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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING APPEAL BOARD

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

UNION OF CONCERNED SCIENTISTS' PROPOSED FINDINGS
OF FACT AND CONCLUSIONS OF LAW ON REOPENED HEARING

Introduction

1. 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. ALAB-708, at 1.

2. Based on our analysis of the record below and the parties' response to our unpublished memorandum and order issued on November 5, 1982, we concluded that the existing record was unclear as to whether adequate core decay heat removal can be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 small break loss of coolant accident. ALAB-708, at 42. The TMI-2 accident involved both a loss of main feedwater and a small break loss of coolant accident. Thus, our conclusion was that the evidence below did not support a conclusion that TMI-1 could withstand the type of accident which occurred in TMI-2.

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3. Reserving for later the question of emergency feedwater reliability, we stated:

> 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, <u>infra</u>), 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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> 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.
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> ALAB-708, Sl.op. at 9-10.

4. Having reached this conclusion, we had two choices for further action. Having found that record insufficient to support restart of TMI-1, we could have issued a decision reversing the Licensing Board's conclusion that TMI-1 was safe enough to restart.

5. Our second choice was to reopen the record and afford the Licensee and Staff another opportunity to try to prove that TMI-1 is safe enough to restart.

6. We decided upon a limited reopening of the record and directed the Licensee and/or Staff to provide supplemental testimony in response to 11 questions we posed. ALAB-708, at 42-45. We turn now to addressing the evidence we solicited.

7. Appeal Board Question 1. asks for "[t]he exact size and flow rate of the vents to be installed in the hot legs." ALAB-708. at 43.

8. This question arose because previous statements of the NRC Staff and Licensee concerning the size of the hot leg high point vents were inaccurate. See ALAB-708 at 17, n. 25 and 19, n. 30.

9. In fact, the vents are quite small, having an orifice of 0.371 in diameter. Capodanno, ff. App. Tr. 43 at 1.[1]

10. Appeal Board Question 2 was as follows: When and under what circumstances such vents would or would not be useful to promote natural circulation, including reasons for the conclusions reached.

11. Opening of the hot leg high point vents would provide virtually no benefit for recovering natural circulation during the early phases of a small break LOCA. Jones and Lanese, ff. App. Tr. 53 at 5.

12. While the vents might provide some incremental assistance in the latter phase of a SBLOCA, this limited benefit does not outweigh the complexities associated with determining the conditions under which the vents may be opened. Id. at i.

13. As the Board noted in ALAB-708, both UCS and the Staff have pointed out 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. ALAB-708 at 18-19.

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[1] The transcript of the Appeal Board hearings of March 7, 8, 16 and 17, 1983 are cited as "App. Tr."

14. Current B&W and NRC computer codes are incapable of calculating the amount of subcooling required to prevent flashing to steam. App. Tr. 92, Sheron.

15. B&W withdrew its proposed guidelines for use of the vents because it is unable to resolve the complexities inherent in their use to the Staff's satisfaction without either integrated systems testing obtained from a facility geometrically similar to a B&W plant and/or analytical confirmation. App. Tr. 60-63, Lanese; App. Tr. 86, Sheron. Neither the testing nor analysis has yet been provided. App. Tr. 86, Sheron.

16. While there are negotiations ongoing concerning modification of GERDA to add a loop and make it closer in configuration to TMI-1, no contracts have been signed. App. Tr. 65, Lanese. The NRC would have to approve an engineering design and none has been presented to it yet. App. Tr. 100, Sheron. The most that can be said is that some relevant data could become available in mid-1985. App. Tr. 66-67, 79. That, of course, does nothing to provide reasonable assurance of safe operation now.

17. The TMI-1 operators have not been trained in use of the vents during recovery from a small break LOCA and no guidelines exist. App. Tr. 61. The letter from the Director, Division of Licensing, NRC, referenced at ALAB-708, p. 28 and n, 39, stating that the Staff understands that the operators will be trained to use the vents to remove steam bubbles, is incorrect.

18. Under these circumstances, it is obvious that the Board's concerns cannot be resolved by relying on the high point hot leg vents.

19. Appeal Board Question 9 is: Whether and under what circumstances reliance on feed and bleed is necessary at TMI-1.

20. Licensee continues to maintain that feed and bleed is only necessary for "those beyond-design-basis events, involving an extended loss of both main and emergency feedwater..." Jones and Lanese, ff. App. Tr. 111 at 2.

21. In fact, total loss of feedwater is not beyond-design-basis for TMI-1. Since main feedwater is not safety grade (App. Tr. 204, Sheron), total loss of feedwater would be beyond-design-basis only if emergency feedwater (EFW) were safety grade. Indeed, Licensee has conceded that total loss of feedwater would be a design basis event if the EFW system were not seismically qualified. Tr. 5709, Lanese.

22. Therefore, feed and bleed is a necessary means of decay heat removal for design basis events for TMI-1. Licensee's testimony to the contrary quoted above is not true.

23. Moreover, there has never been any question that the scope of the Restart Proceeding includes review of the reliability of the EFW system, whether the need for that system is occasioned by a SBLOCA or a loss of main feedwater from any cause. This was the essence of Board Question 6. See, e.g. 14 NRC 1211 at 1356 (para. 1010), 1358 (paras. 1016, 1017), 1360 (para. 1024), and 1364 (para. 1039).[2]/ If the loss of main feedwater is caused by an earthquake (as one must assume, since main feedwater is not safety-grade and not seismically qualified), that same earthquake would also disable EFW, leading to complete reliance on feed and bleed to remove core decay heat. The TMI-1 EFW is not safety grade for all loss of main feedwater events.

24. Belated claims that the scope of the restart proceeding is limited to small break LOCA's are utterly inconsistent with Board Question 6 and

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[2]/ Even the Appeal Board characterized the issues as 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 at 7 n. 5.

constitute a transparent attempt to avoid facing the consequences of the fact that the TMI-1 emergency feedwater system does not meet the Commission's minimum requirements for systems important to safety and therefore cannot be relied upon for decay heat removal.

25. Furthermore, the question of whether TMI-1 has sufficiently reliable means of decay heat removal could not be more central to this proceeding. The importance of decay heat removal was a crucial lesson learned from the TMI-2 accident. As the Licensing Board noted, the TMI-2 Lessons Learned Task Force found that "the need for an emergency feedwater system of high reliability is a clear lesson learned from the TMI-2 accident." 14 NRC 1211, 1356 (para. 1008) citing NUREG-0578, p. 10.

26. Licensing Board Question No. 6 reads:

> a. Is a loss of emergency feedwater following a main feedwater transient an accident which must be protected against with safety-grade equipment? Would such an accident be caused or aggravated by a loss of non-nuclear instrumentation, such as occurred at Oconee?
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> b. In what respect is the emergency feedwater system vulnerable to non-safety-grade system failures and to operator errors?
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> c. What has been the experience in other power plants with failures of safety-grade emergency feedwater systems, if they have such systems in other power plants?
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> d. What operator action is required to operate in a feed-and-bleed mode following a loss of emergency feedwater?
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> e. If the emergency feedwater system were to fail, what assurance do we have that the system can be cooled by the feed-and-bleed mode? This is of particular concern if the PORV's and safety valves have not been tested under two-phase mixtures.
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> f. Can the system be taken to cold shutdown with the feed-and-bleed cooling only? Are both high pressure injection (HPI) pumps required to dissipate the decay

heat in the feed-and-bleed mode? The Board would like an evaluation of the reliability of the feed-and-bleed system. Has there been any experience using that system?

g. If there is a loss of steam in the secondary system which results in failure of the turbine-driven feedwater pumps, will both motor-driven pumps be required to supply the requisite amount of feedwater? Does this meet the usual single-failure criteria since it appears that a redundant system requires multiple components to operate?

h. Can the turbine driven pumps and valves be operated on Direct Current, or are they dependent upon the Alternating Current safety buses?

i. Will the reliability of the emergency feedwater system be greatly improved upon conversion to safety-grade, and is it the Licensee's and Staff's position that the improvement is enough such that the feed-and-bleed back-up is not required?

j. Will the short-term actions proposed improve the reliability of the emergency feedwater system to the point where restart can be permitted?

k. Question 6 should be addressed with reference to Florida Power & Light Co. (St. Lucie, Unit 2), ALAB-603, (July 30, 1980); i.e., whether loss of emergency feedwater is a design basis event notwithstanding whether design criteria are met. (PID para. 1005)

27. The Board explained in great detail the consideration which led it to pose and pursue these issues. 14 NRC 1211, 1355, ff.

28. It noted that the NRC's early evaluation of the TMI-2 accident led it to the view that B&W reactors appear to be unusually sensitive to transients originating in the secondary system and place more reliance on the reliability and performance characteristics of the EFW system than do other PWR designs. PID para. 1007.

29. The Board also noted that the TMI-2 accident "highlighted the importance of the EFW system." para. 1008. Unavailability on demand of the EFW system was a potential contributor to the severity of the accident. Id.

