Only Licensee and the NRC Staff presented direct testimony. Licensee responded to Issues 1 through 9, and the Staff addressed Issues 2 and 4 through 11. (Eight of the eleven issues were designated specifically as being addressed to one party (either the Staff or Licensee), leaving the other party with the option of responding.) Written testimony was filed on February 16, 1983, and sessions of the reopened hearing were held on March 7, 8, 16 and 17, 1983.

In a Memorandum and Order ruling on a UCS request for the issuance of subpoenas, the Appeal Board again stated its determination not to swerve from the concept of the limited reopening called for in ALAB-708:

> UCS characterizes the "heart of the issue" before us as "the adequacy of decay heat removal." Such characterization is too broad. The reopened hearing will not examine all aspects of decay heat removal but simply those discrete matters -- not including the reliability of high pressure injection -- raised in ALAB-708.

Metropolitan Edison Company, et al. (Three Mile Island Nuclear Station, Unit No. 1), ALAB-715, 17 N.R.C. ___, slip op. at 8 n.7 (Feb. 28, 1983).

During the hearing itself there were numerous occasions on which UCS attempted to pose questions on cross-examination, or to introduce documents into evidence, on matters outside the scope of the reopened proceeding and, on occasion, outside the scope of the TMI-1 Restart proceeding altogether.5/ Licensee submits that these Appeal Board rulings on the metes and bounds of its reopening of the record are correct.

The reopened proceeding is a limited one for good reason. There previously had been a lengthy and exhaustive

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5/ See infra pp. 53-55 (seismic qualification of the emergency feedwater system).

adjudicatory proceeding on plant design and procedures issues. The Appeal Board has before it the Licensing Board's Initial Decision and the substantial underlying record, the briefs of and oral arguments by the parties on appeal, and the parties' views, filed on November 22, 1982, on the Appeal Board's preliminary views and concerns, expressed in its Memorandum and Order of November 5, 1982. The Appeal Board has not issued a decision yet on any of the design issues and clearly was in the position to define its own scope for, and properly limit, the reopened hearing to those matters necessary for the Appeal Board to reach a final decision.

While affording interested parties the opportunity to participate fully in the reopened hearing, as UCS did with lengthy cross-examination, the Appeal Board was not required to provide UCS with a renewed opportunity to pursue issues of its own, even if arguably related to the subject of the reopened proceeding, which were fully addressed, or which UCS had the opportunity to address, in the hearings before the Licensing Board. The Appeal Board correctly did not issue a broad invitation for parties to advance evidence with respect to any matters which arguably relate to decay heat removal capability at TMI-1. Requests to add to the scope of the reopened proceeding constitute motions to reopen the record which must be judged by the applicable standards. See Kansas Gas and

Electric Company et al. (Wolf Creek Generating Station, Unit No. 1) ALAB-462, 7 N.R.C. 320, 338 (1978). UCS at no time even attempted to make the showing necessary to meet those standards.

Turning to the evidence presented in the reopened proceeding, Licensee has organized its brief into the following four subjects, each of which address one or more of the eleven areas designated in ALAB-708: (1) hot leg high point vents (Issues 1-3); (2) boiler-condenser cooling (Issues 4-7); feed and bleed cooling (Issues 9-11), and emergency feedwater system status (Issue 8).

II. HOT LEG HIGH POINT VENTS

In ALAB-708, the Appeal Board observed from the record that in the event of a small break loss of coolant accident or a main feedwater transient at TMI-1, reactor core decay heat may be removed to the steam generators using the emergency feedwater system through liquid natural circulation or the boiler-condenser process. Because liquid natural circulation may be interrupted by steam formation for any break in the reactor coolant system larger than about 0.005 ft^2 (if only one HPI pump is operating), and because for breaks of 0.02 ft^2 and smaller the energy discharged through the break is not sufficient to remove decay heat, there is a range

of breaks for which the boiler-condenser cooling mode may be required. ALAB-708, supra, slip op. at 4-5, 15-16; LBP-81-59, supra, 14 N.R.C. at 1227 (¶607); Jones and Lanese, ff. Tr. 53, at 2.

Because the Appeal Board had concerns about the adequacy of the record before the Licensing Board to demonstrate the viability of boiler-condenser cooling, the Appeal Board inquired, in its Memorandum and Order of November 5, 1982, whether the high point vents to be installed in the reactor coolant system hot legs could be used to eject the steam collected at the system high points and thereby restore liquid natural circulation. The parties agreed that the capability of these vents to remove steam from the high points of the hot legs sufficiently to re-establish liquid natural circulation is not demonstrated on the record before the Licensing Board. ALAB-708, supra, slip op. at 16.

The Commission already had required, in its hydrogen control regulations, the installation of high point vents at all light water reactors by the end of the first scheduled outage beginning after July 1, 1982, and of sufficient duration to permit required modifications. See 10 C.F.R. § 50.44(c)(3)(iii). Every indication is that the Commission has required installation of these vents to provide a means of venting noncondensible gases from high points in the primary

system. See Interim Requirements Related to Hydrogen Control and Certain Degraded Core Considerations (Proposed Rule), 45 Fed. Reg. 65466, 65468 (1980); Licensing Requirements for Pending Operating License Applications (Proposed Rule), 46 Fed. Reg. 26491, 26497 (1981); Interim Requirements Related to Hydrogen Control, 46 Fed. Reg 58484 (1981); NUREG-0737 at 3-56.

The Appeal Board has taken the position that while the Commission has required the installation of high point vents in connection with hydrogen control, it is not clear that the only permissible use for the vents is the removal of noncondensible gases. ALAB-708, supra, slip op. at 20. Consequently, the Appeal Board requested testimony on the following subjects as a part of the reopened proceeding:

> 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).

Id. at 43.

The uncontradicted testimony presented by Licensee shows that the vent flow path from each hot leg consists of nominal half-inch diameter piping, two solenoid-operated

valves, one manually-operated valve and a flow detecting orifice. The piping has an internal diameter of 0.464 inches and an outside diameter of 0.840 inches, and the manual valve has a minimum internal diameter of 0.562 inches. The orifice opening is 0.371 inches in diameter. The vent path is sized such that it will pass 7.6 pounds per second of subcooled water at design conditions of 2500 psig and 600°F. The sizing follows the guidance in NUREG-0737 and ensures that failure of the vent line would not result in a loss of coolant accident as defined in Appendix A to 10 C.F.R. Part 50. The capability of the vents to remove saturated steam varies with reactor coolant system pressure. For perspective, however, that capability per vent is less than 1 lbm/sec at 1000 psig, and is 2 lbm/sec at 2000 psig. Dempsey, ff. Tr. 53.6/

The small size of the hot leg high point vents was pivotal to the positions of the witnesses presented by the Staff and Licensee on the usefulness of this vent system to restore liquid natural circulation by relieving steam. Opening

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6/ The testimony physically incorporated in the transcript, as it was filed, is entitled "Licensee's Testimony of Gary R. Capodanno in Response to ALAB-708 Issue No. 1 (Hot Leg High Point Vent Sizing)." Mr. Dempsey appeared in a substitute capacity in place of Mr. Capodanno, who was ill at the time of the hearing. Mr. Dempsey adopted the testimony as his own, and the record shows that he is fully qualified to sponsor it. Tr. 46-50.

of the high point vents, as a means of recovering natural circulation, has been examined at various points in the sequence of reactor coolant system response for the break sizes of interest -- between 0.005 and 0.02 ft^2. Opening the vents during the two-phase natural circulation period of the transient, the earliest situation of interest here, could be useful if by doing so the depressurization rate of the primary system was materially increased, thereby aiding high pressure injection flow. Opening of the vents when the system is in two-phase natural circulation would provide an additional energy removal path from the reactor coolant system, and would lead to some increase in the depressurization rate. Since the primary system is saturated during this phase of the transient, however, the liquid in the system would flash, retarding the depressurization rate. Additionally, because of the small size of the vent, which is the equivalent of only a 0.00085 ft^2 break in the reactor coolant system, the addition to the depressurization rate during the two-phase natural circulation stage would be small in any event. Jones and Lanese, ff. Tr. 53, at 3, 4. This steam produced by flashing would rise to the top of the hot leg U-Bend to replace the steam being vented. Liquid as well as steam might be expelled when the vents were opened and would further deplete reactor coolant. Sheron and Jensen, ff. Tr. 83, at 3. Obviously, opening the vents here would be of limited or no value.

