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' BRIEF ON EXCEPTIONS TO THE PARTIAL INITIAL DECISION OF DECEMBER 14, 1981 - PART 2

April 14, 1982

DS03
s
1/1

F. UCS Contention 10
Exceptions 46-56
PID 723-746
UCS Proposed Findings 241-313

Summary

UCS Contention 10 is as follows:

> The design of the safety systems at TMI is such that the operator can prevent the completion of a safety function which is initiated automatically; to wit: the operator can (and did) shut off the emergency core cooling system prematurely. This violates Section 4.16 of IEEE 279 as incorporated in 10 CFR 50.55a(h) which states:
>
> > The protection system shall be so designed that, once initiated, a protection system action shall go to completion.
>
> The design must be modified so that no operator action can prevent the completion of a safety function once initiated.

The Contention was limited by the Board to core cooling and containment isolation systems and further narrowed by UCS in two ways. First, it was limited to automatically initiated safety functions, i.e., manually initiated safety functions were not addressed. Second, the phrase, "no operator action," was limited to operator actions involving the equipment normally used by the operator to terminate the safety function. (Tr. 6544, Pollard)

As the Board found, premature operator intervention in the operation of the high pressure injection system was the cause of core damage at TMI-2. (PID 745) UCS's testimony was generally to the effect that the accident graphically demonstrated the unacceptable consequences of permitting the operator to interfere with the functioning of safety systems and that a clear lesson of the accident is that the plant design should be modified so as to preclude such intervention until the conditions specified in the plant's design basis for termination of safety systems has been achieved. (Pollard, ff. Tr. 641, at 10-1 to 10-4, 10-16 to 10-19) This could be accomplished by relatively minor modifications to the plant circuitry. (Tr. 6431-6432, Pollard)

This result can be reached either: 1) by interpreting the language of GDC 20 of 10 CFR Part 50, Appendix A, and IEEE Std. 279-1968 in the light of the TMI-2 accident and later standards development work in connection with IEEE Std. 603 (UCS PF 246-252, 286-310); or 2) by concluding, also in the light of the TMI-2 accident and after weighing the safety advantages of such a system against the alleged disadvantages, that a significant increase in safety would result from adopting the approach advocated by UCS. (UCS PF 253-285)

While the Board spent a good deal of time on the question of the interpretation of IEEE Std. 279, an issue which it resolved against UCS (PID 723-739), it virtually ignored all of the UCS evidence, cross-examination and proposed findings on the latter question, that is, whether a substantial improvement in safety would result from preventing premature operator termination of safety systems. Instead, it relied totally on Licensee's assertions that such a design would be impractical, complicated and detract from safety (PID 741-744) without devoting so much as a word to addressing the questions raised during the extensive cross-examination of the Licensee's witness or to the points made in UCS's direct rebuttal of his assertions. Indeed, a reader of this decision would have no hint of the vigorous dispute centering on this question. In fact, an interlock system to prevent premature operator intervention would not be overly complex (Tr. 6431-6432, Pollard) and would in no way restrict the operator's scope of action after completion of the safety function. (UCS PF 274) Moreover, under questioning not one witness in this proceeding was able to postulate even one concrete example of a situation where any conceivable hazard to the public could result from the design suggested by UCS. (UCS PF 272-275) Failing this, Licensee and Staff fell back on the notion that, for unforeseen accidents, the proposed design might preclude appropriate operator action. (UCS PF 275; Tr. 6299-6300, Clark; Tr. 6646, Sullivan) This assertion, also, was shown to have little merit. It requires

positing a situation where first, an unforseen event beyond the design basis occurs (an event now again claimed by NRC to be "incredible"); second, permitting the safety systems to operate until the safety function is completed (i.e. following post-TMI plant emergency procedures) would be the wrong response; and third, that the operator in the midst of this (by definition) "unforeseen" event, will be able to divine and take the appropriate action of terminating the safety system. The product of the probabilities of each of these necessary elements is so small that the "advantages" of a design allowing premature operator interference with safety system operation are far outweighed by the disadvantage of allowing improper operator action during a design basis accident. (Tr. 6423-6, 6563-4; Pollard; UCS PF 276-285)

Exceptions:

46. The Board erred in finding that "[t]here is no basis to apply the IEEE Std 279 to the completion of a subsequent safety function." (PID 729)

47. The Board erred in disregarding and failing to confront evidence showing that the purpose of IEEE Std. 279 is nullified if one accepts the interpretation of the Staff and Licensee.

48. The Board erred in disregarding evidence that the requirements or principles of IEEE Std. 279 have been applied in the past to equipment in safety systems that is not part of the protection system.