30. Indeed, the TMI-2 Lessons Learned called for upgrading of the EFW system to safety grade--not just for SBLOCA's but for all design basis accidents. See Staff Ex. 1., p. C8-37.

31. As the ASLB stated: "Our task is to decide whether the requirements of the Commission's orders have been met and whether the improved EFW reliability is adequate to protect the health and safety of the public." emphasis added. (p. 1009) This is also the Appeal Board's task.

32. In response to Board Question 6, both NRC Staff and the Licensee claimed that the EFW system for TMI-1 was adequately reliable. In the Licensee's case, it relied primarily upon a comparison of the EFW system against NRC requirements for systems important to safety. It asserted, inter alia, that "[t]he EFW piping system is.... designed and qualified to the seismic Class 1 requirements." Lic. Ex. 15, Table 1, p. 1

33. The NRC Staff performed a quantitative reliability analysis of the TMI-1 EFW system. It too, assumed that, with a few nongermane exceptions, the EFW systems was seismically qualified. We now know that this is far from the case. See discussion on Appeal Board Question 8, infra.

34. Therefore, one of the basis for the ASLB's conclusions regarding EFW system reliability was false. The EFW system is less reliable than the ASLB believed. It does not even meet the current licensing requirements used by NRC for systems important to safety; it is not safety grade.

35. The Board can only conclude, therefore, that feed and bleed is needed at TMI-1 to remove core decay heat for design basis events.

36. With respect to the degree to which feed and bleed is relied on at TMI-1, the Staff stated that feed and bleed is "neither relied upon nor necessary to remove decay heat for events within the design basis for TMI-1." Sheron and Jensen, ff. Tr. 83 at 22. As in the case of the Licensee's

similarly unequivocal assertion, the statement is not true. Only small break LOCA's and certain loss of main feedwater transients were considered. App.Tr. 202, Sharon. For example, seismic events or losses of main feedwater due to earthquakes were not considered.

37. In fact, there are design basis events for which feed and bleed is needed; however, in the Licensee's and the Staff's view, these are beyond the scope of the reopened hearing.

38. The Staff conceded that, if the EFW system were not safety grade, there would have to be an alternate, safety-grade means for decay heat removal under current NRC rules. App. Tr. 202-203, Sheron. In the case of SBLOCA and loss of main feedwater, feed and bleed is the only "alternate" means and the Staff has never said that it is safety grade. *Id.* at 203.

39. Moreover, the Staff position during this proceeding was that feed and bleed was necessary to meet NRC rules pending full upgrade of the EFW system to safety grade.

40. NRC Staff Proposed Finding, paragraph 435 is as follows:

> Until EFW system upgrading at TMI-1 is completed, the Staff is relying on the feed and bleed mode of core cooling to protect against events for which the EFW system is not fully safety grade. (emphasis added)

41. NRC Staff Proposed Finding 441 is as follows:

> Based on our consideration of the evidence on the record of this proceeding, we find that although the EFW system at TMI-1 will not be fully safety-grade at the planned time of restart, it will have been upgraded to significantly improve its reliability, that operator action within about 20 minutes to actuate the safety-grade HPI pumps and initiate feed and bleed cooling can protect against failures of both the main and emergency feedwater systems, that feed and bleed cooling can be continued until feedwater is restored and thus that there is reasonable assurance that the public health and safety will be adequately protected

> <u>against feedwater transients if TMI-1 is allowed to restart prior to full upgrading of the EFW system to safety-grade</u>. (emphasis added)

42. There can be little question that, at restart, feed and bleed is relied upon to meet NRC's minimum safety requirements for decay heat removal. This was the NRC's own testimony above. The ASLB noted additionally that the Staff relies on feed and bleed to meet 50.46 for certain small breaks, with manual HPI initiation. 14 NRC 1211, 1334 (para. 948).

43. We note in this connection that the Staff originally considered a <u>fully</u> safety-grade EFW system to be so important that it rejected Licensee's proposal to delay this modification until after restart. See St. Ex. 1 at C8-37. NUREG-0737 "required" EFW to be fully safety grade by July 1, 1981. Staff Ex. 14 at 36. The Staff later found "reasonable progress" on the basis of a projected completion date of late 1982, Staff Ex. 14 at 38.

44. In response to questions of Dr. Gotchy, Staff Witness Jensen claimed that the emergency procedures for loss of feedwater instruct the operator to depressurize the steam generators and use the condensate pumps, instead of going to feed and bleed. App. Tr. 204, Jensen. Mr. Jensen was wrong. When presented with the emergency procedure for loss of feedwater, EP 1202-26A, UCS Ex. 45, Mr. Jensen could not substantiate his claim.

45. In fact, the procedure instructs the operator to turn on HPI and go into feed and bleed. App. Tr. 208-209, Jensen. Contrary to Jensen's testimony, the operator is <u>not</u> instructed to depressurize the steam generators and use low pressure pumps. <u>Id</u>. at 209. We regret to say that Mr. Jensen's testimony generally cannot be accorded much weight by this Board when we consider his history of giving evidence beyond his area of technical competence; which, upon probing, does not withstand scrutiny. <u>See, e.g.</u> UCS Proposed Findings of Fact and Conclusion of Law, June 1, 1981, paras. 21, 22, 67-69.

46. Finally, we wonder what to make of Staff Counsel's curious attempt to get his own witness to disclaim any relationship between the reliability of a system and whether or not it is safety-grade. App. Tr. 211. Since it is precisely the intention of the NRC General Design Criteria defining the requirements of a safety grade system (redundancy, diversity, seismic and environmental qualification, testability, etc.) to make those systems highly reliable which are needed for safety, (App. Tr. 365, Wermiel; See also Pollard, ff. Tr. 8182 at 3-5, 3-6) we cannot believe that Counsel intended to suggest that the NRC rules are entirely ineffective in achieving their purpose. In any case, we take Mr. Jensen's lack of knowledge of any relationship between the reliability of a system and whether or not it is safety grade as a reflection of the limits of his own knowledge rather than as evidence of the lack of any such relationship.

47. In ALAB-708 at n.5, p. 7, we stated:

> 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.

48. While we believe that our authority to order the Licensee to seismically qualify the TMI-1 EFW system may be in question, it is our obligation to determine whether the short and long term measures are sufficient to protect public health and safety and if they are not, to prevent operation of the

plant until it is sufficiently safe to operate. We noted that the fact that the TMI-1 EFW is not seismically qualified "does raise the possibility, however, that reliance may have to be placed on other plant systems to provide adequate core cooling." Thus, our conclusion that feed and bleed is relied upon to provide core cooling in light of the fact that EFW is not safety grade is entirely consistent with ALAB-708 and our obligations under the Commission's Order. We now go on to consider if feed and bleed is adequately reliable as a means of decay heat removal.

49. The Staff's testimony on Appeal Board Question 9 failed to allay our concerns regarding reliance on feed and bleed.

50. While conceding that feed and bleed cooling involves using systems under conditions for which they were not specifically designed, the staff stated in its prefiled direct testimony: "we believe that there is a high probability that these systems will perform successfully...." Sheron and Jensen, ff. App. Tr. 83 at 22. The systems in question include the HPI pumps, pressurizer safety valves and PORV.3/ App. Tr. 184, 185, Sheron. On cross-examination by UCS, however, it was established that the Staff's "belief" is at best speculation.

51. The Staff has "no idea" how long the HPI pumps might be required to pump against high pressures for feed and bleed. App. Tr. 187, Sheron. There is no design basis for feed and bleed. Id. at 188. The Staff has not carried out any detailed analysis to determine the capabilities of long-term feed and bleed. App. Tr. 186, Sheron. There have been no tests of the HPI pumps under feed and bleed conditions. Id. at 188. It has been established earlier in this proceeding that the Staff has not evaluated the nature of the demands

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3/ The Staff would disapprove any emergency procedures that didn't direct operators to use the PORV. App. Tr. 201, Sheron. See UCS Brief on Exceptions, p. 44.

that would be placed upon the safety valves during feed and bleed -- either the number of times they would be called upon to operate or the flow quality they would be required to relieve. Tr. 8920, 8930-33, Tr. 9012-3. The EPRI valve testing program is not capable of resolving these questions since the test facility cannot simulate the rapid repressurization associated with feed and bleed. Tr. 8920-22.

52. The Staff's basis for believing in the capability of the HPI pumps for feed and bleed is an assumption that "one has to be able to restore some sort of feedwater within a few hours." Id. at 187. However, this convenient "assumption" is not accompanied by any understanding or consideration of the cause of the loss of feedwater nor therefore, any basis for concluding that it would or could be restored. It is, at most, a hope unsupported by anything approaching analysis.