Opening of the vents after natural circulation is lost was also examined. Since the steam flow through the vents (approximately 3 lbm/sec total) is only 4 percent of the steam production rate from the core at one-half hour, for example, opening of the vents after natural circulation is lost also would not result in recovery of natural circulation. Jones and Lanese, ff. Tr. 53, at 4.

The Staff witnesses indicated that during the recovery period of a small break LOCA in which the primary system is refilling with subcooled water, or during the recovery from any transient event which results in steam bubble formation at the top of the hot legs and in which the primary system is refilling with subcooled water, opening the high point vents would aid in the recovery of liquid natural circulation. Sheron and Jensen, ff. Tr. 83, at 2, 3; see also Jones, ff. Tr. 53, at 4, 5.

In summary, in the long term opening of the hot leg high point vents could provide a means of recovering the system inventory earlier, thereby reestablishing natural circulation.7/ Opening of the vents would provide virtually no

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7/ The B&W Owners Group in 1981 submitted to the Staff proposed guidelines for utilizing the hot leg high point vents. The guidelines addressed use of the vents during the refill phase of a small break LOCA. However, because of the marginal benefit to be derived from vent use in this situation, those portions of the guidelines were withdrawn from Staff considera-

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benefit, however, for recovering natural circulation during the early phases of a small break LOCA. Thus, the vents are not capable of replacing the role of the steam generators for small break loss of coolant accidents. Jones and Lanese, ff. Tr. 53, at 5.

The hot leg high point vents will be used at TMI-1 during situations of inadequate core cooling. Guidelines have been developed and included in the Abnormal Transient Operating Guidelines (ATOG) program and are under review by the NRC Staff. Id. Installation of the system is in progress, and the present schedule (which necessarily is adjusted as engineering is completed, materials receipt is finalized and as work progresses) indicates an earliest system operable date of May 21, 1983. Manganaro, ff. Tr. 53; Tr. 77 (Manganaro).

(Continued)

tion in April, 1982, after the B&W owners and the Staff agreed that certain questions raised about the guidelines could not be resolved without an extensive testing and analytical effort to demonstrate to the Staff that use of the vents under certain conditions would not be detrimental to plant safety. Jones and Lanese, ff. Tr. 53, at 5, 6; Tr. 59-62 (Lanese), 86 (Sheron). Cf. ALAB-708, supra, slip op. at 22-23 and nn. 39-40 (citing correspondence prior to April, 1982); Tr. 97-99 (Sheron).

III. BOILER-CONDENSER COOLING

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); Jones, ff. Tr. 453, at 12. It is for breaks in between these sizes that the boiler-condenser process, which B&W analyses predict will occur, may be required. Of course, this entire discussion assumes the availability of only one high pressure injection (HPI) train. If two HPI pumps are available, there are no small break LOCAs which require steam generator heat removal. Jones, ff. Tr. 453, at 11 n.1; Licensee Ex. 87 at 1-2, 3-4; see also ALAB-708, supra, slip op. at 32 n.68.

In the boiler-condenser process, which assumes continued availablitiy 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. ALAB-708, supra, slip op. at 5; Jones, ff. Tr. 453, at 13; Keaten and Jones ff. ASLB Tr. 4558, at 7; ASLB Tr.

4852-54 (Jones); Jensen (Natural Circulation), ff. ASLB Tr. 4913, at 6.8/

A. Reliance Upon NRC-Approved Analyses

The Appeal Board has expressed the preliminary view that the record before the Licensing Board was not adequate to demonstrate the ability of the boiler-condenser mode of natural circulation to remove enough decay heat to prevent core damage. See ALAB-708, supra, slip op. at 24. As a part of the exploration of its concerns, the Appeal Board posed three questions for the reopened hearings which appear to be aimed at whether the Appeal Board must and/or should rely on the results of the B&W analyses which predict the boiler-condenser process because they are the product of NRC-approved emergency core cooling system (ECCS) evaluation models under Appendix K to 10 C.F.R. Part 50. The first of these questions is:

> 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).

8/ Unless otherwise indicated, the abbreviation "Tr." refers to the transcript of the reopened hearing. The transcript of the hearings before the Licensing Board is cited as "ASLB Tr.".

Id. at 43. While the question, no doubt unintentionally, appears to equate "model" and "code," the terms are not synonymous. The computer codes are part of the evaluation model. *See* 10 C.F.R. § 50.46(c)(2).

The B&W analyses performed prior to the TMI-2 accident to demonstrate the conformance of TMI-1 to 10 C.F.R. § 50.46 used the NRC-approved B&W ECCS evaluation model and, for certain break sizes (*e.g.*, the 0.04 ft^2 break), the results of these analyses also exhibited the steam generator heat transfer characteristics associated with boiler-condenser cooling. Jones, ff. Tr. 453, at 2.

In the above question, "modified B&W ECCS evaluation model" apparently refers to the model used for some of the additional small-break analyses performed after the TMI-2 accident. The same CRAFT-2 computer code is used in both the NRC-approved ECCS evaluation model and in the modified model, and it is the approved Appendix K code used to predict system response for these breaks. Sheron and Jensen, ff. Tr. 83, at 4; Jones, ff. Tr. 453, at 3. The CRAFT-2 computer code contains the equations and assumptions for heat transfer, including heat transfer by the boiler-condenser process, between the reactor system and the steam generator. Sheron and Jensen, ff. Tr. 83, at 4. Therefore, according to the Chief of, and a Senior Nuclear Engineer in, the Staff's Reactor Systems Branch:

> . . .[T]he equations and assumptions dealing with the boiler condenser process which are utilized in the B&W modified evaluation model have been approved by the NRC under Appendix K to 10 CFR 50.

Id.

The model used in some of the post-TMI-2-accident analyses was modified, however, to add two control volumes (or nodes) in order to provide a more detailed examination of plant response under boiler-condenser conditions. The additional control volumes, one in each reactor coolant system loop, more explicitly represent the upper head, or plenum, region of each steam generator. The analytical impact of the addition of the control volumes was to allow for a more accurate representation of the formation of a steam bubble between the steam generator emergency feedwater injection point and the 180° U-bend in the top of each reactor coolant system hot leg. Jones, ff. Tr. 453, at 3; Licensee Ex. 5, § 6.2.4.2; see also Sheron and Jensen, ff. Tr. 83, at 4.

It is only because of the additional noding that the modified evaluation model is not considered NRC-approved, since the Staff originally approved a different nodal description. Licensee's Exhibit 5, however, presented analyses of small breaks utilizing both the modified B&W ECCS evaluation model and the NRC-approved B&W ECCS evaluation model. Both models predicted that the boiler-condenser process would be effective

in removing decay heat if a condensing surface were uncovered within the steam generators. Sheron and Jensen, ff. Tr. 83, at 4, 5.

In short, the narrow answer to the Appeal Board's question is "technically, no." However, the answer should not stop there and the Appeal Board should consider the evidence cited above, that: (1) the CRAFT-2 computer code used in the modified model was the NRC-approved code and includes the equations and assumptions dealing with the boiler-condenser process; (2) the addition of the two nodes, the only modification made to the model, can only be viewed as an improvement to the model's capability to predict the system response of interest here; and, (3) the question overlooks the fact that prior to the subject modification (in analyses performed both before and after the TMI-2 accident) the NRC-approved B&W ECCS evaluation model predicted the boiler-condenser process for breaks of 0.04 ft^2 (before the accident) and smaller (after the accident).

The next question asks:

> 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).

ALAB-708, <u>supra</u>, slip op. at 43.

The equations and assumptions dealing with heat transfer between the reactor system and the steam generators, including heat transfer by the boiler-condenser process, are contained in the CRAFT-2 computer code which is part of the B&W ECCS evaluation model. The B&W ECCS evaluation model and the CRAFT-2 code that is included in the model have been reviewed and approved by the NRC Staff. Following the TMI-2 accident, B&W performed a number of small break LOCA calculations for break sizes smaller than those which had been evaluated to demonstrate compliance with Appendix K to 10 C.F.R. Part 50. These calculations, which are well documented in the Licensing Board record, indicated that for certain small break sizes, heat removal by the boiler-condenser process would be required to remove decay heat from the reactor system. The calculations were performed to provide a basis for revisions to small break LOCA emergency procedures. The Staff did not re-review the equations and assumptions contained in the CRAFT-2 code at that time. The Staff did perform audit calculations of small breaks in B&W designed plants using the RELAP-4 computer code. These calculations are documented in NUREG-0565 (Board Ex. 4). As did the B&W analyses, RELAP-4 predicted the boiler-condenser mode of natural circulation to be effective in removing decay heat and providing continued core cooling. Sheron and Jensen, ff. Tr. 83, at 5, 6.