54. The Board erred in misinterpreting IEEE Std. 603, particularly Sections 4.4 and 3.10. (PID 737-738)

The Board found that "[t]here is no basis to apply the IEEE Std 279 to the completion of a subsequent safety function." (PID 729) UCS conceded that a literal reading of IEEE 279 extends its scope only from the sensors to the actuation device input terminals. (UCS PF 287) The literal language of the standard could be met by a design which permits the operator to terminate an automatically initiated safety system as soon as an actuation signal has been transmitted to the input terminals and before the safety system has even begun to operate.

However, UCS testified that such an interpretation of the standard ignores the purpose of the standard and would permit trivial differences unrelated to the safety of the design to determine the acceptability or unacceptability of safety system designs. (Pollard, ff. Tr. 6410, at 10-4 to 10-6)

The example given by UCS compared a design where the operator can interrupt the initiation signal into the actuation device input terminals and a design where the operator can interrupt the initiation signal on the other side of the actuation device input terminals. According to the staff's view of the standard, the former design would violate IEEE Std 279 and the latter would not, but in either case the safety function would not go to completion. (Pollard, ff. Tr. 6410, at 10-5 to 10-6) It is thus apparent that the narrow interpretation of the requirement renders it nearly valueless to assuring plant safety.

Nor did the Board consider the testimony establishing that such a narrow application of its requirements overlooks the history of IEEE Stds. 279 and 603, the NRC's past practice in applying the standards, and the lessons to be learned from the accident. (Pollard, ff. Tr. 6410, at 10-4)

Several examples were given of cases where IEEE 279 has been applied to equipment that is not strictly part of the "protection system." (UCS PF 292-297) The staff conceded that the "principles" of the standard have been so applied. (Tr. 6626-6627, Sullivan) Moreover, IEEE Std 603 is being developed explicitly to extend IEEE 279 to the systems <u>actuated</u> by the protection system. (UCS PF 298-301) Paragraphs 736-739 of the PID misconstrue IEEE Std 603 by failing to put Section 3.10.3 in its proper context. Section 3.10 is entitled "Design Basis" and requires the design basis to document, <u>inter alia</u>:

> 3.10.3 The plant conditions after which a deliberate operator intervention may prevent the completion of protective action at the system level.

UCS's testimony was that this is entirely consistent with and supportive of its interpretation of IEEE Std 603, to wit: Section 3.10 of IEEE Std 603-1977

requires, in part, that the design basis for a safety system set forth "[t]he plant conditions after which a deliberate operator intervention may prevent the completion of protective action at the system level," and "[t]he point in time, or plant conditions, which define completion of the protective action at the system level." (UCS Ex. 15, at 13) Section 4.4, "Completion of Protective Action," of IEEE Std 603-1977 then requires that "[t]he safety system shall be designed so that, once initiated automatically or manually, the intended sequence of protective actions at the system level shall continue until completion," except that "[t]his requirement shall not preclude . . . the provision for those operator interventions which are identified in 3.10 of the design basis. (UCS Ex. 15, at 14, emphasis added) Thus, the standard calls upon a designer to define those conditions where operator intervention is permissible and then to design the plant so that operator intervention is precluded unless those conditions are met. (Pollard, ff. Tr. 6410, at 10-17 to 10-19; See UCS PF 298-304) The Board simply failed to discuss this testimony.

The staff attempted to meet the UCS argument by claiming that the pertinent provisions of IEEE Std 603 are not exclusive - i.e., they do not preclude operator intervention during any unspecified conditions. (Tr. 6623-6624, Sullivan) This position is not supportable. If the staff were correct, the requirement to specify the design basis conditions under which operator intervention in safety system operation is acceptable would be a nullity. Furthermore, the requirement for a protective action to continue until completion could simply have been worded to allow operator intervention at any time rather than allowing operator intervention during those plant conditions specified in the safety system design basis. (UCS PF 303-304) Once again, the Board failed to mention or consider these arguments.

The Board's decision with respect to the proper interpretation of IEEE 279 and 603, in the light of the TMI-2 accident, failed to confront the evidence presented by UCS.

Exceptions:

49. The Board erred in giving weight to Licensee's testimony that the provision of automatic circuitry to prevent the operator from prematurely terminating a protective action would be impractical, would seriously complicate the plant and would detract from safety. (PID 741)

50. The Board erred in disregarding and failing to confront evidence demonstrating that the provision of automatic circuitry to prevent the operator from prematurely terminating a protective action would not be impractical, would not seriously complicate the plant and would not detract from safety. (PID 741)

51. The Board erred in disregarding and failing to confront the fact that neither the Licensee nor the Staff were able to postulate any scenario where prohibiting premature termination of an automatically initiated protective action would pose a hazard to public safety.

