UCS 3/19/84

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE COMMISSION

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

UNION OF CONCERNED SCIENTISTS' BRIEF ON THE COMMISSION'S REVIEW OF ALAB-729

INTRODUCTION

By order dated January 27, 1984, the Commission took review of certain issues decided in ALAB-729 and ALAB-744. Those issues are briefed below. It is disheartening to note that the formulation of most of the issues foreshadows the possibility that the Commission may avoid grappling with the substance of the serious safety and policy questions presented by manipulating the scope of the hearing to exclude them. Indeed, the history of this proceeding shows a successive narrowing of the scope until it excludes all issues that cannot be "favorably" resolved.

The buck, of course, stops with the Commission. Putting aside the question of whether the rules of this proceeding have been a matter of discretion or are dictated by law, the limits of discretion are not so broad as to allow the agency to turn a blind eye to serious safety deficiencies in

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TMI-1. We particularly urge the Commission to give the fullest consideration to the fundamental question of whether there is assurance of a highly reliable means of decay heat removal at TMI-1.

ARGUMENT

1. Environmental Qualification of Safety Equipment

The operative facts relevant to this issue are straightforward:

1. UCS had a contention admitted in this proceeding, which was later adopted and amplified by the Licensing Board, claiming that the pre-accident means of determining the qualification of safety equipment to survive severe accident environments were inadequate to meet GDC 4 and that TMI-1 should not be permitted to operate until GDC 4 was met. Board Question/UCS Contention 12 quoted at ALAB-729, 17 NRC 814, 891-2 (1983).

2. UCS prevailed on this Contention; indeed, the facts were virtually uncontested. No party even attempted to show that TMI-1 meets GDC 4 or that all safety equipment is environmentally qualified. LBP-81-59, 14 NRC 1211, 1409, para. 1181, (1981). Nor was factual evidence submitted by any party to justify a conclusion that TMI-1 is sufficiently safe to operate despite noncompliance with GDC 4.

Indeed, the Staff "defaulted" on the issue. Id. at 1402, para. 1155. While the licensee's witness offered the sanguine view that 95% of the TMI-1 equipment was documented to be qualified to the standards adopted in CLI-80-21 and that the remainder would be resolved by February 1, 1981 (Id., at 1400, para. 1149), his testimony was so vague that the Board concededly could not

rely upon it even to reach "a qualitative judgement of the risk of allowing interim operation prior to June 1, 1982." Id., at 1403, para. 1157.[1/]

Faced with the undeniable reality that the factual record of this proceeding contains no basis for a finding on this issue that TMI-1 can be operated without posing undue risk -- indeed, the "facts" are to the contrary -- both the Licensing and Appeal Boards have sought refuge in the Commission's generic pronouncements on environmental qualification. In so doing, they have alternatively misunderstood and misapplied those pronouncements.

In CLI-80-21, 11 NRC 707 (1980), the Commission, in response to a UCS petition, adopted strict new industry-wide standards for establishing environmental qualification and set a compliance deadline of June 30, 1982. The Commission also specifically directed the staff to continue a detailed review of the status of plant equipment and directed further: "These deadlines, however, do not excuse a licensee from the obligation to modify or replace inadequate equipment promptly."[2/] 11 NRC 707, 715, emphasis added.

The Commission found that the actions taken in CLI-80-21 "provide reasonable assurance that the public health and safety is being adequately

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1/ Moreover, the licensee's testimony was false. As of November 5, 1982, almost two years after Braulke's testimony, the Franklin Research Institute concluded that of 120 "equipment items" (i.e., types of equipment) in TMI-1, only 2 "equipment items" were fully qulified, 1 was "qualified pending modificatiion," 40 had not established qualification, 19 were deficient for qualified life, 25 were "exempt," 22 were "not in the scope of the review" (i.e., not reuired for cold shutdown and/or exposed only to mild environment) and for 11, insufficient documentation was supplied to categorize them. Technical Evaluation Report, Review of Licensees' Resolution of Outstanding Issues from NRC Equipment Environmental Qualification Safety Evaluation Reports, TMI-1, Vol. 1, p. 4-3, Nov. 5, 1982. A copy of the pertinent page is attached. This material was provided to the Commission via Board Notification 82-133.

2/ UCS is baffled by the Appeal Board's statement that UCS "argues that" CLI-80-21 did not excuse licensees from the obligation to promptly replace inadequate equipment. This is not an "argument." It is a direct quotation from CLI-80-21 and thus is a fact.

protected during the time necessary for corrective action." The Licensing Board cited this language (14 NRC at 1399, para. 1145) and later concluded that "the question of interim operation has already beeen addressed and decided by Commission Order CLI-80-21." Id. at 1409, para. 1181.

But this is manifestly an erroneous and legally impermissible interpretation of CLI-80-21. The Commission had not a shred of evidence before it on the status of safety equipment in TMI-1. At most CLI-80-21 represents a statement of policy that if all equipment is fully qualified by June 30, 1982, and if the staff in the meantime ensures prompt replacement of equipment discovered during the review process to be unqualified (as opposed to temporarily missing some documentation), those actions make a generic shutdown of plants unnecessary. The Commission did not and could not have used CLI-80-21 to resolve factual issues properly raised in a plant-specific proceeding. Surely, the Commission could not by simple edict overcome the facts in this record. Cf. Minnesota v. NRC, 602 F.2d 412 (D.C. Cir., 1979).

Subsequent to the Licensing Board's decision, the Commission suspended the June 30, 1982, deadline for compliance with the new environmental qualification standards. It based that industry-wide suspension, inter alia, upon the following determination:

> The Commission has received, and the staff has evaluated, each operating plant licensee's justification for continued operation. On the basis of the analyses,3/ the Commission has determined that continued operation of these plants pending completion of the equipment qualification program, will not present undue risk to the public health and safety.
> 47 Fed. Reg. 28363 (June 30, 1982).

3/ It has recently been conceded by the staff that these "analyses" consisted of determining whether licensees had asserted that the plant could be safely operated. No "analysis" of the basis for these claims was made by the staff. Discussion/Possible Vote on Equipment Qualification Policy Statement and Proposed Rule, NRC Response to Court of Appeals Decision, Transcript of Commission Meeting, January 6, 1984, pp. 63-64.

The Appeal Board cited this language and concluded that the issue had thus been removed from this case. 17 NRC at 893-4. The only possible relevant distinction between the above-quoted pronouncement of June 30, 1982, and the earlier-quoted pronouncement of CLI-80-21 is that the former at least asserts that plant-specific data had been reviewed. However, in UCS v. NRC, D.C. Cir. No. 82-2000, June 30, 1983, the Court held that this determination of plant-specific safety was both necessary as justification for the Commission's deadline suspension and was unlawfully made without any opportunity for public participation, in violation of the Atomic Energy Act, the Administrative Procedure Act and the NRC's own rules. Id., Sl. op. at 19. It therefore obviously cannot serve as the basis for a conclusion that TMI-1 is currently in a condition to operate without undue risk to the public.

In ALAB-744, the Appeal Board rejected UCS' request that it reconsider its treatment of this issue. The Appeal Board's reasoning appears to center on its belief that "[t]he Commission originally decided to address the issue of environmental qualification generically," and therefore the Licensing Board had no obligation to look at the current safety of TMI-1. ALAB-744, October 6, 1983, Sl.op. at 3. Again, this represents both a misconstruction of both CLI-80-21 and the Commission's "determination" of June 30, 1982, and an evasion of the crucial legal issue.