53. As the Staff witness stated when pressed for the reasoning behind his beliefs:

> Well, I would like to go back and just state that, again, we are not requiring feed and bleed for any specific scenario. There is no design base for feed and bleed, and its very difficult for me to sit and speculate under what conditions it will or won't work and for how long and to design a specific scenario when we don't have one. App. Tr. 188, Sheron.

54. The key word above is "speculate". The Staff's testimony on this subject consisted of speculation. The optimistic nature of that speculation does nothing to lend it credence. Again, to quote directly from the transcript:

> Q. Is it correct then that your testimony is you're not aware of any information that the TMI-1 HPI pumps cannot perform as required during feed and bleed cooling, nor are you aware of any information that they can perform as required during feed and bleed cooling?
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> A. (Witness Sheron) I think that is a fair statement.

55. The other witness subsequently suggested that, because the HPI pumps are designed to operate indefinitely at 2200 psi under normal conditions, there is reason to believe that they could operate for a long period at or above 2500 psi. App. Tr. 194-197, Jensen.

56. However, one cannot extrapolate from the pump's performance at 2200 psi to predict successful long-term operation at or above 2500 psi. As the shutoff head of the pump is approached, there is a point at which insufficient water passes through the pump to keep it cooled. App. Tr. 199, Jensen. The witness hadn't looked at this. Id.

57. We note that Staff Counsel attempted to explain the lack of support for the Staff's testimony apparent during cross-examination by claiming that these were not the appropriate witnesses to discuss the capability of the equipment needed for feed and bleed. App. Tr. 191, Cutchin. The fact is that the witnesses flatly stated on page 22 of their prefiled testimony: "we believe that there is a high probability that the systems will perform successfully." Only during questioning was the utter lack of factual support for this "belief" revealed. The witness stated that the Staff did not intend to convey any confidence in the operability of the systems needed for feed and bleed. App. Tr. 192, Sheron.

58. Indeed, the Staff has no understanding of either the requirements for feed and bleed nor the degree to which the TMI-1 equipment can be relied upon to meet those requirements. Under those circumstances, the above-quoted testimony is more than gratuitous, it is seriously misleading. The NRC staff

should not offer opinions to the Board which it is not able to support. Nor did the Licensee's testimony provide the needed assurance that the TMI-1 components can be relied upon to perform as they may be required during feed and bleed.

59. In the proceeding below, the Licensing Board asked the following question regarding UCS contention 6:

> "The Board wants more than just a schedule for testing of reactor coolant safety and relief valves, as is required pursuant to NUREG-0578. Is there reasonable assurance that the tests will be successful, e.g., that there is good evidence that the valves will indeed perform in an accident environment?"

60. The Licensee testified that, because of the construction of the safety valves, there is no reason to expect that liquid or two-phase flow conditions would have a detrimental effect on the ability of the valves to perform their required function. Correa and Urquhart, ff. Tr. 8746 (Board Question on UCS Cont. 6), at 2. Similarly, the Staff testified that "there is presently no evidence that [the safety] valves will not operate properly during the anticipated transients which produce two phase flow and solid fluid flow," and that "operation of TMI-1 prior to completion of the EPRI test program would not endanger the health and safety of the public." Zudans, ff. Tr. 8824, at 7.

61. However, results from the EPRI tests of the type of safety valves used at TMI-1 show that the "expert" judgments of both the Staff and Licensee were wrong. The results of those EPRI tests showed that the safety valves would not operate properly under conditions of two-phase or liquid flow without, at a minimum, certain modifications. These involved changing the inlet piping from a long inlet to a short inlet arrangement and changing the safety valve settings to increase blowdown from 3% to no more than 20%. Jones and Lanese, ff. App. Tr. 111, at 3-4.

62. There were a total of only 31 tests performed with a model of safety valve identical to the TMI-1 safety valves, App. Tr. 137, Lanese, but in only four of those tests were the valves required to relieve liquid. Jones and Lanese, ff. App. Tr. 111, at 4. Furthermore, in those four tests, the valve opened and closed a total of four times -- one opening and closing per test. App. Tr. 146. (Lanese).

63. After the witness whose prefiled testimony had been submitted was demonstrably unable to justify a conclusion that the safety valves would perform as required, the Licensee was permitted to call another witness, Mr. Correa, who testified orally that the 31 tests provide a satisfactory demonstration of the ability of the safety valves to perform their intended functions. App. Tr. 383 (Correa). For the reasons discussed below, we decline to accept Mr. Correa's views on the adequacy of the EPRI tests to demonstrate proper safety valve performance for the feed and bleed cooling mode.

64. Licensee relies on all 31 tests - including the 26th test during which the valve chattered (opened an undetermined amount, reseated, reopened, etc.) approximately 1250 times -- to support the proposition that the safety valves will perform properly during feed and bleed cooling. App. Tr. 383-386, 402 (Correa).

65. However, on cross-examination, the witness admitted that, during the EPRI tests, it was not always possible to determine whether the safety valve was experiencing flutter or chatter. App. Tr. 394 (Correa). Flutter is the valve moving between intermediate lift (open) positions but not hitting the valve seat. Chatter is the valve moving from some lift (open) position back to the seat (closed) and then lifting (reopening) again. Id. at 393. The Witness admitted that if the damage observed from the chattering during the

26th test was caused by partial opening and reclosing, the damage to the valve could be greater if chattering involved full opening and closing of the valve. Id. at 402. Contrary to the witness' testimony before the Licensing Board, the EPRI tests disclosed new phenomena - flutter and chatter - which the witness had not expected. Id. at 406-408. Despite this first hand experience of the fallibility of his "expert" engineering judgment, and despite the evidence that safety valve performance keeps on getting less satisfactory as the subcooling of the liquid discharge increases, the Licensee concurred in cancellation of a planned test at 2500 psi using 400-degree water. Id. at 417-419. Increased subcooling of the liquid discharged through the TMI-1 safety valves is precisely what will occur if the feed and bleed cooling mode is used and is successful. The Licensee's witness claimed to be unfamiliar with system response during feed and bleed. Id. at 419. This is somewhat surprising since Licensee presented this witness in purported response to UCS's concerns that two-phase (or liquid flow) through the safety valves might affect the ability to feed and bleed and that the safety valves are not qualified to perform the "bleed" function. ALAB-708, at 39-40.

66. Licensee's witness did not know whether a failure mode analysis had been done to determine whether a failure mode exists that would prevent the safety valves opening. He nevertheless proffered the opinion that "there should be no damage occuring during a water discharge which would keep the valve closed." App. Tr. 424-425 (Correa). Subsequent testimony during cross-examination showed such damage is possible.

67. When questioned by us about the damage to the safety valve at Crystal River, the witness read selected portions from a document not previously available to the parties other than Licensee. Id. at 427. He testified that the valve seat and disk mating surfaces were extensively steam cut, but that

inspection of valve parts "revealed no evidence of damage upset metal, galling or searing" and some mating surfaces "should no evidence of excessive loads, unstable operation, or banging of the internal parts." Id. at 426-427.

68. First, of all, the damage to the Crystal River safety valve was much more extensive than the witness chose to disclose. UCS, during a ten-minute recess (App. Tr. 431), perused the document[4] from which the witness selectively read and found passages describing serious damage to the safety valve:

> "The valve seat and disk plating surfaces were extensively steam cut. Steam cutting resulted in several hundred radial marks across the seating faces with depths of several mils. The lower surface of the disk holder (part II) adjacent to the disk also showed erosion damage. Liquid penetrant examination of the seating surfaces of the seat and disk and the lower surface of the disk holder showed fine radial cracks 30 to 60 mils deep scattered around the lower surfaces of the disk holder. These cracks were not visible to the unaided eye. The seat and disk were steam cut but not cracked. First, the antirotation pin was found dislodged inside the bonnet cavity. This pin was bent and had become dis-engaged from the upper spring washer and fallen into the valve bonnet cavity." App. Tr. 441-442.

69. The Crystal River safety valves are identical to the TMI-1 safety valves. App. TR. 446 (Correa). Both Crystal River safety valves had the antirotation pin dislodged inside the bonnet cavity. Id. at 442, 444. A loose part such as this pin could have an effect on valve operability. Id. at 442.

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[4] "Examination and Test of Crystal River Unit No. 3, Power Operated Relief and Safety Valves, PWR Safety and Relief Valve Test Program EPRI," NP-80-13-LD, December 1980. App. Tr. 437-438.

70. Similarly, the witness did not voluntarily disclose the extent of damage during the EPRI tests. After the 26th EPRI test, the safety valve was extensively repaired. Nine new parts were installed and five others were "refurbished." Licensee's witness said the refurbishing consisted of hand polishing as opposed to machining. Id., at 429-430. This was an assumption without a reliable basis: "It says polished. I assume it was a hand operation. If it said remachined, it would have been machined." Id. at 430 (emphasis added). We attach no weight to the witness' assumption about the degree of "refurbishment." For example, the support plate assembly had to be straightened with a press. Id. In any event, it is clear that the valve which underwent the water tests was extensively repaired and for all practical purposes was a brand new valve.