The Staff has concluded that the heat transfer mechanisms involved in the boiler-condenser process are adequate to remove decay heat from the reactor system and will prevent core uncovery if at least one train of ECCS is operable. This conclusion is based on both the B&W CRAFT-2 calculations and the RELAP-4 audit calculations, as well as the Staff's evaluations of the heat transfer mechanisms involved in the process and discussed in commonly available heat transfer texts. The Staff has evaluated the mechanism involved in the boiler-condenser heat transfer process, and has concluded that the condensing surface that would be available would be capable of removing all decay heat generated by the core if an adequate supply of feedwater were available. Id. at 6.

The third Appeal Board question directed at the extent to which reliance should be placed on the analyses documented in the Licensing Board record is:

> 6. Whether only breaks slightly smaller than 0.07 ft^2 must be analyzed (from the staff).

ALAB-708, supra, slip op. at 43.

The Commission's regulations, at 10 C.F.R. § 50.46, establish the criteria for an acceptable emergency core cooling system. Appendix K to 10 C.F.R. Part 50 sets forth the required and acceptable features of an evaluation model used to show compliance with 10 C.F.R. § 50.46. ECCS cooling

performance is to ". . . be calculated for a number of postulated loss of coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the entire spectrum of postulated loss of coolant accidents is covered." See 10 C.F.R. § 50.46(a)(1).

The purpose of analyzing a spectrum of breaks is to ensure that the worst break size is analyzed. For small break LOCAs, compliance with the requirement historically has been demonstrated by selecting a limited spectrum of break sizes to determine the break size which produces the maximum amount and duration of core uncovery, and hence the highest cladding temperatures, amount of oxidation, and other parameters. Sheron and Jensen, ff. Tr. 83, at 7.

B&W has calculated that a postulated small break size of 0.07 ft^2 produces the highest peak cladding temperature and the greatest amount of core uncovery. Id. The smallest break analyzed in the demonstration, prior to the TMI-2 accident, of TMI-1 conformance to 10 C.F.R. § 50.46 was of the size 0.04 ft^2. See Jones and Broughton, ff. ASLB Tr. 5038, at 12 (Table 1); Licensee Exs. 3 and 4.

Witness Jones of B&W explained fully the considerations which led to B&W's selection of the spectrum of small breaks to be evaluated pursuant to section 50.46, and why breaks smaller than 0.04 ft^2 do not need to be analyzed to

demonstrate the conformance of TMI-1 to section 50.46. See Jones, ff. Tr. 453, at 5-8. Essentially, very small breaks -- i.e., those smaller than 0.04 ft^2 -- are not required to be evaluated because they are bounded by the larger breaks. Id. at 7.

Therefore, while breaks smaller than the spectrum analyzed to demonstrate compliance with 10 C.F.R. § 50.46 may involve different system behavior (i.e., the repressurization cycle which is caused by the interruption of natural circulation), core cooling is dependent upon maintaining core coolant inventory. Regardless of the specific sequence of events during a very-small-break LOCA, before core uncovery can occur reactor coolant pressure will decrease to a point (approximately 1000 psig) where high pressure injection has been demonstrated to provide adequate core cooling for the maximum core decay heat level. Id. at 8, 9.

The additional small-break LOCA analyses performed after the TMI-2 accident provided further confirmation of the validity of the above described methodology. While these evaluations were for the purpose of providing an improved analytical basis for emergency operating procedures, rather than to demonstrate compliance with 10 C.F.R. § 50.46, several breaks smaller than the previously analyzed 0.04 ft^2 break were addressed. Specifically, breaks of 0.005 ft^2 and 0.01

ft^2 were evaluated. See Jones and Broughton, ff. ASLB Tr. 5038, at 6-7 and 17 (Table 6). These analyses for the 0.005 ft^2 and 0.01 ft^2 breaks would be sufficient to demonstrate conformance to 10 C.F.R. § 50.46 pursuant to Appendix K. The results indeed showed that, compared to the larger break sizes, an increased margin relative to core uncovery exists for the extended spectrum below 0.04 ft^2.9/ Jones, ff. Tr. 453, at 9.

The Staff explained the considerations on which it bases its confidence that core uncovery will not occur for breaks much less than about 0.07 ft^2:

(1) Analyses performed to date by B&W and the Staff's own contractors do not predict any core uncovery for breaks much less than about 0.07 ft^2.

(2) The relative elevation of the condensing surface in the steam generators with respect to the top of the core is such that an ample condensing surface for steam condensation and decay heat removal will be exposed before the primary system inventory would drop below the top of the

9/ Calculations have been performed to demonstrate that the necessary conditions have been established to assure that long-term cooling will be developed. Follow-on calculations from there are not necessary to meet the requirements of 10 C.F.R. § 50.46(b)(5). Tr. 473, 532-533 (Jones); Licensee Ex. 86 at 6-1, 6-2.

core. Because of the establishment of the condensing surface and consequential decay heat removal, the primary system will be depressurized so that safety injection flow can exceed break flow and replenish primary system inventory before the core can become uncovered.

(3) Vent valves which allow pressure equalization between the vessel upper plenum and downcomer assure that the liquid level in the core cannot be significantly mismatched with the liquid level in the steam generators.

Sheron and Jensen, ff. Tr. 83, at 8. The Staff confirmed yet again that it finds TMI-1 to be in compliance with 10 C.F.R. §50.46 and Appendix K. Tr. 701 (Sheron).

In conclusion, it is Licensee's position that the Appeal Board should rely on the B&W modeling work discussed above, which has been reviewed and audited by the Staff. Not only do these analyses carry the imprimatur of compliance with Commission regulations, they have been shown, through years of checking and re-analysis, to be conservative and technically sound.

B. Confirmatory Analysis and Experimental Testing

The Appeal Board has characterized its concern as ". . . 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." ALAB-708, supra, slip op. at 32. In the reopened proceeding, the Appeal Board requested:

> 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).

Id. at 43. In response, both Licensee and the NRC Staff presented at the reopened hearing additional computational analyses (beyond those in the Licensing Board record) to demonstrate decay heat removal for the break sizes of interest (those with some reliance upon the steam generators). Licensee also presented a discussion of available test data which supports its models and the boiler-condenser process.

Since the analyses which are documented in Licensee's Exhibit 5 were performed, and subsequent to the Licensing Board hearings, B&W has revised its ECCS evaluation model and the CRAFT-2 code in response to item II.K.3.30 of NUREG-0737. The revised evaluation model and code have been submitted to the NRC (in November, 1982) for review. Within the modified CRAFT-2 code, an upgraded steam generator model has been incorporated which includes heat transfer correlations specifically oriented to the boiler-condenser mode of cooling.10/

10/ The new ECCS evaluation model is fully described in Licensee's Exhibit 86. Other changes include a non-equilibrium

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A new analysis of a 0.01 ft^2 break has been performed using the latest model, and shows that boiler-condenser cooling is established. Extrapolation of the results demonstrate that adequate core cooling is maintained for breaks of the size for which boiler-condenser cooling is predicted to occur. Jones, ff. Tr. 453, at 14; Licensee Ex. 86, Appendix E; Tr. 522-525 (Jones).

This new analysis shows different system behavior from that predicted by the old model. Sheron and Jensen, ff. Tr. 83, at 12. The basic difference involves steam generator behavior. However, the overall phenomenon of system repressurization due to loss of natural circulation is exhibited by both steam generator models. Licensee Ex. 86 at E-5. Comparison of the steam generator heat removal rates calculated in the analyses with the presently approved CRAFT-2 code (Licensee's Exhibit 5) to that which would be obtained by using the theoretical formulations in the new model show reasonable agreement. That is, an approximate three-foot adjustment in the condensing length in the earlier anlaysis would yield the same heat transfer as predicted with the new model. This small

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pressurizer model, two-phase slip model, and a two-phase reactor coolant pump model. Tr. 465 (Jones).

loss of inventory, approximately ten percent of the available inventory above the top of the core, would not affect core cooling. Jones, ff. Tr. 453, at 13, 14.