The fact is that the Commission has always perceived and based its actions upon a clear understanding that environmental qualification is both a generic and a plant-specific issue. In CLI-80-21, it took the generic action of adopting an industry-wide standard and deadline. But it also understood that the agency has an overriding duty at all times to ensure the safety of each individual plant and therefore it directed that the generic deadline not function as an excuse to operate when equipment in individual plants is

discovered to actually be unqualified or when documentation is poor. In the former case, prompt replacement or modification was required by the Commission and in the latter, a "technical judgment" justifying operation. 11 NRC 707, 715.

On June 30, 1982, the Commission again dealt with both the plant-specific and generic dimensions of the issue. Generically, it suspended (and later extended to 1985 or beyond) the industry-wide compliance deadline. However, it also made the determination quoted above, purportedly based on plant-specific staff evaluations, that each plant had individually justified interim operation. Those plant-specific determinations were a necessary precondition to generic deadline extension and were struck down by the court.

Moreover, a fundamental truth seems to have become lost in the muddle of buck-passing that has characterized the NRC's tortuous evasion of the question posed by UCS. The evidence in this case shows that GDC 4 is not met at TMI-1 and no-one, not the staff, the licensee or the Boards have pointed to any <u>facts</u> which show that the plant is safe enough to operate nonetheless. The Commission edicts made without providing parties the right of participation are not a lawful way to resolve issues fairly raised in a hearing. <u>Minnesota v. NRC</u>, 602 F.2d 412 (D.C. Cir. 1979).

Perhaps the most bizarre episode in this entire shell game is the fact that even the staff has refused to present technical justification for operation of TMI-1. When the Staff was given the opportunity by the Appeal Board to explain how adequate protection of the public health and safety is provided pending resolution of one aspect of this issue -- the lack of an environmentally qualified pathway to cold shutdown -- the Staff, going full circle, stated that the Commission had "itself determined" that plants could operate safely despite lack of environmental qualification. NRC Staff's

Response to Appeal Board's Order of July 14, 1982, Affidavit of Zoltan R. Rosctoczy, p. 3, August 9, 1982.

This issue was fairly raised in the restart hearing. It clearly has a "nexus" to the TMI-2 accident and, just as clearly, UCS has prevailed. The licensee has not remotely approached meeting its burden of proving that safety components in TMI-1 are capable of surviving a TMI-2 type accident or that the plant is nonetheless sufficiently safe to operate. It cannot be seriously disputed that any intervenor in any licensing case is free to raise and litigate a plant-specific safety issue concerning environmental qualification, just as any intervenor is free to demonstrate that failure to meet any safety requirement is a basis for denying or modifying a license. Moreover, in the TMI-1 restart case, intervenors were also free to challenge the sufficiency of the NRC requirements to protect safety, in light of the TMI-2 accident. CLI-79-8, 10 NRC 141, 148 (1979). This important safety issue was raised by USC, has not been met, and cannot be lawfully "removed".

The Commission has asked what the "proper scope" of the contention is. The scope of the contention, as affirmed by the Licensing Board, is the capability of safety components in the containment and auxiliary buildings to survive an accident at least as severe as TMI-2 with 30-50% fuel failure (14 NRC at 1397, para. 1140) and hence, compliance of such equipment with GDC 4.

The Commission has also asked whether the qualification status of equipment can be "certified" by the Staff or is an issue to be litigated in a hearing. The certification procedure is offensive to the most basic principles of administrative due process. A decision must be made the "basis of the record after fair opportunity for exploration of the facts by the parties." Seacoast Anti-Pollution League v. Costle, 572 F.2d 872 (1st Cir. 1978). Employees of the agency "engaged in the performance of

investigative or prosecuting functions for an agency in a case may not ... participate or advise in the decision." 5 U.S.C. 554 (d). See Transworld Airlines v. Civil Aeronautics Board, 254 F.2d 90 (D.C. Cir. 1958); FTC v. Atlantic Richfield Co., 576 F.2d 96, 102 (D.C. Cir. 1977); King v. Caesar Rodney School District, 380 F. Supp. 1112, 1118 (D.C. Cir. 1974). The Staff and licensee had an opportunity to present facts on the record. Having failed to do so, the Staff, which was an adversary party opposed to UCS, cannot be permitted to "certify" a new record to the Commission -- a record which will not be subject to UCS' probing, will be made only by the Staff and licensee, and which would take the place of the hearing record.

The Appeal Board recognized also that matters which go beyond the implementation of a decision and involve the resolution of disputed questions must be determined by an adjudicatory body not the Staff. 17 NRC at 888.4/ Both the questions of 1) whether specific equipment is qualified and 2) if not, whether operation can nonetheless safely be permitted are the fundamental matters in dispute raised by the contention. Their resolution perforce cannot be delegated to the Staff and the mechanism for their resolution, assuming it is other than a finding that the record does not support restart, must provide UCS a fair opportunity to present evidence and question witnesses presented by the other side.

Moreover, if the Commission is interested, as it must be, in assuring itself of the safety of TMI-1 rather than in finding some way to avoid the issue, it should welcome an open airing of the facts. Right now, GPU is

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4/ This is a long-established precedent in AEC and NRC case law. See Consolidated Edison Co. (Indian Point Station, Unit 2) CLI-74-23, 7 AEC 947, 951-2 (1974); Public Service Co. of Indiana (Marble Hill Nuclear Generating Station, Units 1 and 2), ALAB-461, 7 NRC 313, 318 (1978); Cleveland Electric Illuminating Co. (Perry Nuclear Power Plant, Units 1 and 2), ALAB-293, 2 NRC 730, 736-7 (1975).

attempting to justify operation of the plant with unqualified equipment in the EFW system on the remarkable basis that the feed and bleed cooling mode can be used if EFW fails. H.D. Hukill to J.F. Stolz, February 22, 1984, Justification for Continued Operation, first unnumbered page. This is despite the fact that the Appeal Board ruled that feed and bleed has not been demonstrated to be a viable method of cooling the core. 17 NRC at 852. Another rationale offered by GPU is the undefined "low probability" of a high energy line break. GPU Nuclear Technical Response to Union of Concerned Scientists' Petition for Show Cause Concerning TMI-1 Emergency Feedwater System, February 24, 1984, p.4. This purported justification is perhaps even more revealing than the former, since the staff has made it clear for several years that when a piece of safety equipment is found to be qualified, justification for continued operation can not be based on asserting that the accident is of low probability.[5/] Environmental qualification requirements are in place precisely to protect against relatively low probability but high consequence accidents. That GPU would nonetheless put this forward as justification reveals a company that is either intransigent or still ignorant of its responsibilities in this area.

We have as yet seen no response by the Staff to these particular assertions; but we present them as illustrative of the need for close scrutiny of these issues. UCS has no confidence and cannot be legally required to rely upon the Staff to effectively provide this scrutiny.

5/ The criteria for determining whether a plant may safely operate with nonqualified equipment specifically do not include that the accident in question is of low probability. SECY 82-51, "Staff Requirement -- SECY 81-603B -- Proposed Rulemaking, Environmental Qualification of Electrical Equipment in Nuclear Power Plants," App. A, Feb. 4, 1982. See also 41 Fed. Reg. 2876, 2879, Cols. 1 and 2, Jan. 20, 1982.