55. The Board erred in not articulating its balancing, if any, of the merits of the opposing positions of UCS and the Licensee/Staff. (PID 746)

56. The Board erred in not confronting substantial evidence that the safety advantages of UCS's position are far greater than those to be obtained by Licensee's position. (UCS PF 271-284)

At PID 741, the Board states the assertions of Licensee's witness Clark to the effect that the provision of automatic circuitry to prevent the operator from prematurely terminating a protective action would be impractical, would seriously complicate the plant and would detract from safety. At paragraphs 742-744, it further elaborates on the Licensee's testimony and at 746 concludes that, while the UCS position "has merit" in light of the TMI-2 accident, the Licensee and Staff prevail "upon the record of this proceeding." Reading only this, one would scarcely believe that UCS was present, much less that we presented substantial evidence and arguments which directly contradict the Licensee's and Staff position, which was further weakened during cross-examination. This portion of the PID is remarkable for its wholesale failure to confront the record. (See UCS PF 253-285) The record is summarized as follows:

The Licensee and B&W have clearly defined the plant conditions constituting completion of a safety function, e.g., TMI-1 emergency procedures state that HPI should not be throttled unless LPI flow is greater than 1000 gpm in both loops and stable for over 20 minutes or the degree of subcooling in the primary system is at least 50°F and throttling is necessary to prevent pressurizer level from going off-scale high. (UCS PF 254-258) These procedures are specific (Tr. 6246, Ross) and even take precedence over the normal restrictions on pressure-temperature limits for the reactor vessel. (Lic. Ex. 48, at 8.0) The UCS position is that, given that these conditions have been clearly defined, the plant can and should be designed so as to preclude termination of safety systems until those conditions are attained. (UCS PF 257) It should be noted that in future potential accidents, despite the emergency procedures, the operator may again be confronted with control room indications that could lead to premature termination of safety systems. (UCS PF 260-262)

The Licensee's witness Clark, who asserted that an interlock system to achieve this goal would be complex and a detriment to safety, had only a vague understanding of the circuitry in question, did not understand the pertinent emergency procedures and postulated an unnecessarily complex circuit to define stability. (Tr. 6277-6278, 6280, Clark) Actually, as UCS testified, relatively minor circuit modifications could be used to incorporate the same signals used by the operator, without adding major complexity. (Tr. 6431-6432, Pollard)

The members of the IE TMI Investigation Team who testified on March 18, 1982, in connection with UCS's pending motion to reopen the record (See "Union of Concerned Scientists Comments Subsequent to Preliminary Hearing of March 18, 1982, Concerning the 'Martin Report'", March 26, 1982, at 13-14) confirmed UCS's testimony in this regard. They had independently recommended a physical hardware lock-in device to prevent operator interference with the operation of ESF equipment. (Tr. 27,141, Hunter) Interlocks to perform this general function

are used at other plants and are not a particular source of failure. (Tr. 27,142 - 27,144, Hunter) Such a system could be over-ridden under predetermined management controls, thus negating Licensee's fear about permanently "locking out" the operator. (Tr. 27,142 - 27,146, Hunter) While the Licensing Board has not yet ruled on whether this evidence will be accepted, we cite it for the Appeal Board's consideration. Since our present intention is to appeal a decision to exclude it, if that should be the decision, this would come before the Appeal Board in any case.

As to the vague assertions that this system would be a detriment to safety, they were convincingly rebutted. (UCS PF 271-285) None of the witnesses who testified on this contention could identify even a single situation involving emergency core cooling, auxiliary feedwater or containment isolation where preventing premature termination would pose a hazard to the public. The Licensee's witness purported, in direct testimony, to give several examples of situations during which termination of safety systems was necessary. (Clark, et al, ff. Tr. 6225, at 6, 7) During cross-examination of the witness, it was demonstrated that none of the examples was relevant to the contention. (Tr. 6291 - 6292, Clark)

The Staff's witness was not able to discuss the specifics of the TMI-1 design and instead postulated an abstract design and common mode failures. (Tr. 6641-6646, Sullivan) In the end, the Staff could give no example where any conceivable hazard to the public could result from operation of a safety system prior to completion of its safety function.