Prior to the reopened hearing, the Staff raised a concern relevant to decay heat removal capability at TMI-1 and, in particular, to the adequancy of B&W's model as it predicts successful boiler-condenser cooling. The concern, according to the Staff, was ". . . the ability to establish an effective condensing surface at the elevation of the auxiliary feedwater sparger ring in light of new data which shows limited penetration into the tube bundle of feedwater entering the steam generator from the emergency feedwater sparger ring." Board Notification 83-21 (Feb. 18, 1983), Enclosure at 1.

Licensee fully responded at the reopened hearing to this concern raised by the Staff.11/ See Licensee Ex. 87. The

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11/ A second concern raised in BN-83-21 was with the adequacy of emergency operating procedures to assure that a sufficient condensing surface would be established in the steam generators under all design basis conditions for which decay heat removal by the steam generators was required. The Staff subsequently confirmed that the procedures are adequate. See BN-83-21A (March 11, 1983). Further, the concern for overcooling if the steam generator level is raised for non-LOCA events is not new. Tr. 700 (Jensen). While relevant to decay heat removal, the plant procedures were not a subject of the reopened proceeding because the Licensing Board record (which includes the B&W operating guidelines, the TMI-1 plant procedures, and the Staff's favorable safety evaluations of them) is complete and remains viable.

data to which the Staff referred was from tests conducted in 1978 at B&W's Alliance Research Center. Id. at 2-6 to 2-8. Results of those tests were documented and provided to the Staff in April, 1982. Id. at 2-1, 2-2. The data has been utilized in B&W's upgraded steam generator model.12/ Id. at 2-19 to 2-26.

Licensee described the support for emergency feedwater (EFW) spray effectiveness as employed in its small break LOCA analyses from: instrumented laboratory tests, visual laboratory tests, specifically instrumented in-plant tests, a plant transient benchmarked correlation, and review of data from the TMI-2 accident. The results of this re-evaluation demonstrate the capability of the steam generator to remove decay heat via EFW spray to provide adequate core cooling under small break LOCA conditions.13/ See Licensee Ex. 87, chapter 2.

Licensee also presented a new heat transfer analysis which confirms the capability of the steam generator to remove

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12/ The model only takes credit for wetting a maximum of 10% of the steam generator tubes. Tr. 476, 479 (Jones).

13/ The Staff concluded that: "Analyses by B&W show that after accounting for the reduced EFW penetration into the steam generator tube bundle, and after accounting for plugged tubes in the TMI-1 steam generators, the EFW spray cooling will still provide effective decay heat removal." BN-83-21A at 11.

sufficient core decay heat during a small break loss of coolant accident. Since, during such an accident, decay heat removal rate determines the system pressure and, hence, the HPI flow being provided, to demonstrate adequate core cooling it is only necessary to show that sufficient decay heat removal is provided, prior to core uncovery, to allow the HPI system to replace the inventory being boiled off by the core decay heat removal. In this manner, coolant level in the core can be maintained above the top of active fuel rods. Jones, ff. Tr. 453, at 12, 14; see also ALAB-708, slip op. at 33.

The point of the analysis was to determine whether the boiler-condenser mode would assure a pressure/time relationship, before the core becomes uncovered, to yield adequate HPI to keep the core covered (i.e., injected flow greater than or equal to core boiling). The heat transfer analysis of the steam generator, while operating in the boiler-condenser mode, was performed to develop the pressure/time relationship. Prior to any possible uncovering of the core, the full condensing surface of the steam generator will be exposed.14/ Using this

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14/ "Full condensing surface" refers to the area above the overflow point of the reactor coolant pump. On the figure which follows Tr. 461, the relevant area is between points A and D for the steam generator level case, and between points A and B for the emergency feedwater spray case. Tr. 460-461 (Jones).

surface area, an analysis was performed to determine the reactor coolant system temperature, and hence pressure, as a function of time, that is necessary to condense all the steam being generated as a result of core decay heat removal. It should be noted that since none of the generated steam is assumed to be removed via the break, this analysis would over-predict the reactor coolant system pressure that could exist just prior to possible core uncovery. Results of the steam generator heat removal analysis for cooling on the steam generator level (at 95 percent on the operating range) and the emergency feedwater spray were shown.15/ Jones, ff. Tr. 453, at 15, figs. 2, 3.

Combining the results of the HPI cooling and steam generator heat removal analyses, it was shown that boiler-condenser heat removal will provide sufficient pressure control to result in HPI flows necessary to assure adequate core cooling after 1650 seconds. Id. at 15, 16, fig. 4; see also Licensee Ex. 87 at 3-5 to 3-10; Tr. 507-511 (Jones).

It is clear from numerous analyses that uncovering of the core would not occur prior to 1650 seconds for the break size range for which boiler-condenser heat removal is

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15/ The actual distribution of the plugged steam generator tubes at TMI-1 was used in the analysis. Tr. 535 (Jones).

necessary. In fact, the analyses show that the core cannot become uncovered prior to 3200 seconds for the break sizes of interest. Licensee Ex. 87 at 3-7. Since the boiler-condenser cooling mode assures adequate pressure control after 1650 seconds to enable the HPI to match or exceed the core boil-off, adequate core cooling is assured. Jones, ff. Tr. 453, at 16, 17. The Staff performed its own scoping heat transfer calculation, and concluded that an adequate fraction of the total steam generator surface area would be available to remove decay heat in the boiler-condenser mode. Sheron and Jensen, ff. Tr. 83, at 20, 21.

In response to the Appeal Board's question, the Staff directed its contractor, EG&G Idaho, in the performance of an analysis for TMI-1 of a 0.01 ft^2 break, using the RELAP-5 computer code. The results showed a different system response than did the B&W analysis results in Licensee's Exhibit 5. Boiler-condenser natural circulation was not calculated to be established, but rather, decay heat was removed by intermittent establishment of a bubbly, two-phase "chugging" type circulation.16/ Sheron and Jensen, ff. Tr. 83, at 9-17. The analysis showed that the core remained adequately cooled. Tr. 605 (Jensen).

16/ The Staff used different input assumptions than did B&W. See Sheron and Jensen, ff. Tr. 83, at 12.

The Staff performed a second calculation with RELAP-5 for TMI-1 in which conditions necessary for establishing steam generator heat removal in the boiler-condenser mode were imposed on the analysis through the scenario assumption. Based on this calculation, the Staff concluded that boiler-condenser natural circulation is an effective means for decay heat removal. Sheron and Jensen, ff. Tr. 83, at 17-20. As a demonstration of steam generator heat transfer capability, this hypothetical scenario nevertheless provides a valid analysis for the Appeal Board's inquiry.

There are several data sources available, or planned, which demonstrate the capability of the steam generator to remove decay heat in a boiler-condenser mode, as predicted by the analyses. First, there is the TMI-2 accident itself. After all of the reactor coolant pumps had been tripped at 100 minutes, filling of the steam generator by emergency feedwater commenced. During the fill period, heat removal from the reactor coolant system occurred which controlled the primary system pressure within 100 psi of the secondary side pressure. The only explanation for the pressure curves tracking together is the effect of boiler-condenser cooling in removing decay heat. See UCS Ex. 1 (minutes 100 to 125). If the HPI system had been actuated and maintained at this time, adequate inventory would have been maintained to prevent core damage.

Thus, the TMI-2 accident did not demonstrate an inadequacy of reactor coolant system heat removal (i.e., an inadequacy of boiler-condenser cooling), but rather showed the importance of maintaining adequate core inventory via the HPI. Jones, ff. Tr. 453, at 17, 18, Licensee Ex. 87 at 2-17, 2-18, fig. 2-16. Cf. ALAB-708, supra, slip op. at 31.

Tests have also been run at the Alliance Research Center (ARC) which examined condensation phenomena in a high pressure facility.17/ In these tests, a single steam generator tube was tested by exposing a condensing surface by adjusting water level on the inside surface of the tube. Then, by varying steam flow to the test section, temperature measurements were taken in order to determine the heat transfer coefficient. The calculated coefficients for these tests have confirmed the conservatism of the heat transfer model employed in the upgraded CRAFT-2 code. Jones, ff. Tr. 453, at 18.