71. We find that four or five cycles of a brand new valve is insufficient to demonstrate reliable performance during feed and bleed cooling. Since the safety valves have not been tested over the full range of conditions, i.e., temperature and flow quality, and in the manner of operation (repeated opening and closing for number of hours or days) required for feed and bleed cooling, UCS' arguments that the safety valves have never been tested or qualified under conditions that would prevail during feed and bleed cooling are as true today as prior to the EPRI test program. UCS Brief on Exceptions, at 21-24. There is insufficient basis on this record that the values would perform as required.

72. Appeal Board Question 10 calls for "[r]esults 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."

73. As we note in ALAB-708, the results of feed and bleed tests at the Semiscale test facility, S-SR-1 and S-SR-2, were not reported to us until after the oral argument. See BN-82-93 and BN-82-107. In neither test was steady-state feed and bleed achieved. UCS brought the implications of these tests to our attention. <u>See</u> UCS Response to Board Notification BN-82-93, October 1, 1982, and UCS Reply to Appeal Board Order of October 15, 1982, October 29, 1982.

74. The EG&G report on S-SR-2 concludes that the viability of feed and bleed "depends on plant-specific characteristics and postulated scenarios." ALAB-708 at 38, citing EGG-SEMI-6022, "Analysis of Primary Feed and Bleed Cooling in PWR Systems," September 1982, attached to BN-82-107. EG&G concluded further that feed and bleed is theoretically possible only within a certain band of primary system pressure. <u>Id.</u> at 2-3. There is no evidence on this record of a plant-specific analysis to demonstrate the viability of feed and bleed for TMI-1. The existence of high head HPI pumps at TMI-1 does not resolve this concern; there has been no demonstration that the set point of the safety valves is less than the theoretical upper bound of the plant-specific feed and bleed pressure band for TMI-1. <u>See</u> UCS Response to ALAB-708, June 19, 1983 at 5-6 and UCS Response to Board Notification BN-82-93, Oct. 7, 1982 at 7-8.

75. We stated that 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. ALAB-708 at 42.

76. The Staff hired EG&G to run a RELAP5 calculation using the HPI flow and

steam generator secondary heat losses that occurred during S-SR-2. See Sheron and Jensen ff. Tr. 83 at 2-3 (not it should be noted, the actual plant-specific data for TMI-1).

77. The Staff's conclusion is that RELAP5 "was capable of predicting the data to within the accuracy of the experimental uncertainties." These uncertainties, however, are critical. The Staff concedes that small uncertainties in the inventory calculation could produce significant uncertainties in the level calculation and, consequently, the degree to which core uncovery would be expected. Id.

78. To put it more clearly, if actual core level were significantly different than the calculated level, but still well within the range of experimental uncertainty associated with this analysis, one could change from the conclusion that feed and bleed is adequate to the conclusion that it is not adequate to prevent core uncovery. App.Tr. 235-237, Sheron. Therefore, the RELAP5 re-analysis of S-SR-2 does not "confirm" the capability of the code to predict feed and bleed.

79. As EG&G noted additionally, there are some plant parameters that are so uncertain that the code may predict adeuqate cooling where, in reality, there would not be.

> It is evident that plausible variations in these parameters [i.e., ECCS flow, safety valve and/or PORV flow, and decay heat] can lead to the elimination of a steady-state operating pressure range.
>
> The inability to maintain system inventory once the PORV [or safety valves] mass flow rate dropped to reflect steam flow is subject to experimental uncertainties. Uncertainties exist in the actual PORV orifice characteristics. HPIS injection rate and the measurement thereof, system heat loss, and fluid leakage.
>
> P. North EG&G to R.E.Tiller, DOE, August 6, 1982, Attached to BN-82-93, P.7-8, footnote omitted.

80. Thus, the uncertainities in the actual values of critical parameters such as heat loss, mass injection or feed rate and mass loss or bleed rate, are within experimental uncertainties. A computer code may predict successful core cooling where, in reality, it might not be achieved.

81. Under these circumstances, even if the RELAP5 calculation and S-SR-2 were consistent, one can draw little assurance from that about the efficacy of feed and bleed for TMI-1.

82. Hence, EG&G's conclusion is compelling:

> The extreme sensitivity to measurement uncertainties makes an analysis based upon mass and energy balance more meaningful than a side-by-side comparison of RELAP5 calculations and data. UCS Ex. 46, p. 20.

83. No such analysis based upon mass and energy balance has been done for TMI-1. App. Tr. 252, Sheron.

84. There are other extremely troubling indications that the comparison of RELAP5 against S-SR-2 does not provide confidence in the ability of the code to predict the feed and bleed phenomenon nor the viability of feed and bleed at TMI-1. EG&G observed that so long as the PORV was open during S-SR-2, the core was proceeding towards uncovery but, had the PORV been cycled, steady state feed and bleed would have been achieved. UCS Ex. 46, p. 18-20, App. Tr. 237-8. This might also be true for TMI-1. App. Tr. 241. Sheron.

85. All applicable TMI-1 emergency procedures for feed and bleed instruct the operator to open the PORV. UCS Proposed Findings of Fact, para. 208. Contrary to simplistic assertions made earlier in this proceeding, lowering RCS pressure is not always beneficial, as EG&G's observation indicates. Yet no plant-specific analysis has been done to demonstrate that, assuming the plant is operated as the procedures dictate, steady state feed and bleed can be achieved. It can clearly no longer be accepted as true that "initiation of

feed and bleed is a very simple operation and can be continued indefinitely." 14 NRC 1211, 1371 (para. 1052).

86. We believe that the uncertainties in this area are so great as to preclude the conclusion that feed and bleed is viable for TMI-1 based upon a comparison of RELAP5 and S-SR-2.

87. There were many other problems with the Staff's presentation of the RELAP5 calculation of S-SR-2. Not the least of these was the Staff witnesses' general inability to answer specific questions about the test results and calculations and lack of familiarity with the analysis in any detail. For example, the Staff witnesses did not know whether the RELAP5 calculations were run for a period corresponding to the entire length of test S-SR-2. App. Tr. 218-9, Sheron.

88. When questioned about why RELAP5 was changed from a thermal non-equilibrium mode to a thermal equilibrium mode at 1350 seconds, the Staff "presumed" that EG&G had decided that the change would not alter the conclusions. App. Tr. 215-6, Sheron.

89. Another very questionable aspect of the RELAP5 results to which we could receive no satisfactory explanation concerns the inconsistency posed by the fact that while the EG&G report states that differences between measured and calculated HPI flow are due solely to differences between measured and calculated primary system pressure, and the code calculated an increase in HPI flow at around 1900 seconds while the test data trended consistently down. App. Tr. 225-227.

90. When UCS questioned the Staff about this at a pre-hearing deposition, the Staff was prompted to question EG&G. The Staff still does not know why the peak is there, but thinks that the method used by EG&G to precontrol or specify the characteristic of the pump may have had some sort of anomaly which

produced this. App. Tr. 227-229, Sheron. The actual HPI pumps at TMI-1 do not have this characteristic. App. Tr. 229, Sheron. The witnesses never adequately explained why, if the calculation inputed the actual HPI test characteristics, there is not a flow peak around 1900 seconds in the actual data. App. Tr. 232-233. Their answer was a "guess". Id.

91. The Staff did agree that HPI flow is an important parameter in analyzing whether feed and bleed will successfully cool the core and further agreed that at some points, particularly around 1000 seconds, the difference between calculated and experimental flow was roughly 50%. App. Tr. 233-234, Sheron. This does not seem to the Board to represent good correspondence between the test and the calculation, particularly when the differences are not well explained and the uncertainty in ultimate conclusion is so great.

92. We agree with the implications of the EG&G report; absent a plant-specific feed and bleed mass and energy balance analyses for TMI-1 (i.e., "operating map") the Board cannot find assurance that feed and bleed is a viable means of decay heat removal for TMI-1. We are not persuaded otherwise by the Staff's observation that, while "useful", such a map would not be conclusive absent a detailed calculation accounting for the variations in decay heat rate over time. App. Tr. 252-254, Sheron. It may well be true that several "operating maps" at several selected points in time would be required in order to provide the necessary level of confidence. (Just as in the case of a LOCA analysis, a suitable spectrum of breaks is selected to represent all breaks.) The fact remains that the information presented to date does not provide a basis for finding that feed and bleed will cool the TMI-1 core.

93. Appeal Board Question 11 called for "results of a RELAP5-type analysis to determine whether feed and bleed will successfully provide core cooling at TMI-1". The Staff contracted with EG&G to perform what is referred to as a "feed and bleed analysis" for TMI-1 with RELAP5. However, the analysis assumed only a loss of feedwater; no small break LOCA was postulated. App. Tr. 317, Sheron. The analysis is therefore of very limited usefulness; it does not even model the situation which occurred during the TMI accident.