2. Emergency Feedwater (EFW) Reliability

The Commission has asked whether the Appeal Board erred in its treatment of the Licensing Board' quantitative analysis of the EFW system and, if so, whether there is sufficient evidence in the record to support a finding that the EFW system is adequately reliable under a quantitative or other rationale.

The need for a highly reliable EFW system was one of the primary lessons learned from the TMI-2 accident. NUREG 0578, July, 1979, p.10. The Staff has since determined that the safety functions performed by this system are so vital that even meeting the single-failure criterion is not sufficient to ensure system adequacy. Cognizant AEOD staff wrote the following:

> The AFW system, in my opinion, is probably the most versatile and vital of the plant safety systems. It is typically used during normal plant operation, i.e., startup and shutdown, as well as in the mitigation of postulated events such as main steamline break, small break loss of coolant accident, loss of feedwater, steam generator tube rupture, and loss of offsite power.[6] So crucial is the availability of this system during a loss of offsite power that it is required by the staff to have at least two full-capacity independent systems powered by diverse sources and is the only safety system designed to function during a total loss of AC, loss of offsite power and failure of the redundant onsite emergency AC power. Further, it is the only safety system for which a reliability analysis must be performed demonstrating an unreliability in the range of 10^{-4} and 10^{-5} per demand. C. Michelson, Director, AEOD, to H. Denton, Director, NRR, Technical Review Report, "Postulated Loss of Auxiliary Feedwater System Resulting from Turbine Driven Auxiliary Feedwater Pump Steam Supply Line Rupture," Feb. 16, 1983, Attachment at 3, emphasis added.[7]

In design reviews since the issuance of the Standard Review Plan, the auxiliary feedwater system is treated as a safety system in a pressurized water reactor plant. It is required to satisfy the decay heat removal

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6/ We are aware that the TMI-1 EFW system is not used for normal startup and shutdown. However, that does not detract from the force of this statement.

7/ UCS offered this document into evidence in the Appeal Board's reopened hearings on decay heat removal. The objections of the staff and GPU were sustained, however.

requirements set forth in General Design Criterion 34 of Appendix A to 10 CFR Part 50. It also plays a significant role in the mitigation of feedwater transients, which are anticipated operational occurrences. NUREG-0578 at A-30.

General Design Criterion 20 of Appendix A to 10 CFR Part 50 requires, in part, that the protection system shall be designed to initiate automatically the operation of appropriate systems to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences.

Appendix A to 10 CFR Part 50 defines and explains anticipated operational occurrences as "those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit and include but are not limited to loss of power to all recirculation pumps, tripping of the turbine generator set, isolation of the main condenser, and loss of all offsite power.

In the event of an anticipated operational occurrence such as a main feedwater transient or a loss of all offsite power, the TMI-1 systems initially available for decay heat removal are the emergency feedwater system and the high pressure injection system. Keaten, et al., ff. Tr. 16,552 at 6-8. GPU argued that the PORV or pressurizer safety valves operating with HPI in the so-called "bleed and feed" cooling mode, could be relied upon in the event of EFW failure. The Appeal Board, however, agreed with UCS that bleed and feed has not been demonstrated to be a viable mode of core cooling. 17 NRC 814, 848-852.

After the reactor coolant system has been cooled and depressurized to about 250°F and 320 psig, the low pressure injection system (also called the decay heat removal system) can be used to continue the cooling process until the conditions of cold shutdown (reactor coolant system average temperature less than 200°F) are reached. Id. at 5, 9.

The Licensee has performed no evaluation to determine the probability of loss of main feedwater at TMI-1. However, the generic data for B&W plants over a two year period for five plants (i.e., 10 unit-years) shows that the frequency of loss of main feedwater was 0.3 per plant-year. The Licensee estimated that the uncertainty attached to this frequency is less than a factor of 10. Tr. 16,618-20, Keaten. This represents a high probability of loss of main feedwater and a consequently high rate of demand for emergency feedwater.

The Licensee also does not know either the probability of failure of the emergency feedwater system or the probability of failure of all decay heat removal system at TMI-1. Tr. 16,629, Keaten. It offered no failure analysis.

The Licensee, under cross-examination, agreed that use of the EFW system for decay heat removal relies upon the operation of other non-safety grade equipment such as the atmospheric dump valves, the turbine bypass valves, and/or the main condenser. Tr. 16,557-59, Keaten. There is no way to remove decay heat from the steam generators without the use of non-safety grade equipment. Id. This introduces an inherent unreliability into the system.

The reactor operator is relied upon to manually control steam generator level. Automatic control of steam generator level is provided by the non-safety grade integrated control system, but not at a sufficiently high level for adequate heat removal in the two-phase mode of natural circulation. Tr. 16,561-62, Ross.

In summary, the Licensee presented no convincing quantitative or qualitative evidence upon which one could conclude that the reliability of the methods for decay heat removal is sufficiently high to justify restart in light of the lessons learned from the TMI-2 accident.

The Staff testified that it had performed a reliability assessment of the TMI-1 EFW system and concluded that the EFW system with the modifications to be implemented by the time of restart would be sufficiently reliable to allow restart of TMI-1. Wermiel and Curry, ff. Tr. 16,718, at 1. There was substantial questioning about the basis for this conclusion.

The Staff presented reliability estimates of the TMI-1 emergency feedwater system design as it existed in mid-1979 and as it will exist after planned changes are completed. Id. at 31. In the latter case, the reliability estimate assumed that all of the long-term modifications had been completed. Tr. 16,733, Curry. The Staff's analysis used failure rate estimates from WASH-1400. Wermiel and Curry, ff. Tr. 16,718, at 33-34, Tr. 16,962, Curry.

The Staff analyzed three specific plant "transients" that result in the demand for EFW -- loss of main feedwater, loss of offsite power concident with loss of main feedwater, and loss of all AC power coincident with loss of main feedwater. Wermiel and Curry, ff. Tr. 16,718, at 32. To estimate the probability of EFW failure, the Staff defined failure as failure to provide 460 gpm flow to at least one steam generator within five minutes. Id. at 31.

Given a loss of main feedwater, the Staff estimated the probability of EFW failure to be $8X10^{-3}$ for the mid-1979 design and $4.5X10^{-4}$ for the design after all long-term modifications are completed. Id. at 35, 37, and Attachement 3. Given a loss of offsite power coincident with loss of main feedwater, the Staff estimated the probability of EFW failure to be approximately the same as that for loss of main feedwater. Id. at 35, 37. For a loss of all AC power, the Staff estimated the probability of EFW failure to be about $6X10^{-2}$ for the mid-1979 design and about the same for the design after all long-term modifications are completed. Id.

The Staff also estimated that, for the loss of main feedwater transient, the probability of EFW failure is about $3X10^{-3}$ for the design as it will exist at the proposed restart date. Tr. 16,738, Curry. The Staff did not present an estimate of probability of EFW failure for the restart design for the loss of offsite power and loss of all AC "transients." The Staff witness believed that his estimates were accurate within an uncertainty range of a factor of 10. Tr. 16,965, Curry.