In the future, additional experimental data on the boiler-condenser mode of cooling and small break LOCA response will be developed at ARC. At present, an integrated systems

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17/ The Staff testified that at present there are no experimental data from a test facility geometrically similar to the B&W reactor design confirming the boiler condenser mode of natural circulation. Sheron and Jensen, ff. Tr. 83, at 8. The Staff, however, apparently has not reviewed the results of these ARC tests.

test facility at ARC (GERDA) is being tested. It is a scaled single-loop, full height, full pressure test facility of a B&W NSS and is of similar size to Semiscale. This facility was developed for the BBR company in Germany in order to examine small break LOCA phenomena. The data from this facility is expected to be available in mid-1983. Id.; Sheron and Jensen, ff. Tr. 83, at 9; Tr. 62-63 (Jones).

The B&W Owner's Group, in conjunction with the NRC, is presently exploring a two-loop facility to further examine plant response to small break LOCA and other transients. This data will be used to provide additional confirmation of the adequacy of the computer models. Through the computer codes, this data will then enhance the understanding of plant response for improved operator training and procedures. Data from this facility is projected to be available in mid-1985. Jones, ff. Tr. 453, at 19; Sheron and Jensen, ff. Tr. 83, at 8, 9; Tr. 78-79 (Jones), 87, 103 (Sheron).

In conclusion, while there has not been experimental verification of code predictions of boiler-condenser cooling at a test facility geometrically similar to the B&W reactor design, that cooling mode has been predicted by B&W ECCS evaluation models which have been benchmarked and approved pursuant to Commission regulations, as well as by new and revised B&W ECCS evaluation models.18/ The new model has been

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18/ Data from test facilities cannot be applied directly to a large pressurized water reactor. Rather, it is used to

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benchmarked to demonstrate the code's capability to track the various modes of natural circulation observed during a small break loss of coolant accident. Licensee Ex. 86, Appendix G.

The various analyses in the record do not predict precisely the same system behavior for the breaks of interest here, and the Staff is pursuing, with the B&W owners, additional testing ". . . to satisfy the confirmatory research needs for the B&W design, and to provide additional confirmation of operating guidelines." Sheron and Jensen, ff. Tr. 83, at 9. As to adequate core cooling, however -- the subject of the Appeal Board's concern in the reopened hearing -- there is no question. It is clear that the boiler-condenser process is an extremely effective heat transfer mode. All of the analyses show that for the small breaks where boiler-condenser cooling may come into play, there are large margins to core uncovery, so that any uncertainties in the heat transfer predictions would not be expected to result in core uncovery. This has been confirmed by the new heat transfer analyses presented by Licensee and the Staff. It is the number and diversity of

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demonstrate the ability of a computer code to predict the relevant thermal hydraulic phenomena so that sufficient confidence can be gained that the code can be applied to predict the behavior of a large pressurized water reactor. Tr. 259-261 (Sheron).

these analyses, with different models and input assumptions, which provide confidence in the decay heat removal capability of the TMI-1 plant for small break loss of coolant accidents. See Tr. 537-538 (Jones); see also Tr. 753-754 (Ornstein).

IV. FEED AND BLEED COOLING

In its Memorandum and Order of November 5, 1982, the Appeal Board expressed its tentative view that the viability of the feed and bleed mode of core cooling had been called into question, based upon certain feed and bleed experiments performed at the Semiscale facility and subsequent pleadings filed by UCS, Licensee and the Staff concerning these tests. See Memorandum and Order (Nov. 5, 1982) at 1, 4, 10. Responses to the Appeal Board's preliminary findings and proposed requirements were filed by Licensee, the Staff and UCS on November 22, 1982. The Appeal Board subsequently found that the information provided by the Staff supported its "position that feed and bleed would provide adequate core cooling at TMI-1" and that "the existence of high head HPI pumps at TMI-1 appears to remove the concern for a feasible feed and bleed pressure band." ALAB-708, supra, slip op. at 8, 39. Nonetheless, the Appeal Board ordered a limited reopening of the record on the issue of feed and bleed cooling and requested the Staff and Licensee to submit testimony responsive to the following questions:

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).

Id. at 44. Based upon these questions and upon the scope of the examination during the reopened hearing, Licensee perceives the Appeal Board's concerns to be focused upon the following issues: reliance on feed and bleed cooling; the mechanical capability to perform feed and bleed cooling;19/ and, detailed analytical proof of the ability of feed and bleed to provide adequate core cooling. Each of these issues will be addressed in turn.

19/ The Appeal Board had previously indicated its concern regarding the ability of the safety valves to pass two-phase flow. See ALAB-708, supra, slip op. at 39. Additionally, during the reopened hearing, the Board and UCS pursued the issue of the capability of the HPI pumps to successfully perform in a feed and bleed mode.

A. Reliance on Feed and Bleed Cooling

The Appeal Board, in ALAB-708, cited several instances of testimony presented below before the Licensing Board which it believed raised questions regarding the extent to which feed and bleed cooling is relied upon to mitigate design basis events. ALAB-708, supra, slip op. at 36, 37 n.76. Licensee has previously provided the Appeal Board with its review and analysis of the cited testimony and that explanation will not be repeated in full here. See "Licensee's Memorandum of Law Regarding ALAB-708 Issue No. 9 (Reliance on Feed and Bleed Cooling)", February 16, 1983. Licensee contends that the testimony cited by the Appeal Board is not inconsistent with the position of the Staff and Licensee regarding feed and bleed (discussed below), and we urge the Appeal Board to consider Licensee's Memorandum of Law in its determination of this issue.

Both Licensee and the Staff presented testimony in response to ALAB-708 Issue No. 9 which affirmed their previous positions that feed and bleed cooling is required only for those beyond-design-basis events involving an extended loss of both main and emergency feedwater. Jones and Lanese, ff. Tr. 111, at 2; Sheron and Jensen, ff. Tr. 83, at 22. Neither Licensee witness Jones nor Staff witness Sheron could identify

any design-basis events, within the scope of the Restart proceeding, for which feed and bleed would be required at TMI-1.20/ Tr. 112 (Jones); Tr. 201-202 (Sheron).

While neither Licensee nor the Staff rely on feed and bleed cooling to mitigate design-basis events, this cooling mode is recognized as an additional, backup method of providing forced cooling which could be utilized as a defense in depth procedure for events beyond the design basis. Jones and Lanese, ff. Tr. 111, at 1; Sheron and Jensen, ff. Tr. 83, at 22. As Dr. Sheron testified, the Staff would encourage the use of any available alternatives for providing core cooling and, indeed, would challenge an applicant for failing to provide procedural guidance on the use of such available alternatives. See Tr. 200-201 (Sheron). Thus, it is clear that the recognition of feed and bleed as an alternative cooling mode is a prudent action on the part of Licensee and the Staff, but does not imply that reliance is, or is required to be, placed on such an alternate method.

20/ UCS attempted to expand the scope of this reopened hearing to consider the need for feed and bleed in responding to events having no nexus to the TMI-2 accident (e.g., high energy line breaks and seismic events). See Tr. 112-136, 201-202. Licensee submits that consideration of such events would not be in accord with the long-recognized scope of the TMI-1 Restart proceeding, and that the Appeal Board properly limited the testimony to small break LOCAs and loss of main feedwater transients. See Metropolitan Edison Company, et al. (Three Mile Island Nuclear Station, Unit No. 1), ALAB-705, 16 N.R.C. ___, slip op. at 21-22 (Dec. 10, 1982) (nexus requirement).

B. Feed and Bleed Capability

Although the Appeal Board had stated that its primary concern with feed and bleed did not involve the reliability of the plant equipment, a question was raised regarding the ability of the safety valves to successfully pass two-phase flow. See ALAB-708, supra, slip op. at 34, 39. In response to this concern, Licensee presented testimony regarding the results of tests performed by the Electric Power Research Institute (EPRI)21/ on safety valves of the same model as are installed at TMI-1.

The EPRI program included thirty-one tests of the TMI-1 type safety valves. Of these, four tests of the valves' ability to pass liquid flow were conducted, utilizing the same inlet configuration and ring settings as at TMI-1.22/ Jones and Lanese, ff. Tr. 111, at 4; Tr. 137-138 (Lanese). Of these four tests, the valve flow rate met the test acceptance

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21/ The EPRI test program was reviewed during the Licensing Board hearings in connection with the litigation of former UCS Contention 6 and a related Board Question. See LBP-81-59, supra, 14 N.R.C. at 1377-1379 (¶¶ 1080-1083).