94. The Staff has no written report from EG&G. App. Tr. 276, Sheron[5] They think that the TMI-1 pressurizer geometry was modeled but have not examined the input to the calculation. App. Tr. 277, Sheron and Jensen. They do not know whether the safety valve relief capacity used in the calculation was identical to TMI-1. App. Tr. 278[6]

95. There is a great deal of confusion which the Staff was unable to clarify concerning the assumption used for safety valve relief capacity. App. Tr. 278-284. The calculation did not model relief capacity over time as a function of changing flow quality but instead used an "effective flow area." App. Tr. 275-6, Sheron. The Staff does not know what effective flow area was used. App. Tr. 280, Sheron.

96. However, EG&G states that it used a relief capacity 15% below the tested relief capacity for steam for the "Dresser-type" valve used at TMI-1. App. Tr. 278. The Staff does not know what tests are referred to by EG&G (App. Tr. 279, Sheron) and does not know whether the valve is identical to those at

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[5] The transcript contains an error on page 276. Line 12 should read: "That you still do not have the written report for that ...."

[6] Note also that the B&W analyses all assume the use of two HPI pumps, App. Tr. 162, 306. This assumption is inconsistent with the assumptions made for NRC safety analyses. Feed and bleed does not meet the single failure citerion.

TMI-1. Id. at 278. These are critical parameters to the analysis, the appropriateness of which this Board cannot take on faith.

97. The Staff states that EG&G estimates that the uncertainty in relief capacity is plus or minus 15% of the rated capacity for steam flow--but the Staff does not know the bases for that uncertainty and could therefore obviously not answer any questions about it. App. Tr. 279, Sheron.

98. With regard to the liquid relief capacity, EG&G states that the flow calculated by RELAP5 is an average of 9% above the measured flow but the Staff did not know where the flow was measured or the basis for assigning an uncertainty of plus or minus 15% to that calculation. App. Tr. 279-280.

99. The Staff could not explain why EG&G used a flow discharge area for liquid relief that was 15% smaller than measured flow. Sheron and Jensen, ff. Tr. 83 at 40.[7/] App. Tr. 281. While conceding that inlet piping affects the critical flow through the valve, the Staff did not know whether EG&G took account of the change at TMI-1 from long to short inlet piping. App. Tr. 284-5, Jensen.

100. All the Staff could pass on was what was reported to it orally by EG&G. Apparently, the Staff did not ask any of these questions of EG&G and clearly had no independent knowledge of its own. It acted simply as a mouthpiece to pass on what it stated to be EG&G's general conclusions. Neither the parties nor the Board has any way of probing these conclusions nor even of understanding their bases. We have determined that this amounts to a denial of UCS's right to cross-examination as required for a "full and true

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7/ The prefiled testimony is less than clear. It states: "[EG&G] reported that the safety valve liquid flow calculated by RELAP5 is an average of a percent above the measured flow, and the uncertainty on this value is ± 15 percent. However, the flow discharge area was sized to 15 percent smaller in the analyses."

disclosure of the facts" under 10 CFR 2. 743(a).[8] In any case, we are unable to accord significant weight to this testimony when there was no effective means of resolving questions properly raised concerning the validity of its assumptions regarding critical parameters.

101. The Staff stated on page 41 of its direct testimony that, for feed and bleed to be a viable means of decay heat removal, the safety valve flow must be almost essentially steam flow, not liquid or two-phase. It described a "conceivable" phenomenon whereby steam entering the surge line could entrain liquid in the pressurizer, resulting in two-phase discharge. According to the Staff witness, S-SR-2 showed that the code "was not doing a reasonable job of calculating inventory in the pressurizer once the surge line is uncovered and begins to pass steam." App. Tr. 295, Sheron. It is possible that this phenomenon would occur for TMI-1. App. Tr. 297, Sheron.

102. The Staff concludes that in order for feed and bleed to be effective, the safety valve discharge must be steam. Sheron and Jensen, App. Tr. 83 at 42. However, based on the above, it is not possible to conclude that the discharge would be steam.

103. The Staff tried to explain, contrary to its direct testimony, that this would not be significant. However, its explanation assumed a favorable mass and energy balance. App. Tr. 316-317, Sheron. We must remember here that no operating map has been constructed for TMI-1 showing a favorable mass and energy balance (with use of the PORV) culminating in steady state feed and bleed. In addition, the EG&G analysis in question did not postulate a LOCA, but only a loss of feedwater. App. Tr. 317, Sheron. Therefore, the Staff's explanation is unsatisfactory.

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[8] UCS has anticipated this situation and requested the presence of witnesses from EG&G. UCS Response to ALAB-708 and Request for Modification of Schedule, June 9, 1983, p. 11. The Staff was fairly on notice.

104. The Staff purported to account for code uncertainties by assuming a 25% uncertainty in calculated vessel inventory and directing EG&G to make a calculation based on that. Sheron and Jensen, ff. Tr. 83 at 42, App. Tr. 298-299. The 25% figure was picked by the Staff (not EG&G) because it is greater than the 20% uncertainty estimated by EG&G to be associated with the calculation described in connection with Question 10, the comparison of RELAP5 with S-SR-2. App. Tr. 298, Sheron. This latter uncertainty estimate is, of course, not particularly relevant. The Staff has no basis for concluding that the input data used for the TMI-1 calculation is not more than 25% different from reality. App. Tr. 299-300, Sheron. For example, no test has been done for the TMI-1 HPI pumps to determine flow rate versus pressure. Id. Moreover, since the Staff does not even know what input assumptions were used, they are in no position to select or defend any particular uncertainty estimate. We also question why EG&G was not asked to estimate the appropriate uncertainty, since EG&G did the rest of the work.

105. Finally, UCS attempted to show that no RELAP5 type analysis has been done to determine whether long-term feed and bleed at TMI-1 would cause conditions leading to pressurized thermal shock to the reactor vessel. App. Tr. 300-303. While we sustained objections to this at the time, we have re-evaluated the issue and now conclude that it is a significant one.

106. Question 11 asked for analysis to determine whether feed and bleed "will successfully provide core cooling at TMI-1." It is apparent that feed and bleed, which involves operating at high RCS pressure, poses a risk as the RCS cools down, of exceeding the pressure and temperature limits for the reactor vessel. Obviously, if the vessel cracks at some point during feed and bleed, one cannot conclude that the core has been successfully cooled or will continue to be successfully cooled.

107. This was previously acknowledged by the very witness presented by the Staff. In a memorandum from Mr. Sheron to Carl Kniel, dated March 31, 1981, Mr. Sheron stated: "High pressure feed and bleed is not recommended due to vessel structural considerations. Feed and bleed should be performed at lower pressures." App. Tr. 318-319, UCS offer of proof. Of course, feed and bleed at TMI-1 would be at high pressure - about 2500 psi.

108. Considering that the Staff's own witness had reached the conclusion that feed and bleed as proposed for TMI-1 should not be recommended, we do not understand how the Staff could support the conclusion that feed and bleed can be relied upon for decay heat removal in this hearing.

109. For all of the reasons stated above, the Board concludes that there is no RELAP5 analysis which provides us with sufficient assurance that feed and bleed will successfully cool the core at TMI-1 for loss of feedwater and small break LOCA events.

110. Appeal Board Question 8 seeks "clarification of the apparent inconsistencies and confusion concerning the safety-grade status of components in the EFW system."

111. The Licensee testified that all of the assertions made in Lic. Ex. 15 with respect to the safety grade status of the EFW system components remain true. App. Tr. 325, Chisholm. This testimony is not the whole truth, however but reflects GPU's and the Staff's view that the scope of this proceeding is limited. Licensee Ex. 15 contains the following statement: "The EFW piping system is however, designed and qualified to the seismic Class one requirements." App. Tr. 325. That statement is false. See App. Tr. 345, Wermiel. Therefore, one cannot conclude that the EFW system is safety grade. Id.

112. In addition, even with respect to small break LOCAs, the EFW flow control valve is not safety grade. There is only one valve per steam generator, in violation of the single failure criterion. In addition, the associated circuitry is not safety grade. App. Tr. 344-5, Wermiel.

113. The Licensee testified that the non-safety grade manual control capability for the EFW valves is adequate because there is adequate time to dispatch an operator to the vicinity of the EFW control valves to manually manipulate the valves. Capodanno and Chisholm, ff. Tr. 324 at 3, 4, 5.

114. In reaching this conclusion, the Licensee did not determine or consider the other tasks which this operator may be required to perform, (App. Trr. 337, Chisholm and Dempsey) although the applicable procedures do give the same operator other tasks. App. Tr. 328-332, Dempsey. The Licensee's statement that the operator has sufficient time was limited to assuming that he has to perform only the task of manually controlling the EFW flow control valve. App. Tr. 338, Dempsey. This assumption is simply not a realistic one.