These are, of course, relatively high failure rate estimates, particularly considering that the demand rate for the emergency feedwater system is also high. This is because EFW is required to remove decay heat for anticipated operational occurrences such as loss of main feedwater and loss of offsite power. Loss of main feedwater has historically occurred at B&W plants at the rate of 0.3 per plant year. Tr. 16,618-20, Keaten.

Thus, while it could conceivably be acceptable to tolerate lower reliability levels for safety equipment which is called upon to function only very rarely, the evidence shows that emergency feedwater is needed perhaps once a year or within that range. Given this demand rate, an EFW failure rate in the range which the staff presented is intolerable.

Moreover, the failure rate is in fact higher than indicated by the staff. The staff witness biased the results of his fault tree analysis by simply assuming that at least one of the diesel generators would function when called upon. That is, he assumed that one diesel generator was available and the probability of failure of the other was 10^{-2}. Tr. 16,971, Curry.

UCS requested the Board to take official notice of the diesel generator failure rate estimates used in WASH-1400 for failure of a diesel generator to start of $3X10^{-2}$. WASH-1400, App. III, Section 2, Table III 2-1. The request was denied on the ground that the figures presented in WASH-1400 are not universally accepted. This ruling was incorrect.

We noted above that the failure rates used by the staff in the TMI-1 EFW analysis were generally derived from WASH-1400. A review of the record also shows that the Licensee has relied heavily on WASH-1400 component failure rates in calculating accident probabilities. (Eg, Tr. 11,107, 11,130, 11,140, Levy; Levy, ff. Tr. 11,049, at 14, 15.

In addition, the Appeal Board decision in Florida Power and Light Co. (St. Lucie Nuclear Power Plant, Unit No. 2), ALAB-603, 12 NRC 30 (1980) is instructive. The issue in that proceeding centered around the likelihood of total loss of AC power. In that connection, one staff witness on diesel generator reliability used the WASH-1400 demand failure rate of $3X10^{-2}$. Id. at 47. The Appeal Board noted that this was an appropriate use of WASH-1400. Id. at n. 60, p. 47. Based upon this figure, they determined that the probability of failure of both diesel generators is in the range of 10^{-3} to 10^{-4}. Id. at 48.

Considering that the Licensee's objections to the Board's taking official notice of the failure rate of diesels was general in nature and presented no facts suggesting that these figures are inaccurate, and considering that the staff has itself used these figures in testimony very recently, we see no reason why they cannot be officially noticed.

If the diesel generator failure rate were factored into the staff's analysis, the effect would be to make the probability of failure of emergency feedwater even greater, although the exact magnitude of the change cannot be determined.

Judging from the Staff's failure probability estimates for the loss of main feedwater "transient," it can be concluded that relatively little of the improvement in EFW reliability attributable to hardware changes will be incorporated prior to restart. Tr. 16,742-43, 16,746, Curry.

The Staff attempted to downplay the significance of the relatively high probability of EFW failure in two ways. First, the Staff claimed that if more time for operator action were considered, i.e., if the definition of EFW failure was changed to allow more than five minutes to deliver flow to at least one steam generator, the estimated reliability of EFW would improve. This was one of the prime grounds cited by the Appeal Board for rejecting the quantitative analysis. 17 NRC 814,832-3. Second, the Staff claimed that the availability of the bleed and feed cooling mode could be recognized as a backup to EFW for decay heat removal. We now address these two factors to explain why such testimony cannot be given any weight.

The Staff acknowledged that it had done no analysis of TMI-1 EFW failure probability for a time interval longer than five minutes. Tr. 16,744, 16,746, Curry. The Staff opined nonetheless that if a longer period of time were analyzed (i.e., if the definition of EFW failure allowed more time to deliver EFW to the steam generators), operator action could introduce additional failure modes, but that it was more likely that operator action would correct failures. Tr. 749, Curry. However, that "opinion" was not based on review of the TMI-1 emergency procedures or operator qualification training, or on any expertise of the witness and is little more than sheer speculation. Tr. 16,758-9, Wermiel. There is, in fact no basis for concluding that accounting for operator action would significantly improve the chances of success on a quantitative basis, particularly considering the potential for operator misaction.

When the Board (Dr. Jordan) asked the Staff to explain why Westinghouse plants have an order of magnitude higher EFW system reliability than TMI-1 (Wermiel and Curry, ff. Tr. 16,618, at 35, 37), the Staff attributed this to the difference in the success criterion. That is, since Westinghouse steam

generators dry out in the absence of EFW more slowly than TMI-1, much more credit can be given for operator recovery action. Tr. 17,075-76, Curry.

However, B&W did analyze EFW reliability for 5, 15, and 30-minute intervals, and in no case were the reliability estimates as high as the best Westinghouse reliabilities. Tr. 17,076, Curry. Therefore, this record indicates that, even if the success criteria had been loosened, no great improvement in EFW reliability would be demonstrated.

The Staff made no attempt whatever to analyze the effect of assuming longer time periods. Tr. 17,076-77, Curry. The Appeal Board erred in accepting unsupported speculation as a basis for deciding that lengthening the time for success would significantly alter the results of the quantitative analysis, particularly when B&W analyses indicate otherwise.

The Appeal Board's only other basis for rejecting the quantitative failure analysis was its observation that, since EFW systems in B&W plants can vary widely in design, they could not be "sure" that the EFW challenge rate is applicable to TMI-1. 17 NRC 814, 832. This objection seems fundamentally hypocritical to UCS. The best available data is that from B&W plants; one can never be "sure" that it will precisely match TMI-1 performance. Indeed, since all such historical data represent averages, it would be miraculous if the TMI-1 EFW challenge rate were precisely the average. But there was no evidence introduced at all to indicate that TMI-1 would be significantly different within the uncertainty inherent in a quantitative analysis. If one were to accept the Appeal Board's reasoning here, no quantitative analysis would ever be accepted.

In addition, licensee's objection, adopted by the Appeal Board, focused on whether or not the EFW system would be challenged at the same rate that it had been challenged in some pertinent B&W plants. However, the reliability

criterion concerns availability on demand. The assumption is properly made that the EFW system will be demanded because loss of main feedwater (a non-safety grade system) is an anticipated operational occurrence. That is, it is expected to occur during plant lifetime.

Moreover, as UCS has shown above, there is good reason to believe that the failure rate presented by the staff is generous to TMI-1. For one thing, the analyses simply assumed that at least one diesel generator would always be available and that the other would fail at a rate of 10^{-2}. Tr. 16,971, Curry. In addition, the quantitative analysis assumed, with a few non-germane exceptions, that the EFW system was seismically qualified. We now know that this is very far from the truth. App. Tr. 345 ff., Wermiel.[8/] See also UCS Comments on The Commission's Ex Parte Meeting of December 17, 1982....", January 7, 1983, pp. 21-23.

The Appeal Board's rejection of the quantitative EFW failure rates accepted by the Licensing Board was unjustified.

However, even if the quantitative analysis is disregarded, that does not help GPU or provide any rational basis for concluding that the EFW system is sufficiently reliable. One must ask, if quantitative analysis cannot be used, where can the Commission look for a standard by which to judge reliability and a measure by which to determine that the standard is met? The only other standard, indeed the standard normally applied, lies in the Commission's deterministic design criteria for systems important to safety; that is, the General Design Criteria of 10 CFR Part 50, App. A, represent the agency's longstanding judgment that the design of systems important to safety must conform to specified criteria ensuring redundancy, diversity, testability,

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8/ "App. Tr." denotes the transcript of the Appeal Board reopened hearings on decay heat removal, held March 7 and 8, and March 16 and 17, 1983.

seismic and environmental qualification, etc., in order to ensure sufficient reliability. Judged by this standard, the TMI-1 EFW system patently fails.