22/ The TMI-1 safety valve inlet piping has been changed from a long-inlet, loop seal arrangement to a short inlet configuration; and the ring setings have been revised to duplicate those used during the EPRI test program. Those modifications were undertaken in order to eliminate valve instabilities observed during the EPRI program. Jones and Lanese, ff. Tr. 111, at 4; Tr. 411-414 (Correa).

criteria for three cases. In the final liquid flow case, pressure in the test loop was not controlled by the valve, but the valve flow rate exceeded the requirements for controlling pressure and providing core cooling at TMI-1. In sum, then, these four tests showed the valve capable of performing its intended function without sustaining any damage. Jones and Lanese, ff. Tr. 111, at 4, 5; Tr. 365 (Correa).

It is Licensee's position that, beyond the four liquid tests discussed above, the EPRI test program as a whole provides more than adequate assurance of the safety valves' ability to perform during feed and bleed cooling. The bases for this position may be summarized as follows:

- The thirty-one tests were all performed with the same valve and constituted at least thirty-two openings and closings of the valve. Tr. 167 (Lanese).
- During five of the tests on the long-inlet configuration, the valve experienced approximately 1400 flutter or chatter cycles.23/

23/ It was this flutter/chatter phenomenon exhibited during the long inlet, loop-seal tests which led to the decision to modify the TMI-1 inlet piping configuration and the valve ring settings. Tr. 409-414 (Correa); see also supra n.22. UCS attempted to elicit testimony to the effect that the decision to revise the ring settings was made subsequent to the decision to modify the inlet piping, asserting that Licensee initially believed that the piping change alone would eliminate the

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Although the valve experienced some damage during the last of these tests (i.e., test number 26), it was still capable of responding to system pressure. Tr. 383-386 (Correa).

- The heat-up and cool-down cycles inherent in the EPRI test program resulted in the valve being exposed to thermal transients in excess of those which would be experienced during feed and bleed cooling. Tr. 432-436 (Correa).

In addition to the EPRI test program, the examination and testing of the Crystal River 3 safety valves following a transient24/ during which the feed and bleed cooling mode was utilized provides an added degree of confidence in the ability of the valves to perform in this mode. The Crystal River valve was removed, disassembled, reassembled and tested, exhibited no internal damage, was subjected to three steam set tests by EPRI, and was capable of performing repeated lifts (at 2392, 2388 and 2388 psig). Tr. 400-401, 425-429 (Correa).

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observed instabilities. It is not clear what relevance that hypothesis could have, if proven. In any case, UCS was unable to make this showing and Licensee's witnesses testified that these decisions were made over the same time frame. See, e.g., Tr. 139-144 (Lanese, Jones); Tr. 409-414 (Correa).

24/ See Jones, ff. ASLB Tr. 4588, at 3, 4, for a description of the Crystal River transient.

A second issue, raised by UCS, concerns the ability of the HPI pumps to perform, for extended time periods, at the system pressures associated with feed and bleed cooling. The Staff testified that the HPI pumps are capable of so performing, in that the pump shut-off head is considerably greater than the safety valve set pressure of 2500 psi.25/ Tr. 184-185 (Sheron). Further confidence in the ability of the pumps to perform during feed and bleed is provided by the fact that since one pump is in use continually for normal reactor coolant system makeup, the pumps are designed to run for an indefinite period at normal plant design pressure (approximately 2200 psi). The flow associated with normal design pressures would not differ considerably from those required for feed and bleed at 2500 psi. Tr. 194-197, 210-211 (Jensen); see also Licensee Ex. 1, Supp. 1, Pt. 3, response to Question 1.

In sum, the record shows that the safety valves and HPI pumps, along with those other components needed for feed and bleed cooling, will perform as required.

25/ The HPI shut-off pressure is 2900 psi. Tr. 695 (Jensen). Additionally, the pumps are safety-grade and are qualified to operate in a small break LOCA environment. Tr. 194 (Sheron).

C. Analyses of Feed and Bleed Cooling

As noted above, the recent Semiscale experiments of feed and bleed cooling contributed to the Appeal Board's concerns regarding this cooling mode. Part of this concern was due to the fact that test S-SR-2, which resulted in excessive heatup of the core simulator, was not properly predicted by the RELAP-5 analysis of this test. ALAB-708, supra, slip op. at 38, 41.

A second RELAP-5 analysis of Semiscale test S-SR-2 was performed by EG&G Idaho, Inc. (EG&G), in order to demonstrate the code's ability to predict experimental test results. Sheron and Jensen, ff. Tr. 83, at 22; see generally UCS Ex. 46. For this analysis, EG&G revised the code to model the HPI flow versus pressure function to agree with the actual flow values delivered during the test, and also modified the core power level in order to more accurately compensate for steam generator secondary side heat losses. UCS Ex. 46, cover letter at 1; Sheron and Jensen, ff. Tr. 83, at 23. EG&G then compared the results of the RELAP-5 analysis to the S-SR-2 test data and found that RELAP-5 predicted all the major phenomena observed during the test and, within experimental uncertainties, most RELAP-5 results were in agreement with the test data. UCS Ex. 46, report at 21; Sheron and Jensen, ff. Tr. 83, at 23; see

also Tr. 250-251 (Jensen). Thus, this reanalysis shows that the RELAP-5 code is capable of predicting as-observed experimental phenomena associated with feed and bleed cooling.

In order to provide the Appeal Board with additional detailed analytical calculations documenting the ability of feed and bleed to provide adequate core cooling at TMI-1,26/ the Staff requested EG&G to perform a TMI-1 specific feed and bleed analysis using the RELAP-5 code, and also contracted with Los Alamos National Laboratory (LANL) to undertake a similar analysis using the advanced TRAC computer code. Sheron and Jensen, ff. Tr. 83, at 33, 43; UCS Ex. 47.

The EG&G RELAP-5 analysis used the TMI-1 plant-specific HPI pump flow vs. pressure curve27/ and pressurizer geometry, and further assumed a safety valve relief capacity

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26/ In conjunction with the additional small-break analyses performed following the TMI-2 accident, B&W performed a CRAFT code calculation of a complete loss of feedwater transient, which is a bounding calculation for the TMI-1 class of plants. (This analysis assumed the availability of only one HPI train at 20 minutes and utilized a realistic decay heat value of 1.0 times the 1971 ANS standard.) The results of the analysis showed that, at 8900 seconds following initiation of the event, HPI flow exceeds core boil-off and the system begins to refill. The core remained covered throughout the transient and the 10 C.F.R. § 50.46 cladding temperature criteria were satisfied. See generally Licensee Ex. 9.

27/ Licensee provided EG&G with the HPI performance data as set forth in Licensee Exhibit 1, Supplement 1, Part 3, response to Question 1. See Tr. 321.

based upon the tested relief capacity for the Dresser-type safety valves used at TMI-1. Sheron and Jensen, ff. Tr. 83, at 36, 42; Tr. 276-278 (Jensen, Sheron). The analysis, which assumed the initiation of one HPI train at 20 minutes and utilized the draft 1973 ANS standard for decay heat, showed that:

- the safety valves relieve sufficient fluid to remove the decay heat;
- the collapsed liquid level remains well above the top of the core throughout the transient, thereby assuring core cooling;
- fuel cladding temperatures rise to only slightly above saturation temperatures; and,
- at approximately 9000 seconds, net mass loss from the system begins to decrease, thereby beginning system recovery.28/

Sheron and Jensen, ff. Tr. 83, at 35-36 and Figures 11-1 through 11-4. Clearly, this calculation shows that feed and bleed would provide adequate core cooling and decay heat removal at TMI-1. The Staff, however, undertook a further investigation of this analysis by examining the uncertainties

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28/ This data point compares well with the previous B&W analysis, which predicted system recovery at approximately 8900 seconds. See supra n.26.

in the HPI and safety valve flows and uncertainties in the code itself.29/ This examination uncovered no uncertainty figures which would result in an unfavorable conclusion regarding the ability of feed and bleed to provide sufficient core cooling.30/ See generally Sheron and Jensen, ff. Tr. 83, at 36-42.