115. The Staff witness testified that the EFW system is not safety grade because, inter alia, it is not seismic Category 1. App. Tr. 345, Wermiel.[9]/ The upgrade to fully safety grade is necessary for safety (Id.) but there is not even a schedule yet for seismically qualifying the EFW system. The Staff does not yet know what needs to be done. App. Tr. 348-9.

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[9]/ UCS does not, of course, waive any of its earlier exceptions in this area and draws the Board's attention to the safety implications of the fact that there is only one EFW flow control valve for each steam generator. A main steam line break inside containment requires isolation of all feedwater to the affected steam generator to prevent overpressurization of the containment. H. Hukill to J. Stolz, Aug. 2, 1982, Attachment 1, p. 1. incorporated into Licensee's Response to Appeal Board Order of July 14, 1982 at 20. The MSLRDS automatically isolates main feedwater to the affected steam generator. Since the MSLRDS is not safety grade, one must assume a failure. If one assumes that the failure involves isolation of main feedwater to the intact steam generator and the operator then, following his procedures, isolates emergency feedwater to the affected (footnote continued next page)

116. The systems and components involved would include at a minimum the interface between the suction supply piping from the condensate storage tank and the condenser, the recirculation piping to the condensate storage tank, the controls for the turbine-driven pump, the de-ice line to the condensate storage tank and certain cabling to valves. App. Tr. 341, Wermiel.

117. In its earlier quantitative analysis of EFW reliability, which was relied upon by the ASLB, the Staff claimed that seismic considerations were not important to its conclusions on reliability because the probability of an earthquake is lower than other potential system failures more important to the fault trees. Tr. 16, 769, Wermiel. UCS attempted to show that this is not true if the earthquake in question is an operating basis earthquake, which has a probability higher than many events that have an important effect on the pertinent fault trees. App. Tr. 355, Weiss. An OBE is "that earthquake which could reasonably be expected to affect the plant site during the operating life of the plant." 10 CFR Part 100, App. A, III (d). Lawrence Livermore found that the TMI-1 EFW cannot even withstand an OBE. Technical Evaluation Report, Three Mile Island Nuclear Station, Unit 1. Seismic Qualification of Auxiliary Feedwater System. See UCS Comments on the Commission's <u>Ex Parte</u> Meeting of December 17, 1982...", January 7, 1983, 21-23.

118. Therefore, the available evidence leads to the conclusion that the EFW system is substantially less reliable than assumed by the ASLB.

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steam generator, there remains only <u>one</u> feedwater flow path: emergency feedwater to one steam generator. A failure of the non-safety grade EFW flow control valve results in total loss of feedwater to <u>both</u> steam generators. This scenario does not involve postulating anything beyond what NRC requires for safety analyses. Indeed, since the EFW flow control valves are not safety-grade, even an additional failure could be postulated. <u>See</u> UCS Response to ALAB-708, June 19, 1983 at 3-4.

119. We now proceed to the questions concerning the boiler-condenser mode of cooling. The Licensee discusses results from three different models throughout its testimony. During questioning, these were referred to as 1) the "approved model" (pre-TMI), 2) the "revised model," which modified the noding by adding one node in each RCS loop extending from the mid-point of the 180° U bend to the vessel outlet nozzle, Jones, ff. Tr. 453 at 3, App. Tr. 466-7, and 3) the "new model" that has recently been submitted for Appendix K approval. App. Tr. 462.

120. Appeal Board Question 4 asks "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 10CFR Part 50."

121. The short answer is "no." App. Tr. 470, Jones. The "modified" or "revised" post-TMI model is not approved under Appendix K. Id.

122. Appeal Board Question 5 asks "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."

123. The Staff did not re-review the B&W model after the TMI-2 accident. Sheron and Jensen, ff. Tr. 83 at 5. It asked B&W to look at smaller breaks than those used for the Appendix K analysis "to provide a basis for revisions to small break LOCA emergency procedures." Id.

124. The RELAP4 audit calculation for the .01ft.^{2} break did not predict interruption of natural circulation. App. Tr. 554, Jensen. Obviously, then, the ability of boiler-condenser to effectively cool the core after natural circulation is interrupted could not be confirmed by these audit calculations. The audit calculations did not show the establishment of a condensing surface. App. Tr. 561, Sheron.

125. The B&W calculation of the .01ft.^{2} break predicted different plant behavior than did the RELAP4 audit calculation. App. Tr. 550-8, Sheron and Jensen. It predicted interruption of natural circulation. The operator would see a difference in the behavior of pressurizer level for these two different predictions of plant behavior. App. Tr. 557-8, Sheron. The Staff cannot account for the difference in results in terms of plant behavior during a .01ft.^{2} break between the B&W and RELAP4 audit calculations. App. Tr. 559, Sheron and Jensen.

126. The answer to this question, then, is that the Staff has not reviewed the approved Appendix K model or the revised model to determine the ability of the code to calculate the effects of small breaks including reliance upon boiler-condenser circulation. Neither witness could even say whether the calculations done with B&W's approved model showed any instance where boiler-condenser was needed. App. Tr. 562. The Staff has done only a few audit calculations which never even predicted significant interruption of natural circulation. And, of course, the Staff has not yet reviewed the recently-submitted new B&W Appendix K model. App. Tr. 503-4, Jones, 628, Sheron.

127. The Staff nonetheless opines that the "heat transfer mechanisms involved in the boiler-condenser process are adequate to remove decay heat..." Sheron and Jensen, ff. Tr. 83 at 2. While conceding that "detailed reactor coolant system behavior during the period of natural circulation interruption in the analysis of certain small break sizes is not well understood," (Id.) the Staff states that the system must eventually drain down and expose a condensing surface in the steam generator before the core could begin to be uncovered. Id.

128. There are certain implicit assumptions here -- a critical one being that

the operators do not take any action which might interfere with this process. An apt analogy is to the TMI-2 accident itself, where the operators acted on the basis of misleading signals and the fear of going solid and terminated HPI. UCS attempted to show that operators could be misled again into taking incorrect actions and that the current procedures are inadequate to ensure appropriate operator action, but were prevented from pursuing these questions. App. Tr. 584-588. We recognize, however, that the Staff's (and B&W's) descriptions of expected plant behavior are theoretical.

129. The Staff, through Mr. Eisenhut, wrote to the B&W owner's group approximately a year ago, sending the owners documentation of the Staff's concerns with the B&W small break model. UCS Ex. 51. The Staff witnesses stated that the following areas discussed in that document are still Staff concerns with regard to the ability of the B&W models to predict small break LOCA behavior: (App. Tr. 646-647, Sheron)

1. interruption of natural circulation
2. hydraulic stability following an accident
3. cooldown and depressurization following a small break
4. steam generator tube rupture

130. A perusal of the document suggests the scope and fundamental nature of these as yet unanswered questions concerning the validity of the B&W code. This increases our doubts concerning reliance on the code's predictions regarding plant behavior after an interruption of natural circulation.

131. Appeal Board Question 6 was "whether only breaks slightly smaller than $0.07ft.^2$ must be analyzed?"

132. Both the Licensee and the Staff responded in terms of the requirements of 10CFR 50.46 and Appendix K to Part 50. They did not discuss in direct testimony the section of 50.46 which covers long-term cooling:

> After any calculated successful initial operation of the ECCS, the calculated core temperature shall be maintained at an acceptably low value and decay heat shall be removed for the extended period of time required by the long-lived radioactivity remaining in the core. 10CFR 50.46(b)(5).

133. For certain break sizes at TMI-1, (0.02ft.^{2} or less), high pressure recirculation is required, App. Tr. 471-2, Jones. This might be required for less than a month, depending on break size. App. Tr. 472-3, Jones. No computer analysis over this period of time has been performed to demonstrate conformance with 10CFR 50.46(b)(5). Id. at 473.

134. In any case, the argument that smaller breaks need not be analyzed is entirely dependent upon the premise that boiler-condenser will prevent core uncovery for smaller breaks. See Sheron and Jensen, ff. Tr. 83 at 7-8. It is thus a circular argument: smaller breaks need not be analyzed pursuant to 50.46 by an approved Appendix K code because our other analyses show that core uncovery will not occur. Using this logic, one could argue that no breaks need be analyzed pursuant to 50.46 and Appendix K since none will produce core uncovery.

135. Considering that the small break experienced during the TMI-2 accident produced worse consequences than any previously analyzed and disclosed the need for means of long-term and high-pressure recirculation, (feed and bleed, boiler-condenser), this seems to us to be an evasion of 10CFR 50.46 which, as the Staff concedes, is intended to ensure that the "worst break size is analyzed." Sheron and Jensen, ff. Tr. 83 at 7.

136. Appeal Board Question 7 asked for "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."

137. First, there are no applicable test results from a facility geometrically similar to TMI-1 which would confirm the efficacy of boiler-condenser. Sheron and Jensen, ff. Tr. 83 at 8. For a small break LOCA, such testing is important because plant behavior in the transition to boiler-condenser cannot be confirmed without such test results. App. Tr. 590-1, Sheron.