UCS has detailed the numerous ways in which the TMI-1 EFW system fails to meet the requirements for a safety system in many filings. Union of Concerned Scientists' Petition for Show Cause Concerning TMI-1 Emergency Feedwater System, January 20, 1984; Union of Concerned Scientists' Comments on the Commission's Ex Parte Meeting of December 17, 1982...", January 7, 1983, pp. 7-28; Union of Concerned Scientists' Findings of Fact and Conclusions of Law on...Board Question 6, June 12, 1981, pp. 166-194; UCS' Brief on Exceptions to the Partial Initial Decision of December 14, 1981-Part 2, April 14, 1982, pp. 103-110. The system is not remotely safety grade. The deficiencies cut a broad swath across the spectrum of safety requirements. It does not meet the single failure criterion. It is neither seismically nor environmentally qualified, the main steam line rupture detection system is grossly inadequate. See Union of Concerned Scientists' Petition for Show Cause Concerning TMI-1 Emergency Feedwater System, January 20, 1984. Indeed, GPU itself has filed a 13-page description of the EFW system modifications necessary "to provide increased reliability in its capability to mitigate the effects of design basis accidents when the main feedwater system is not available." H. D. Hukill, Director, TMI-1, to J. F. Stolz, August 23, 1983. None of these modifications will be done prior to restart, if GPU's position is accepted.

In the face of the plain fact that the TMI-1 EFW system meets neither any quantitative or qualitative reliability criterion, the Commission is not free to fall back on warm feelings or the hopeful speculation of witnesses who offered no solid basis for their self-serving assurances.

There is, it should be understood, another dimension to the question of emergency feedwater reliability. It is not only important that the system

deliver sufficient water to the steam generators, but at least equally important that the delivery of water results in establishing a mechanism for adequate core cooling. It is known that for small breaks within a certain spectrum, liquid natural circulation will be stopped by the formation of steam voids at the high points of the primary system. 17 NRC 814, 837. The high point vents, even if they were operable, are too small to assist in restoring natural circulation. Id. at 837-8. "Therefore, other means must be available to provide adequate core cooling during a small break LOCA." Id. at 838.

As noted above, bleed and feed, the primary alternative espoused by GPU, was not demonstrated to be viable and was not accepted by the Appeal Board. Id. at 852. The only other alternative was the so-called "boiler-condenser" mode of natural circulation, discussed at Id., pp. 840-848. The testimony on the subject of the boiler-condenser mode can best be described as unsettling. The Staff hired EG&G to do a computer analysis of a .01 ft^2 break. Sheron and Jensen, ff. App. Tr. 83, at 9. Each of the two staff witnesses offered a different explanation of the phenomena described by the analysis which featured a bizarre "chugging" behavior that is physically impossible. See 17 NRC 814, n. 118, at 845; App. Tr. 597-611, Jensen; App. Tr. 707-723, Sheron. 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, 705-6, Sheron. We agree. In this connection, we strongly urge the Commission to read UCS' detailed analysis of the record on this issue. Union of Concerned Scientists' Proposed Findings of Fact and Conclusions of Law on Reopened Hearing, April 12, 1983.

Staff members in the NRC's Office of Analysis and Evaluation of Operational Data (AEOD) who have independently reviewed the question of boiler-condenser viability concluded and still believe that the proposition

that natural circulation would be established in the boiler-condenser mode is not a certainty, particularly in the absence of experimental data for B&W plants. App. Tr. 746-750, Ornstein; UCS Ex. 53. This represents the official AEOD viewpoint. App. Tr. 752, Ornstein.

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. Furthermore, 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.

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 the AEOD witness as an "outlier." App. Tr. 788, Ornstein. Such an outlier cannot be used to confirm the result of other codes which predicted very different plant behavior, precisely the case here.

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.[9/] These are very

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9/ 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.

important questions which have not been answered. Basically, a far better understanding of the physical phenomena in question is required. See App. Tr. 759, Ornstein.

In general, AEOD's testimony confirmed the need for experimental testing before the Commission could conclude that boiler-condenser is a sufficiently reliable means of decay heat removal. The EG&G analysis did not confirm the accuracy of the B&W codes. The fact that both RELAP5 and B&W predict that the core remains covered is far from conclusive, given the vast difference in the codes, the assumptions used, the plant behavior predicted, and the great sensitivity of the analyses to the input assumptions. 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.

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

In the face of this fundamental lack of understanding of how the plant would physically respond to a SBLOCA after interruption of natural circulation, the Appeal Board fell back on abstract heat transfer equations, purportedly demonstrating the viability of boiler-condenser. The Appeal Board refused to allow UCS to pusue lines of inquiry which would have shown that it is necessary for the operators to know what is actually happening inside the plant in order for them to take the action necessary to maintain adequate core cooling. See App. Tr. 566-584, 585-588. Ironically, the TMI-2 lessons concerning the need to pay heed to the demands on operators (i.e. to minimize

sources of confusion) and to provide them with the necessary instrumentation and training to ensure proper action have been lost on this issue.

In conclusion,, this record does not support a conclusion that the TMI-1 emergency feedwater system is sufficiently reliable on either a quantitative or qualitative basis, considering the vital safety functions which it performs.

3. PORV

The Commission has asked whether the Appeal Board erred in holding that the arguments concerning use of the PORV during low temperature operation and inadequate core cooling conditions were outside the scope of the proceeding, and, if so, whether these "alleged uses" of the PORV require that it be safety grade. The short answer to both questions is "yes". UCS explicitly cited both of these vital PORV safety functions as grounds for requiring the PORV to be safety grade. See Pollard, ff.Tr. 9027 at 5-4 to 5-5, items 3 and 6. See also UCS Proposed Findings of Fact and Conclusions of Law on UCS Contentions 1,2,3,4,5 and 10, paras. 198-234, June 1, 1981. The Appeal Board's ruling that these questions are outside the scope of the proceeding is preposterous. Particularly egregious is the holding that inadequate core cooling events, precisely what occurred at TMI-2, are beyond the scope of the restart proceeding. The issue of jurisdiction has been previously briefed in the Union of Concerned Scientists' Response to Commission order of August 5, 1983, Aug. 30, 1983. A copy of that filing is attached for the Commissioners and incorporated herein.

At low temperatures, the steel of the reactor vessel is susceptible to cracking (i.e. brittle fracture). Until the reactor vessel walls are above the nil ductility transition temperature, the reactor coolant system pressure must be limited to a few hundred pounds per square inch. Since reactor

pressure vessel rupture is an accident beyond the capability of ECCS to mitigate, it is extremely important to maintain the integrity of the vessel. (Pollard, ff. Tr. 9027 at 5-10 to 5-11,)

The PORV is used during low temperature operations to protect against overpressurizing the reactor vessel. This function, the third safety-related function identified by UCS, cannot be performed by the safety valves because their opening pressure set point - 2500 psig - is far above the permissible pressure limit and cannot be changed by the operator. (Id).