Finally, a similar analysis was performed by LANL using the TRAC code.31/ Again, the calculation showed that the fuel temperatures would not rise significantly above saturation temperatures and, at about 8000 seconds, that the flow through the safety valves equalled the flow in from one HPI pump, and the net mass loss rate decreased to zero. Sheron and Jensen, ff. Tr. 83, at 43; UCS Ex. 47, report at 1.

29/ The Staff's review showed that uncertainties regarding liquid or two-phase flows through the safety valves is not a significant factor in determining the viability of feed and bleed cooling. The RELAP-5 analysis and the Semiscale test data show that, once the hot leg becomes voided, and steam enters the pressurizer surge line, the mass discharge through the valves quickly transitions to steam flow. Sheron and Jensen, ff. Tr. 83, at 40-42.

30/ This review included a 25% reduction in the minimum calculated system inventory from that calculated by RELAP-5. While this reduction would result in a collapsed liquid level two feet below the top of the core, the actual two-phase (boiling) level would remain well above the top of the core. Sheron and Jensen, ff. Tr. 83, at 42.

31/ The LANL calculations used an existing input deck (as modified by the TMI-1 HPI and safety valve characteristics) for an Oconee reactor -- which is essentially the same as TMI-1, but with a higher power level. Sheron and Jensen, ff. Tr. 83, at 43; UCS Ex. 47, report at 1; Tr. 228 (Jensen).

Based on the foregoing, the analytical data base is more than sufficient to support a finding by the Appeal Board that the feed and bleed mode will provide sufficient core cooling if called upon to do so. Further, the record before the Licensing Board and in this reopened proceeding establishes that feed and bleed cooling, while not relied upon to meet Commission regulations, is a viable backup cooling method.

V. EMERGENCY FEEDWATER SYSTEM STATUS

The scope of the Appeal Board's current inquiry into the TMI-1 emergency feedwater (EFW) system, as set forth in Issue No. 8, is limited solely to obtaining a

> [c]larification of the apparent inconsistencies and confusion concerning the safety-grade status of components in the EFW system.

ALAB-708, supra, slip op. at 43, 44. This issue arose out of conflicting information presented by Licensee and the Staff in response to a query by the Appeal Board regarding the presence of a safety-grade capability to manually control EFW flow.32/ See Memorandum and Order (Nov. 5, 1982) at 9 n.5.

The Appeal Board was therefore concerned that similar misunderstandings might exist with respect to the safety-grade

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32/ Staff witness Wermiel later testified that the Staff's response to the Appeal Board's November 5, 1982 question was incorrect in asserting that a safety-grade manual control capability exists at TMI-1. See Tr. 342, 343 (Wermiel).

status of other components within the EFW system. The Appeal Board made clear, however, that this limited reopening of the record would not encompass a far-reaching review of the sufficiency of the TMI-1 EFW system, noting that "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." ALAB-708, supra, slip op. at 7 n.5; see also id. at 13-15 (concern regarding manual control capability resolved). Thus, the Appeal Board did not request the presentation of testimony on any aspect of the TMI-1 EFW system beyond a verification of the previously reported safety-grade status of the system and a clarification of the safety classification of the new manual control stations. Written direct testimony responsive to the issues identified by the Appeal Board was presented by Licensee and the Staff. See generally Capodanno33/ and Chisholm, ff. Tr. 324; Wermiel, ff. Tr. 341.

The new manual EFW control stations provide an alternate capability to manually control EFW flow independent of the Integrated Control System (ICS) and are powered from an independent, battery-backed power supply (rather than from an ICS-derived power supply).34/ The manual control circuits

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33/ Due to the illness of Mr. Capodanno, Thomas M. Dempsey appeared at the hearing and adopted those portions of Licensee's written direct testimony sponsored by Mr. Capodanno.

34/ Should the ICS, operating in the automatic mode, prevent the provision of EFW flow to the steam generators, the control

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themselves are highly reliable industrial grade components, and no single failure in the control circuits will result in a loss of system function. Thus, while the manual control stations are highly reliable, they cannot be considered safety-grade in that a single failure in certain of the power supply distribution components can result in the failure of the control stations. Capodanno and Chisholm, ff. Tr. 324, at 3, 4; Tr. 343 (Wermiel).

Although the manual control stations themselves are not safety-grade, excluding seismic considerations35/ the EFW system function as a whole is safety-grade for purposes of responding to loss of main feedwater and small break LOCA transients. Capodanno and Chisholm, ff. Tr. 324, at 2; Tr. 359-360 (Wermiel). As we have explained, one of the attributes considered in determining whether a system function can be considered safety-grade is whether there is adequate time available to perform manual control functions. Capodanno and Chisholm, ff. Tr. 324, at 4. The Appeal Board has previously considered, and found satisfactory, the procedural requirements

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room operators also have the option of attempting to control EFW flow manually through the ICS in the HAND mode. See ASLB Tr. 7104-05 (Broughton).

35/ As we discuss in greater detail below (see infra pp. 53-55), the issue of the seismic qualification of the TMI-1 EFW system is outside the scope of this proceeding.

for assuring that local manual control can be taken of the EFW control valves. See ALAB-708, supra, slip op. at 13. Both the Staff and Licensee have testified -- notwithstanding UCS's attempt to obfuscate the record on this issue36/ -- that sufficient time exists to carry out this local control action. Capodanno and Chisholm, ff. Tr. 324, at 4, 5; Wermiel, ff. Tr. 341, at 3. Thus, it is clear that the non-safety-grade status of the manual control station circuitry does not undermine the Licensing Board's conclusion that, at restart, the TMI-1 EFW system will be safety-grade for small break LOCAs and loss of main feedwater transients. See LBP-81-59, supra, 14 N.R.C. at 1372 (¶ 1057).

36/ UCS's cross-examination of Licensee witnesses Dempsey and Chisholm focused almost exclusively on the procedures directing that an auxiliary operator take local control of the EFW control valves, the time available to perform this function, and whether the same operator would be responsible for performing other actions at the same time. See generally Tr. 329-338. Given the Appeal Board's resolution of its concern regarding manual control capability (ALAB-708, supra, slip op. at 13), it is not surprising that neither Mr. Dempsey nor Mr. Chisholm were in a position to discuss, in detail, these procedures or operator actions. As Mr. Chisholm testified, reliance is placed on the plant staff to determine whether sufficient time exists to perform a specified manual action. Tr. 337-338 (Chisholm, Dempsey). Licensee contends that, in view of the limited scope of the rehearing on the EFW issue and the Appeal Board's previous conclusions regarding manual control capability, any attempt by UCS to negate the conclusions reached by the Appeal Board because of alleged unanswered questions regarding local manual control capability must fail as well beyond the scope of this reopened proceeding.

Licensee performed a review of the underlying record on the safety-grade status of the TMI-1 EFW system at restart which assured that there had been no changes in the status of the EFW modifications reported to the Licensing Board and the Appeal Board. Further, that review uncovered no inconsistencies between Licensee's and the Staff's description of the safety-classification of the EFW system components -- beyond the inconsistency discussed above regarding the new manual control stations. Capodanno and Chisholm, ff. Tr. 324, at 2; see also supra n.32; Licensee's Response to the Atomic Safety and Licensing Appeal Board's Order of July 14, 1982 (Aug. 12, 1982), at 9-13.

A similar review was also performed by the Staff to determine which components of the EFW system, required to perform during design basis events, would not be safety-grade at restart.37/ Wermiel, ff. Tr. 341, at 1, 2. Two of the items described by the Staff -- the non-safety-grade status of the EFW flow control valves38/ and of the condensate storage

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37/ In Licensee's view, there was no possible confusion, and no need for additional testimony, on the facts that the TMI-1 EFW system will not be safety-grade at restart for all events, and that it will be modified to fully safety-grade status at the next refueling outage. See ALAB-708, supra, slip op. at 6, 11; LBP-81-59, supra, 14 N.R.C. at 1353-1375.

38/ Mr. Wermiel's testimony noted two aspects of the EFW flow control function which the Staff does not consider safety-grade: the ICS interconnection and the fact that the valves do not satisfy the single failure criteria for high energy line breaks (HELB) in the intermediate building. Wermiel, ff. Tr.

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tank level indication -- have always been recognized as long-term modifications which would not be completed at restart. See LBP-81-59, supra, 14 N.R.C. at 1363, 1364 (¶¶ 1036, 1037).