138. The GPU and Staff testimony presented the results of several computer calculations using different codes, very different noding arrangements (App. Tr. 492-3, Jones), and different input assumptions for critical plant parameters. App. Tr. 629-30, Sheron. They predict very different kinds of plant behavior, as is described below. There is no analysis of a small break for TMI-1 that assumes steam generator level is taken to 95% on the operating range and one HPI pump is operated. App. Tr. 613-4, Jensen. While the B&W and EG&G calculations predict that the core remains covered, this does not resolve the question for several reasons.

139. First, the B&W code which has been reviewed by the staff makes assumptions regarding heat transfer in the steam generators which do not correspond to reality. These will be discussed below. Second, the EG&G calculation done with RELAP5 cannot be said to "confirm" boiler-condenser, since it doesn't show this phenomenon occurring, but instead predicts a bizarre chugging behavior that seems physically impossible to the Board. App. Tr. 597-601, 707-732, Sheron and Jensen. Nor, of course, does the EG&G computer analysis "confirm" the B&W analysis for the same reason. Basically, the EG&G analysis is, as Mr. Ornstein characterized it, an "outlier." App. Tr. 788. It raises more questions than it answers.

140. Basically, one is still left with the approved and revised B&W model calculations, which utilize highly dubious methods for calculating heat

transfer in the steam generators, for which there is no experimental verification,[10] and as to which even the staff continues to have substantial concerns. We will now discuss the evidence in greater detail.

141. Both the approved and revised B&W models use a fixed heat transfer coefficient and a fixed heat transfer area for calculating primary to secondary steam generator heat transfer in the boiler-condenser mode. App. Tr. 482-5, Jones. This is fixed throughout the calculation. Id. at 485. Heat transfer is calculated solely as a function of the code's calculated difference in temperature between the primary and secondary side. Id. at 488. Moreover, even in the revised B&W model, the entire secondary side of the steam generator is modeled as one node. Id. at 487.

142. In contrast to these simplifications, during boiler-condenser, portions of the steam generator tubes would be surrounded by water, portions by steam, and portions sprayed by emergency feedwater. Id. at 488. On the primary side of the tubes, portions will contain steam, portions water, and condensed steam will be running down the inside of some. Id. at 489-90. This will greatly affect actual heat transfer by factors of a hundred (App. Tr. 517, Jones), but the B&W code cannot account for it. Id. at 490, 492. The presence of condensate film, for one thing, will decrease heat transfer. Id. at 490. In addition, these analyses assume that all tubes are wet by EFW. Id. at 481-2. We have no basis for confidence that the approved, revised, or new B&W models accurately predict heat transfer in the steam generators during boiler-condenser.

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[10]/ We cannot place reliance on the newly-submitted B&W Appendix K model which has not been reviewed yet. The Staff has no opinion as to whether it is accurate or adequate under Appendix K. App. Tr. 628, Sheron. Moreover, the topical report on the new computer code is proprietary and was not even made available to the Board. App. Tr. 503-4, Jones. Only one break size has been analyzed by B&W with this new model, and that was a generic calculation -- .01ft.^{2} App. Tr. 504-5, Jones.

143. Moreover, our consideration of the B&W model highlighted the critical importance of assumptions made regarding operator action and our inability therefore to confirm whether the plant behavior predicted by the code will conform to an actual event as it might unfold in the plant. The B&W calculation assumes that EFW is automatically initiated (App. Tr. 520-1), that steam generator level is raised to 95% on the operating range, and then throttled to maintain that level. Id. at 498-9. It assumes that the operator controls the steam dump and EFW flow. Id. at 500-1.

144. UCS questioned the GPU witness about the consequences of assuming that steam dump is stopped and EFW flow throttled: how long would it take to reach thermal equilibrium between the primary and secondary sides of the steam generator, with little or no heat transfer occuring? While the witness characterized the question on timing as "unanswerable" because of the multitude of variables involved, he agreed that it is true that with the primary and secondary side liquid levels at the same elevation and no EFW flow, there would be little or no heat removal. Id. at 500-502.

145. It is apparent that the conclusions on boiler-condenser cooling reached by B&W are substantially dependent upon assumptions concerning operator action. UCS was not given an opportunity to probe these assumptions.

146. GPU's conclusions are further based on the premise that once the system is "moving in a depressurization manner," the situation will always get better. App. Tr. 532, Jones. This may well be an oversimplification. The system may well repressurize and cycle back and forth into boiler-condenser. Id. at 532-3. The B&W code cannot calculate the transition from boiler condenser back to liquid natural circulation. Id. at 542-3.

147. The staff has postulated that, in recovering from feed and bleed, after restoration of emergency feedwater, a noncondensible steam bubble could form,

preventing re-entry into natural circulation. App. Tr. 682-3, Sheron. Therefore, the ATOG guidelines for B&W plants tell the operator to turn on the reactor coolant pumps to restore circulation. App. Tr. 681.

148. An analogous situation could occur relevant to the boiler-condenser analysis. One assumes a small break with no feedwater, progressing into feed and bleed. At some point, the primary side steam generator level is at about 50% on the operating range. If one assumes that EFW is then restored, there is no difference in the plant's condition between that situation and one where EFW was always available and the plant were operating in the boiler-condenser mode. App. Tr. 689-92, Sheron. Therefore, Mr. Sheron's concerns should apply to this situation as well. UCS was prevented from pursuing this line of questioning. App. Tr. 693. In any case, if the operator is told to "bump" the reactor coolant pumps at this point, it would seem dubious to assume that the plant would be going into boiler-condenser.

149. The GPU witness stated that the approved B&W model was conservative as to heat transfer because it overpredicted heat flow from the secondary to primary side. App. Tr. 541, Jones. While this is conservative pursuant to Appendix K with respect to large breaks, it is not necessarily conservative for small breaks.

150. We now proceed to the Staff's testimony. The Staff hired EG&G to do a RELAP5 analysis of a .01ft.^{2} break. Sheron and Jensen, ff. Tr. 83 at 9. The objective was to duplicate B&W's analysis using the revised model (Id.), but, in fact, neither the input nor the results were duplicated.

151. The inputs used to model plant parameters were different from the GPU analysis, which was B&W generic. App. Tr. 606-7, Sheron. This was done because the available time did not permit using the GPU assumptions. Id.

152. The RELAP5 calculation predicted very diffent plant behavior after a

SBLOCA; it predicted no interruption of natural circulation and a bizarre "chugging" phenomenon which appears physically impossible or at the very least extremely improbable to the Board. App. Tr. 597-601, 707-23, Sheron. The Staff's two witnesses had different explanations of what would actually be happening in the plant during this "chugging." Contrast 707-725, Sheron with 723-732, Jensen. The Staff has not received the written report from EG&G, App. Tr. 624-5, Sheron, hasn't looked at the data from the calculation, App. Tr. 615-6, Jensen, and hasn't even questioned EG&G to determine the physical possibility of the behavior predicted. App. Tr. 717-8.

153. The Staff believes that this demonstrates that the codes are incapable of calculating the phenomena that occur in transition to boiler-condenser or even whether boiler-condenser is achieved. App. Tr. 622, Sheron. See also App. Tr. 705-6, Sheron. We agree. We note also in this connection that, even if boiler-condenser is established, subsequent repressurization and inventory recovery might lead to loss of a condensing surface and could eventually put the plant into feed and bleed. The Staff did not believe this likely, but was only "speculating" on "that whole end" of the scenario. Id. at 739-40.

154. The Staff's testimony must also be criticized on several other grounds. Since RELAP5 was unable to predict stable boiler-condenser, Mr. Jensen performed an analysis which essentially forced the plant into the conditions necessary for boiler-condenser. Sheron and Jensen, ff. Tr. 83 at 17-18. The "hypothetical transient" was unrealistic; it would not actually happen in the plant, as he admitted in questioning. App. Tr. 631, Jensen.

155. In addition, Mr. Jensen's testimony as originally filed presented an evaluation of the capability of the TMI-1 steam generators to remove decay heat in the boiler-condenser mode which consisted of a "scoping heat transfer calculation." Sheron and Jensen, ff. Tr. 83 at 20-21. As originally

presented, the testimony was that 18% of the steam generator tube surface area would be required to remove 2.5% of full power and that 27% would be available. Id.

156. It was necessary for Mr. Jensen to revise his testimony because, contrary to his original assertion, only 15% of the total tube surface area would be available.[11] Had Mr. Jensen kept the rest of his original testimony unchanged, it would have concluded that 18% was required and 15% was available, leading, of course, to the conclusion that a sufficient condensing surface will not be available. App. Tr. 642, Jensen.