UCS's position is supported by NUREG-0578 which states that "[t]he PORV is also used to prevent over-pressurization of the reactor coolant system during operation at low temperatures, an operational mode when the nil ductility transition temperature (NDTT) becomes a consideration for structural integrity of the primary coolant pressure boundary. *** The NDTT protection mode can also be selected, in which case the PORV opens in the event a preselected low-pressure setpoint is reached or [sic] reactor temperatures are below the NDTT limit." (NUREG-0578, at A-3)

The Licensee confirmed that this description is applicable to TMI-1. (Tr. 8755-8756, Jones) The Staff and Licensee agreed that the PORV is used to prevent reactor coolant system overpressure during low temperature operation, but argued that this function of the PORV is only a backup to reactor operator action. (Jensen, ff. Tr. 8821, at 3; Tr. 8755-8756, Jones).

This was the position adopted by the Appeal Board. However, it is incorrect to refer to this function of the PORV as a backup to the operator because under some plant conditions, the only way to limit overpressure is by use of the PORV. (Tr. 9031-9033, Pollard)

During cross-examination by UCS, the Licensee agreed that, if the plant is in cold shutdown condition with the reactor coolant system solid, the PROV "may" serve a safety function in relieving the overpressure. (Tr. 8979, Jones)

Nevertheless, the Licensee still attempted to maintain that the operator has the capability to terminate an overpressure event and the PORV is just a backup. (Id)

This assertion is without merit. Operator action can be relied on only if adequate time is available. In the case of the primary system in a solid condition, i.e., without a bubble in the pressurizer, the operator does not have time to act. (Tr. 8976, Jones) Furthermore, a TMI technical specification requires that the PORV shall not be taken out of service nor shall it be isolated from the reactor coolant system unless the high pressure injection pumps are disabled, the reactor vessel head is removed, or the average primary coolant temperature is above 320^{o}F. (Tr. 9015, Jones).

This specification essentially defines plant conditions where either overpressurization has a low probability of occurrence or the primary system temerature is above the nil ductility transition temperature. In either case, the plant

conditions are such that the low temperature overpressure protection provided by the PORV is not needed. One can reasonably infer that under all other conditions of low temperature operation, the PORV is needed for safety, otherwise there would be no prohibition against taking it out of service.

UCS identified the use of the PORV to depressurize the reactor coolant system during conditions of inadequate core cooling as another safety function which requires that the PORV be upgraded to safety grade. (Pollard, ff. Tr. 9027, at 5-16 to 5-17)

The TMI-1 emergency procedures instruct the operator to open the PORV and leave it open in the event of inadequate core cooling. This action is intended to depressurize the reactor coolant system to allow operation of the low pressure injection system and thereby restore core cooling. (Pollard, ff. Tr. 9027, at 5-16 to 5-17; Lic. Ex. 48, at 28.0)

This depressurization function cannot be performed by the safety valves because they will not open below 2500 psig and they are not controllable by the operator. (Pollard, ff. Tr. 9027, at 5-17)

Use of the letdown line to depressurize the system might be precluded becasue of the high level of radioactivity in the reactor coolant system after core damage. (Pollard, ff. Tr. 9027, at 5-17; UCS Ex. 4, at 6.0)[10/]

10/ There are two Emergency Procedures for TMI-1 covering inadequate core cooling. EP 1202-6B, Attachment 3 is, in different versions, UCS Ex. 6 and Licensee Ex. 48. EP 1202-39 is, in different versions, UCS Ex. 4 and Licensee Ex. 51.

After litigation of this contention was completed, the Licensee amended the TMI-1 emergency procedure referenced by UCS, UCS. Ex. 4. The revised procedure still directs the operator to open the PORV to depressurize the reactor coolant system if feedwater is not available and, if main or emergency feedwater is available, to use letdown flow to help control reactor coolant pressure, but the statement that letdown flow may be prohibited by high activity was deleted from the procedure. (Lic. Ex. 51, at 4.0, 5.0) This change to the emergency procedure does not, of course, change the fact that use of letdown flow to control reactor pressure may be prohibited because of high activity.

The Staff also testified that one function of the PORV is to give the operator a means of depressurizing the primary system that is independent of the steam generators. (Jensen, ff. Tr. 8821, at 3)

The Licensee testified that the PORV is only an additional means of depressurizing the primary system which has a smaller impact than use of the steam generators. (Tr. 8761-8762, Jones) The Licensee also testified that it was acceptable to rely on non-safety grade equipment in this instance because a situation involving inadequate core cooling is not part of the design basis for TMI-1. (Tr. 8762-8763, Jones) These positions were adopted by the Appeal Board. 17 NRC 814,864-5.

The first argument implies that depressurization using the steam generators is an independent method of depressurization which does not call for use of the PORV. This implication is contradicted by the TMI-1 emergency procedures.

The procedures call for the operator to depressurize the steam generator(s) as rapidly as possible to 400 psig or as far as necessary to achieve a 100° F decrease in secondary saturation temperature. At the same time, the operator is directed to use the PORV, as necessary, to maintain RCS pressure within 50 psi of steam generator pressure. (Lic. Ex. 48, at 26.0 - 27.0)

Thus, even if the primary system is being depressurized via the steam generators, the PORV is still used to keep primary system pressure within 50 psi of steam generator pressure. Thus, the PORV is needed in conjunction with use of the steam generators.

Furthermore, in another section of the emergency procedures for inadequate core cooling, the operator is directed to both depressurize the steam generator(s) and open the PORV and, following depressurization, to control reactor coolant system pressure below 150 psig using the PORV. (Lic. Ex. 48, at 28.0)

The Appeal Board's second argument is apparently that use of non-safety grade equipment is acceptable because a situation involving inadequate core cooling is beyond the design basis for TMI-1. See 17 NRC 814,865 and n.235. This argument ignores the lessons learned from the TMI-2 accident and the other requirements that have been imposed on TMI-1 for accidents previously considered to be beyond the design basis.

229. The Lessons Learned Task Force described the TMI-2 accident and the general issues it raised as follows:

"At Three Mile Island, some of the safety systems were challenged to a greater extent or in a differenct manner than was anticiapted in their design basis. Many of the events that occurred were known to be possible, but were not previously judged to be sufficiently probable to require consideration the design basis. Operator error, extensive core damage, and production of a large quantity of hydrogen from the reaction of zircalloy cladding and steam were foreseen as possible events, but were excluded from the design basis, since plant safety features are provided to prevent such occurrences. The Task Force will consider whether revisions or additions to the General Design Criteria or other requirements are necessary in light of these occurences. A central issue that will be considered is whether to modify or extend the current design basis events or to depart from the concept. For example, analysis of design basis accidents could be modified to include multiple equipment failures and more explicit consideration of operator actions or inaction, rather than employing the conventional single-failure criterion. Alternatively, analyses of design basis accident could be extended to include core uncovery or core melting scenarios. Risk assessment and explicit consideration of accident probabilities and consequences might also be used instead of the deterministic use of analysis of design basis accidents."
(NUREG-0578, at 16-17) emphasis added

TMI-1 is being required to install high point vents in the reactor coolant system (Staff Ex. 1, at C8-60) and instrumentation to detect inadequate core cooling. (Id., at C8-14 to C8-21) The venting system (but not each vent path) is required to be safety grade. (Id., at C8-17) The Licensee is also being required to upgrade plant radiation shielding to provide adequate personnel and equipment protection after an accident in which significant core damage occurs. (Id., at C8-33). These measures clearly assume the occurence of an

accident beyond the design basis for TMI-1 when it was licensed and yet the new equipment being installed is required to be safety grade. This is a recognition of the serious nature of the safety functions involved and the grave consequences of their failure.