The third item raised by the Staff regarding the safety-grade status of the TMI-1 EFW system concerns the capability of that system to respond to seismic events. Wermiel, ff. Tr. 341, at 2. This issue was not explicitly considered below by the Licensing Board and has since been ruled by the Commission to be outside the scope of the TMI-1 Restart proceeding.39/ Metropolitan Edison Company (Three Mile Island Nuclear Station, Unit No. 1), CLI-83-5, 17 N.R.C. ___ (March 4, 1983), slip op. at 2; see also ALAB-708, supra, slip op. at 7 n.5.

The subject of the seismic capability of the TMI-1 EFW system was initially brought to the attention of the

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341, at 2. During the course of the hearing, the Appeal Board ruled that the capability of the EFW system to respond to a HELB scenario was beyond the scope of the reopened proceeding, which has been limited to consideration of small break LOCAs and main feedwater transients. See Tr. 115-117.

39/ Given these holdings, which Licensee fully supports, we hesitate to address the matter further, since it should play no role in the Appeal Board's final decision on plant design and procedures issues. Nevertheless, because Licensee anticipates that UCS will discuss the seismic qualification issue in its brief, we record here in abbreviated fashion the status as reflected on the record.

parties to this proceeding by Board Notification 82-118, dated November 22, 1982. This Notification reported the interim results of a review of the TMI-1 EFW system seismic capability (undertaken in response to the generic Multiplant Action C-14, "Seismic Qualification of Auxiliary Feedwater Systems", applicable to all operating pressurized water reactors), and identified a number of unresolved items regarding the TMI-1 EFW system. It is as a result of this review that the Staff has concluded that "EFW system function following a safe shutdown earthquake has not been demonstrated . . .". See Wermiel, ff. Tr. 341, at 2. The Staff is pursuing resolution of this issue at TMI-1 on a schedule consistent with that for all other operating reactors, and Licensee has committed to complete, by startup from the first refueling outage after restart, any necessary modifications to assure that the EFW system is capable of performing during a safe shutdown earthquake. Tr. 351 (Wermiel); Wermiel, ff. Tr. 341, at 2, 3.

From the above discussion, then, it is clear that the seismic capability of the TMI-1 EFW system is a generic issue which will be resolved by the Commission outside of the TMI-1 Restart proceeding. UCS, however, refused to accept this judgment and consistently attempted to raise the seismic capability issue during the reopened hearing, despite repeated rulings by the Appeal Board that this issue was outside the scope of the proceeding.40/ See, e.g., Tr. 117-136, 325-326,

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40/ The scope of the Restart Proceeding has been consistently limited to matters "having a reasonable nexus to the TMI-2

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350, 354-356; see also ALAB-708, supra, slip op. at 7 n.5. Notwithstanding these efforts by UCS to expand the scope of this proceeding and to confuse the record on the safety classification of the EFW system, both the Staff and Licensee are firm in their position that, for the events at issue in this proceeding (i.e., small break LOCAs and loss of main feedwater transients), the TMI-1 EFW system will be safety-grade at restart. Capodanno and Chisholm, ff. Tr. 324, at 2; Tr. 359-360 (Wermiel); see also Wermiel, ff. Tr. 341, at 3.

VI. CONCLUSION

For all of the foregoing reasons, the evidentiary record on plant design and procedures issues, as supplemented in this reopened proceeding, convincingly demonstrates the reliability of existing plant systems to provide adequate decay heat removal in the event of a main feedwater transient or small break loss of coolant accident at TMI-1.

Addressing its concerns with the Licensing Board record on decay heat removal capability, the Appeal Board stated:

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accident" -- a standard proposed by UCS and others. See Metropolitan Edison Company (Three Mile Island Nuclear Station, Unit 1), LBP-81-32, 14 N.R.C. 381, 394 (1981), and ALAB-705, 16 N.R.C. ____, slip op. at 21-22 (Dec. 10, 1982). It is patently obvious that a seismic event has no such nexus to the TMI-2 accident. See CLI-83-5, supra, slip op. at 2.

> 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.

ALAB-708, supra, slip op. at 10. Licensee and the Staff have proven, in this proceeding, the viability of both the boiler-condenser and feed and bleed cooling modes at TMI-1.

Respectfully submitted,

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Dated: April 12, 1983

APPENDIX A

WRITTEN DIRECT TESTIMONY RECEIVED INTO EVIDENCE

Witness	Following Transcript Page
Chisholm, Richard J.	
"Licensee's Testimony of Gary R. Capodanno and Richard J. Chisholm in Response to ALAB-708 Issue No. 8 (Safety-Grade Status of Emergency Feedwater System)"	324
Dempsey, Thomas M.1/	
"Licensee's Testimony of Gary R. Capodanno in Response to ALAB-708 Issue No. 1 (Hot Leg High Point Vent Sizing)"	53
"Licensee's Testimony of Gary R. Capodanno and Richard J. Chisholm in Response to ALAB-708 Issue No. 8 (Safety-Grade Status of Emergency Feedwater System)"	324
Jensen, Walter L., Jr.	
"NRC Staff Testimony of Brian W. Sheron and Walton L. Jensen, Jr. in Response to Appeal Board Questions 2, 4, 5, 6, 7, 9, 10 and 11"	83
Jones, Robert C., Jr.	
"Licensee's Testimony of Robert C. Jones, Jr. and Louis C. Lanese in Response to ALAB-708 Issue No. 2 (Use of Hot Leg Vents in Promoting Natural Circulation)"	53
"Licensee's Testimony of Robert C. Jones, Jr., and Louis C. Lanese in Response to ALAB-708 Issue No. 9 (Reliance on Feed and Bleed Cooling)"	111

1/ Due to the illness of Licensee's proposed witness Mr. Capodanno, Mr. Dempsey appeared at hearing and adopted those portions of Licensee's written direct testimony sponsored by Mr. Capodanno.

<u>DOCUMENTS INCORPORATED INTO THE RECORD</u>

APPENDIX B

EXHIBITS

EXHIBIT NUMBER	DESCRIPTION	IDENTIFIED AT TRANSCRIPT PAGE	ADMITTED AT TRANSCRIPT PAGE
Lic. Ex. 86	"B&W's Small-Break LOCA ECCS Evaluation Model", BAW-10154P, November 1982	451	457
Lic. Ex. 87	"Evaluation of SBLOCA Operating Procedures and Effectiveness of Emergency Feedwater Spray for B&W-Designed Operating NSSS" February 1983, B&W Doc. Id. 77-1141270-00	459	459 1/
UCS Ex. 45	Three Mile Island Unit 1 Emergency Procedure 1202-26A, Loss of Steam Generator Feed to Both Once-Through Steam Generators, Rev. 14, June 4, 1982	207	210
UCS Ex. 46	Letter dated January 14, 1983 from P. North, EG&G Idaho, Inc., to J.E. Solecki, DOE, and attached report PN-08-83, "Extension of Analysis of Primary Feed and Bleed Cooling in PWR Systems"	257	258
UCS Ex. 47	Letter dated February 8, 1983 from N. S. DeMuth, Los Alamos National Laboratory, to R. T. Curtis NRC, and attached report, "Feed-and-Bleed Calculations for TMI-1", by R. J. Henninger and N. S. DeMuth	291	291
UCS Ex. 48	Letter dated April 22, 1982 from T. A. Baxter to Atomic Safety and Licensing Appeal Board, with attachments	415	422

1/ Section 4 of Licensee Exhibit 87 was not offered or received into evidence.

EXHIBIT NUMBER	DESCRIPTION	IDENTIFIED AT TRANSCRIPT PAGE	ADMITTED AT TRANSCRIPT PAGE
UCS Ex. 49	Letter dated May 13, 1982, from T. A. Baxter to Atomic Safety and Licensing Appeal Board with attachments	416	422
UCS Ex. 50	Memorandum dated July 11, 1979 from B. W. Sheron to Z. R. Rosztoczy, "TMI-2 Turbine Overspeed Trip of 3/6/79"	565	REJECTED 584
UCS Ex. 51	Letter from D. G. Eisenhut to J. J. Mattimoe (received stamped April 1, 1982) and enclosure, "Staff Concerns with the B&W Small Break Model"	644	649
UCS Ex. 52	Memorandum dated October 25, 1982 from B.W. Sheron to R. Fraley, ACRS	680	
UCS Ex. 53	Memorandum dated June 2, 1982 from C. J. Heltemes, Jr. to G. Mazetis	746	785