157. Instead, Mr. Jensen altered the other assumptions of his "analysis" as necessary to support his foreordained conclusion. He changed his assumption for the temperature difference between primary and secondary system from 10^{o} to 20^{o} and changed the decay heat rate. Id. at 640-2. The former assumption had to be changed to get from the previous estimate of 18% of tube surface required to only 7%. Id. at 642. No justification was offered for this change. We give no weight to this analysis; on the contrary, it raises serious questions in our minds concerning this witness' credibility. In addition, we are concerned with how close a call this may be, considering that Mr. Jensen's original assumptions, used with a more accurate estimate of available tube surface, would indicate that sufficient tube surface is not available.

158. Just a few days before this hearing began, the Staff filed Board Notification BN-83-21. This notification arose from the Staff's review of the

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[11]/ Mr. Jensen stated that he realized it is "more meaningful" to discuss the condensing surface between the top of the 95% level on the operating range and the cold legs, rather than the top of the core. App. Tr. 640.

transcripts of the trial between B&W and GPU in New York regarding liability for the TMI-2 accident.

159. BN-83-21 identified issues which the Staff believed were potentially significant with respect to achieving and maintaining natural circulation in B&W designed reactors. BN-83-21A, March 11, 1983, Enclosure at 1.

160. The Staff characterized the concerns raised by Dr. Lahey of R.P.I. as follows:

> The first concern was raised by Dr. Lahey. It deals with procedures and relates to whether or not the operators have sufficient instructions and training to assure that they will raise the secondary level of the steam generator to 95 percent of the operating level under all conditions necessary to assure natural circulation. Following the TMI-2 accident, it was learned that the then current procedures instructed operators to raise the secondary level to 50 percent of the operating range. Under certain circumstances, it was possible to postulate that natural circulation would not be reestablished with the secondary level at 50 percent. Subsequently, it was determined that raising the level to 95 percent of operating range would assure natural circulation if the RCS was saturated. <u>However, because of overcooling considerations, it is not desirable to raise the level to 95 percent for all cases of loss of forced circulation. Thus, specific plant circcumstances dictate the appropriate steam generator level and the manner to achieve htis level. The operating procedures and training to describe the correct actions are, therefore, important to the issue</u>. <u>Id</u>. at 2. (emphasis added).

161. Thus, two concerns are raised by Dr. Lahey. One goes to whether the operators will raise steam generator level to 95% when necessary. The other recognizes that raising level to 95% is not desireable for all cases of loss of forced circulation and that, therefore, operating procedures and training that account for specific plant circumstances, distinguish between the two cases, and describe the correct actions are important.

162. In response to UCS's motion, the Appeal Board directed the Staff to

respond to these concerns before the hearing on issues related to boiler-condenser cooling. App. Tr. 37-38. That response was BN-83-21A, March 11, 1983. The Staff did not respond at all, however, to the issue concerning the need to ensure that the operating procedures and training are sufficient to describe the correct actions to the operators, considering that raising steam generator level to 95% is not always desirable.

163. The Staff simply claims that it reviewed selected procedures to confirm that the TMI-1 procedures do instruct the operator to raise steam generator level to 95%. BN-83-21A at 9. However, the Staff witnesses at the hearing had no knowledge of what actually was done for this review and could offer no judgment on whether the Staff response in BN-83-21A is adequate. App. Tr. 652-5, Sheron. Moreover, the Staff and GPU successfully objected to any questions touching on the plant procedures. App. Tr. 653-64. UCS was denied any opportunity to probe at all into the accuracy of the Staff's claim that it has resolved the Lahey issues. We cannot conclude that these issues have been satisfactorily resolved without affording UCS such an opportunity.

164. UCS cannot be presumed to have the resources to file another motion to reopen the record as the Appeal Board suggested would be an appropriate remedy (App. Tr. 661), nor is this UCS's obligation. These issues are fairly before the Board now. One cannot conclude that boiler-condenser or feed and bleed are sufficiently reliable means of decay heat removal without determining, either implicitly or explicitly, that the assumptions made in computer analyses regarding the condition of the plant at the critical times conform to what actually would be happening in the plant, particularly when crucial assumptions are made with regard to operator actions (e.g. proper control of EFW flow, steam dump valve, HPI pumps, etc.). When UCS was denied an opportunity to test the degree to which these assumptions are correct, it has

been denied the opportunity to test the reliability of the computer analyses. If computer analyses in the abstract were sufficient, one has essentially denied that TMI-2 ever happened; the computer analyses classed it as impossible by, among other things, failing to consider the role of the operators. The precise mistake is being repeated.

165. Mr. Ornstein gave some testimony which is enlightening on this point. Even though the B&W code before the TMI-2 accident might have predicted boiling in the core, "People weren't doing too much with what they were finding. Particularly the thing I have in mind is the simulator at Lynchburg and the codes that went into that. But most operators, if you talk to them about a PWR, and talk about boiling taking place in there, they look at you like you were crazy." App. Tr. 782-3, Ornstein.

166. We now address the testimony of Harold L. Ornstein, lead systems engineer in the Office for Analysis and Evaluation of Operational Data (AEOD). Professional Qualifications of Harold L. Ornstein, ff. Tr. 742. Mr. Ornstein, who is in charge of reviewing event reports and other information relating to the B&W reactors (Id.) appeared as a witness for UCS pursuant to a subpoena issued by this Board to present the ACRS views on reliance on feed and bleed and boiler-condenser for TMI-1. He is the author of a memeo raising AEOD concerns about both. UCS Ex.53.

167. The AEOD memo states that the conclusion that natural circulation would be established in the boiler-condenser mode is not a certainty, especially in the absence of experimental data for B&W plants. If natural circulation cannot be established and the reactor coolant pumps are not available, feed and bleed would have to be used. App. Tr. 746, UCS Ex.53. AEOD's opinions remain today as stated in UCS Ex.53. Id. at 752.

168. AEOD's concerns regarding reliance on boiler-condenser arise from the

fact that there are many computer analyses which have a high degree of sensitivity to input parameters (Id. at 747) but AEOD is uncomfortable since there has been no demonstration that what is postulated in theory would actually happen. Id. at 748. And, even if there was a demonstration of one particular break size, with a given set of parameters, that doesn't tell us what will happen with different break sizes. Id. Therefore, the Staff should not give the impression that we can always establish this mode of cooling. Id.

169. The witness agreed that one cannot predict from the collection of computer analyses available, with different parameters and different results, how the plant will behave over a spectrum of SBLOCA's. Id. at 750. The EG&G RELAP5 analysis predicting "chugging" was characterized by Mr. Ornstein as an "outlier." App. Tr. 788. Such an outlier cannot be used to confiirm the events of other codes which predicted very different plant behavior.

170. AEOD "wanted to understand more about the stoppage of natural circulation; we wanted to know more about the re-establishment of circulation; we wanted to know more about how the operators would be able to determine where they were and what they had to do." Id. at 758.[12] These also seem to the Board to be very important questions which have not been answered. Basically, a better understanding of the physical phenomena in question is required. See App. Tr. 759, Ornstein.

171. AEOD also raised the need for obtaining experimental verification of the ability of the PORV and safety valves to perform reliably in feed and bleed. Id. at 759-60, UCS EX.53, item 7, page 2. The TMI-1 safety valves were not

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[12]/ In this connection, UCS asked Mr. Ornstein whether AEOD has reached any judgment about whether, for TMI-1, we can have confidence that the operators would understand what was going on and would take the correct action. GPU objected and the objection was sustained. App. Tr. 789.

only not designed for liquid flow, the manufacturer refused to guarantee them for water operation under any conditions. Id. at 761. While AEOD believes experimental verification is necessary, they have reached no conclusion as to whether the EPRI tests have resolved their concerns. Id. at 763-4. We have explored this issue above and concluded that there is not sufficient basis for concluding that these valves will perform as required.

172. In general, Mr. Ornstein's testimony confirmed the need for experimental testing before this Board could conclude that boiler-condenser is a sufficiently reliable means of decay heat removal. The EG&G RELAP5 analysis does not confirm the accuracy of the B&W codes. The fact that both RELAP5 and B&W predict adequate core cooling is not conclusive, given the difference in codes, assumptions used, and plant behavior predicted and the great sensitivity of the analyses to the assumptions Mr. Ornstein noted. The fact that two different, highly uncertain and doubtful analyses both predict X does not provide proof of X. One or both of the analyses must be convincing on its merits. In this case, as Mr. Ornstein stated, "seeing is believing." App. Tr. 749.

173. We conclude that there is no experimental testing confirming that boiler-condenser circulation flow will adequately remove decay heat. We conclude further that the computer analyses done to date are not sufficient without experimental testing to provide assurance of the viability of boiler-condenser.

Respectfully submitted,

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Date: April 12, 1983

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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

CERTIFICATE OF SERVICE

I hereby certify that copies of "UNION OF CONCERNED SCIENTISTS' PROPOSED FINDINGS OF FACT AND CONCLUSIONS OF LAW ON REOPENED HEARING" have been served on the following persons by deposit in the United States mail, first class postage or as indicated by asterisk, this 13th day of April, 1983.

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