The Appeal Board's resolution of the PORV issue enshrines the proposition that, no matter how important the safety function involved -- there is no serious dispute that these depressuration functions are extremely important -- the system which is designed for and intended to perform this function need not be safety grade if there is any conceivable alternative means that might be available to perform the function,[11/] even if the asserted alternative is at variance with the operators' traning and procedures, adds to the operators burden, thus introducing the possibility for human error, and involves unassessed risks. Such a position represents an extremely dangerous precedent and is at odds with NRC rules intended to ensure that systems "important to safety" are designed and fabricated to ensure reliability; i.e.; they should be "safety-grade." See 10 CFR Part 50, App. A.

4. Systems Interaction

The Commission has decided to review whether allowing the staff to address the need for a systems inteaction study for TMI-1 in a long term generic program is adequate or whether

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11/ As noted above, we emphasize that the evidence establishes that there are not alternative to the PORV in some cases.

such a study should be specifically required for TMI-1. We will not dwell long on this subject.

At first, the Licensing Board ordered that a TMI-1 systems interaction study be done as part of the generic program. 14 NRC at 1353. There are currently no plans to do such a study either before or after restart, despite the fact that the ACRS letter on TMI-1 restart called for "timely" completion of such analyses for TMI-1. UCS Brief on Exceptions. . ., p. 101[12/] The consensus analyses of the TMI-2 accident clearly show that systems presently classified as not important to safety and hence receiving little or no NRC review, can cause accidents and be used to mitigate accidents in ways not considered in the plant's safety analysis. The present NRC classification system does not adequately recognize or take account of either of these kinds of effects that non-safety systems can have on plant safety. NUREG-0578 at 18.

While the Staff has recognized the need to consider upgrading non-safety systems to reduce challenges to safety systems and to improve the capability of non-safety systems to operate during accidents and transients (Id.), the Staff has no program whatever to take the first required step in this process for TMI-1 the undertaking of a plant-specific comprehensive study to identify potential adverse systems interactions at TMI-1.

12/ A PRA is not the same and does not substitute for a systems interaction analyses.

We are at a loss to understand how this could rationally be characterized as "reasonable progress." 17 NRC 814, 884. If anything, it is regression rather than progress. There is now no TMI-1 systems interaction study in sight at all. Surely, mere acknowlegment of the existence of an unaddressed safety problem is not reasonable progress when five years later we are further from its resolution than in 1979.

The Appeal Board found some comfort in the fact that some effort has been made by the staff (e.g. partial "upgrading" of the PORV), Id. at 883. Of course, no-one contests that some changes have been made; their effect on risk reduction and their sufficiency is the matter at issue. On this, the record is grossly insufficient to support a finding that systems interaction has been adequately addressed.

The Staff testified that, in order to justify upgrading a system or component to safety-grade, it would have to be shown that failure of that non-safety system by itself would cause core damage or that use of a non-safety system was required to mitigate an accident even if safety systems operated properly. (Conran, ff. Tr. 8372 at 8-10). The witness believed that non-safety equipment was used only after improper operation of safety system resulted in core damage, therefore, he does not believe upgrading of these (or other) non-safety systems is required. (Id. at 11)

However, on cross-examination the witness could not support this statement. He did not know, for example, whether

pressurizer heaters or the reactor coolant pumps were used before core damage occurred. (Tr. 8603, Conran) In fact, the reactor coolant pumps were used for 1 hour and 40 minutes at the very out set of the accident before core damage occurred. (See proposed findings of fact . . ., ¶ 15) The witness was incorrect and had no basis whatever for claiming that non-safety systems were used only after improper operation of safety systems resulted in core damage. (Tr. 8603-8604, Conran)

Moreover, to the extent that the testimony or the Appeal Board decision implied that "careful analysis" was done by the Staff to determine whether any non-safety grade equipment shold be upgraded, it is inaccurate. Mr. Conran himself never did such an analysis. (Tr. 8547, Conran) He thought that "someone like Mr. Jensen might be involved in that sort of thing." (Id.) When specifically asked what analysis was done by anyone on the Staff of the TMI systems to enable the Staff to determine whether any TMI-1 non-safety systems meet his criteria for upgrading, the only thing he could point to was the B&W computer analyses of transients and accidents discussed in Mr. Jensen's testimony on UCS Contentions 1 and 2. (Tr. 8551-8554, Conran) There is nothing in the description of that work that suggests that it is directed toward identifying adverse systems interactions or addresses itself to the criteria for upgrading put forth by Mr. Conran. (Tr. 8555-8566. Conran, See also Tr. 8103-8107, Pollard)

Based on the foregoing, even if Mr. Conran's criteria for upgrading systems to safety grade are the correct criteria, there is no evidence that they have been applied properly to TMI-1.

Finally, with respect to the issue of whether non-safety grade equipment should be partially upgraded as an exercise of "prudence", the witness was questioned on what basis the Staff used to determine what aspects of the system or equipment should be modified - in other words, which GDC should be applied and which disregarded in the partial upgrade? He stated that a "judgment had to be struck as to whether the additional realiability that might be gained by that was necessary." (Tr. 8613, Conran)

However, there is no indication on this record that <u>anyone</u> on the Staff ever did the review necessary to rationally exercise that "judgment" or even determined what would be needed to make the particular equipment fully safety grade, what would be gained in reliability and what the cost would be. (Tr. 8614, 8619-10, Conran) Mr. Conran knew of no such analysis. He testified that this is because of the "circumstance under which these kinds of judgments were made," that they were "hot coal items". (Tr. 8614, Conran) Apparently the decisions had to be made very quickly on what to include in NUREG-0578, allowing little time for analysis (Id.)

However, even after NUREG-0578 was completed, when there clearly was time for more thought, no such analyses have been

done. (Tr. 8614, 8619-20, Conran) It is apparent that the Staff does not know whether the additional reliability that might be gained by making the PORV or other equipment safety grade is "necessary," or desirable. The staff does not know whether its partial upgrades have significantly improved plant safety. There is no reliable evidence on this record to indicate that they have. Although it claims to have exercised judgment, the Staff is not in possession of the basic facts necessary in order to exercise judgment. Hot coal or not, a perceived need to make decisions quickly does not justify the inability to support those decisions.

It is clearly erroneous, therefore, for the Appeal Board to suggest that the "partial upgrades" of a few pieces of equipment have made a significant dent in the problem of adverse systems interaction. The record does not contain evidence to support that conclusion.

5. Main Steam Line Rupture Detection System (MSLRDS)

The Commission's final questions is whether the Licensing Board erred in deligating to NRC staff the responsibility for approving licensee's solution to the main steam line rupture detection system problem. UCS has previously submitted its views to the Commission on the "adequacy of the proposed solution." In brief, they are that there has been submitted nothing approaching a proposed solution and that the vague principles described by GPU do not contain the information

needed to determine whether the system will adequately address the identified problems. See Union of Concerned Scientists' Comments on the TMI-1 Main Steam Line Rupture Detection System, Feb. 16, 1984.

With respect to the delegation issue, the question is very similar to that posed in connection with the proposal to "certify" the resolution. The Appeal Board recognized that the development of a solution to the steam generator bypass logic problem "may go beyond the implementation of the Board's decision and involve the resolution of disputed matters. "17 NRC 814, 888. Such a determination "must be made by an adjudication body, not the Staff." Id. We see no room for eqivocatory. Having identified the safety problem, the Board may not leave it to the Staff to negotiate a solution with GPU.

NRC long ago decided that the staff may not be delegated the decision-making authority of the adjudicatory boards. Consolidated Edision Co. (Indian Point Station, Unit 2) CLI-74-23, 7 AEC 947, 951-2 (1974); Public Service Co. of Indiana (Marble Hill Nuclear Generating Station, Units 1 and 2), ALAB-461, 7 N C 313, 318 (1978); Cleveland Electric Illuminating Co. ry Nuclear Power Plant, Units 1 and 2), ALAB-293, 2 NRC 736-7 (1975). The NRC Staff is an adversary party to case and its role disqualifies it from acting as judge. Employees of the agency "engaged in the

performance of investigative or prosecuting functions for an agency in a case may not . . . participate or advise in the decision." 5 U.S.C. 554 (d). See Transworld Airlines v. Civil Aeronautics Board, 254 F.2d 90 (D.C. Cir. 1958); FTC v. Atlantic Richfield Co., 576 F.2d 96, 102 (D.C. Cir. 1977); King v. Caesar Rodney School District, 380 F. Supp. 1112, 1118, (D.C. Cir. 1974).

Moreover, the parties are entitled under law to a fair opportunity to explore the facts which form the basis of a decision. Seacoast Anti-Pollution League v. Costle, 572 F.2d 872 (1st Cir. 1978). This principle precludes delegation of the MSLRDS resolution to the staff outside of the hearing context.

CONCLUSION

For the foregoing reasons, insofar as the Appeal Board concluded that the systems in question are sufficiently reliable to permit restart, its decision shold be reversed.

The Appeal Board did not make the overall finding that TMI-1 can be operated without endangering the health and safety of the public. On the contrary, it expressly refused to make this finding because of issues which it believed to be outside the scope of its review. 17 NRC at 823, 895. These included, in particular, the lack of seismic or environmental qualification of the EFW system and the lack of environmental qualification and the depresurization functions of the PORV.

Id. at 895. The Commission should find that the record does not support a conclusion that TMI-1 can be operated without undue risk to the public.

Respectfully submitted,

Ellyn R. Weiss
General Counsel
Union of Concerned Scientists

March 19, 1984

DEC 27 1982

U.S. NUCLEAR REGULATORY COMM

1983 JAN 7 AM 11 14

Docket No. 50-289

MEMORANDUM FOR: Chairman Palladino
Commissioner Gilinsky
Commissioner Asselstine
Commissioner Ahearne
Commissioner Roberts

FROM: Darrell G. Eisenhut, Director
Division of Licensing

SUBJECT: BOARD NOTIFICATION 82-133 - TMI-1
ENVIRONMENTAL QUALIFICATION OF SAFETY-RELATED
ELECTRICAL EQUIPMENT

In accordance with the procedures for Board Notification, the attached letter (Stolz to Hukill dtd December 10, 1982), is provided directly to the Commission. The letter forwards the staff's Safety Evaluation Report and Technical Evaluation Report (TER) for environmental qualification of safety-related electrical equipment at TMI-1. This issue relates to Board Question/UCS Contention 12. The information in the letter and enclosures does not change previous staff positions primarily because it is based on a worst-case LOCA while the staff's hearing testimony pertained to a small break LOCAs. The TER, because of its size, is being distributed only to those parties interested in the environmental qualification issue. The TER will be provided to other parties upon request.

Original signed by
Darrell G. Eisenhut

Darrell G. Eisenhut, Director
Division of Licensing

Enclosures:
SER
TER

cc:
Atomic Safety Licensing Appeal Board
for TMI-1
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DATE	12/13/82	12/13/82	12/23/82	12/23/82	12/17/82	12/23/82

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TECHNICAL EVALUATION REPORT

REVIEW OF LICENSEES' RESOLUTION OF OUTSTANDING ISSUES FROM NRC EQUIPMENT ENVIRONMENTAL QUALIFICATION SAFETY EVALUATION REPORTS (F-11 and B-60)

METROPOLITAN EDISON COMPANY
THREE MILE ISLAND NUCLEAR STATION UNIT 1 VOL. 1 OF 2

NRC DOCKET NO. 50-289
NRC TAC NO. 42513
NRC CONTRACT NO. NRC-03-79-118

FRC PROJECT C5257
FRC ASSIGNMENT 13
FRC TASK 492

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Department Director

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TABLE 4-1

NUMBER OF EQUIPMENT ITEMS IN EACH QUALIFICATION CATEGORY

NRC CATEGORY	CATEGORY DESCRIPTION	NUMBER OF EQUIPMENT ITEMS
I.A	EQUIPMENT QUALIFIED [EQUIPMENT ITEM NO(S).: 110,112]	2
I.B	EQUIPMENT QUALIFICATION PENDING MODIFICATION [EQUIPMENT ITEM NO(S).: 116]	1
II.A	EQUIPMENT QUALIFICATION NOT ESTABLISHED [EQUIPMENT ITEM NO(S).: 1, 2, 3, 6, 7, 10, 11, 14, 15, 26, 27, 28, 29, 32, 45, 46, 49, 50, 51, 53, 57, 60, 66, 67, 71, 78, 79, 81, 93, 98, 106,107,108,109,111,114,115,118,119,120]	40
II.B	EQUIPMENT NOT QUALIFIED	0
II.C	EQUIPMENT SATISFIES ALL REQUIREMENTS EXCEPT QUALIFIED LIFE OR REPLACEMENT SCHEDULE JUSTIFIED [EQUIPMENT ITEM NO(S).: 5, 17, 18, 19, 20, 21, 22, 24, 33, 36, 39, 40, 56, 58, 59, 63, 64, 69, 72]	19
III.A	EQUIPMENT EXEMPT FROM QUALIFICATION [EQUIPMENT ITEM NO(S).: 4, 31, 52, 54, 55, 74, 75, 80, 82, 83, 84, 86, 89, 92, 94, 95, 96, 97, 99,100,101,102,103,104,105]	25
III.B	EQUIPMENT NOT IN THE SCOPE OF THE REVIEW [EQUIPMENT ITEM NO(S).: 8, 9, 12, 13, 16, 23, 25, 30, 34, 35, 37, 38, 41, 43, 48, 61, 62, 65, 68, 70, 73,117]	22
IV	DOCUMENTATION NOT MADE AVAILABLE [EQUIPMENT ITEM NO(S).: 42, 44, 47, 76, 77, 85, 87, 88, 90, 91,113]	11
	TOTAL	120

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UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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OFFICE OF SECRETARY
DOCKETING & SERVICE
BRANCH

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

CERTIFICATE OF SERVICE

I hereby certify that copies of "UNION OF CONCERNED SCIENTISTS' BRIEF ON THE COMMISSION'S REVIEW OF ALAB-729" have been served on the following persons by deposit in the United States mail, first class postage prepaid, this 19th day of March 1984, except as indicated by an asterisk.

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