Failing at this, the Licensee and Staff took another tack; they argued that the design advocated by UCS might preclude correct operator action for an unforeseen (and unforeseeable) event. (Tr. 6299-6300, Clark; Tr. 6646, Sullivan) This proposition, too, was rebutted by UCS. (UCS PF 276-285)

UCS's testimony was that, in order to consider the merits of this position, one must view the issue in the following way. On the one hand, UCS proposes a design which permits completion of a safety function, such as core cooling, for all foreseen accident sequences, both within and outside the design basis. It prevents the operator from prematurely terminating the operation of safety systems during design basis accidents and unforeseen accidents only until the conditions defined as completion of the safety function for design basis accidents are met. On the other hand, Licensee and Staff support the current design which has the disadvantage of risking safety system termination prior to completion of a safety function during design basis (or foreseeable) accidents in exchange for the advantage of allowing the operator maximum freedom for unforeseen events. These two alternatives must be balanced against each other. (Tr. 6423-6426, Pollard)

The following questions, then, become important: First, what is the probability of the occurrence of an unforeseen event? If it is exceedingly low, the potential advantage of freedom for operator action becomes correspondingly small at the outset.

Second, what is the probability that, should an unforeseen accident occur, proceeding in accordance with current emergency procedures - that is, permitting the safety systems to operate until the safety function is completed - would be the incorrect response? Obviously, if the operation of the safety systems as currently called for is the correct response (as it was during the TMI-2 accident, an unforeseen accident sequence), then inhibiting the operator's ability to prematurely terminate such systems has no disadvantage.

Finally, even assuming all the above, what is the probability that the operator in the midst of this unforeseen accident will divine and take the appropriate action of terminating operation of the safety system?

UCS's witness testified that the probability of an unforeseen event for which following the emergency procedures is incorrect and for which the operator takes correct action is so low that a design allowing operator interference with safety system operation prior to completion of its function is not worth the risk of improper operator action during a design basis accident. (Tr. 6423-6426, 6563-6564, Pollard)

The Licensee's witness had no opinion on the probability of the foregoing combination of events, although he was unwilling to even agree that the probability of an unforeseen accident sequence was lower than the probability of a design basis accident. (Tr. 6255, Clark) Of course, if that were true, TMI-1 could not be permitted to restart without regard to UCS's contention since there would be no basis upon which to find reasonable assurance that the plant can be safely operated.

The Staff's witness simply espoused the "philosophy" that because one might not be able to think of unforeseen events is no excuse not to protect against them. He did not eleaborate whatever on how to accomplish that protection. (Tr. 6642, Sullivan)

Thus, the Licensee and Staff's arguments can generally be categorized as of the "arm-waving" variety. There is nothing on this record to support either their conclusions or the Board's bare finding that they "prevail." When the advantages and disadvantages of the two positions are balanced (which the Board never did), UCS should prevail.

Exception:

52. The Board erred in misstating the UCS position and the issue litigated. UCS does not advocate a design which "removes operator intervention under any and all circumstances." (PID 746) UCS advocates a design which conforms to IEEE Std. 603: that the conditions under which safety system termination is permitted or called for are specified in the design basis. (PID 739, 746)

See UCS PF 254-258, 274, 300-302, 305, 311.

Exception:

53. The Board erred in, while noting the importance of operator training, disregarding and failing to confront evidence demonstrating that in future potential accidents, operators may again be confronted with a sequence of events causing control room indications that could lead to premature termination of ECCS, EFW or containment isolation because of inadequate instrumentation, inadequate training and requalification of operators and the current absence of required analyses, procedures and training to improve operator performance during transients and accidents. (PID 744, 746)

As in several places throughout the PID, the Board at PID 744 and 746 "notes" the "importance" - indeed "extreme importance" - of "adequate procedures and thorough training" of operators to its resolution of the issue. One can only conclude that the Board has found, albeit unstated, that the procedures and training are adequate in this "extremely important" instance (putting aside the cheating issues), since it has authorized restart.

However, the Board failed to discuss evidence indicating that, in the future, the operators may again be confronted with a sequence of events causing unforeseen control room indications that could lead to premature termination of ECCS, EFW or containment isolation. (UCS PF 260-261) Nor does the TMI post-accident training and requalification engender confidence that the operators can be relied upon to react appropriately under a range of accident conditions. (UCS PF 262) Moreover, the analyses, procedures and training required to significantly improve operator performance during transients and accidents have not yet been provided. (PID 706-721)

G. UCS Contention 12
Exceptions 57-64
PID 1139-1181
UCS Proposed Findings 631-724

Summary

UCS Contention 12 is as follows:

> The accident demonstrated that the severity of the environment in which equipment important to safety must operate was underestimated and that equipment previously deemed to be environmentally qualified failed. One example was the pressurizer level instruments. The environmental qualification of safety-related equipment at TMI is deficient in three respects: 1) the parameters of the relevant accident environment have not been identified, 2) the length of time the equipment must operate in the environment has been underestimated, and 3) the methods used to qualify the equipment are not adequate to give reasonable assurances that the equipment will remain operable. TMI-1 should not be permitted to resume operation untill all safety-related equipment has been demonstrated to be qualified to operate as required by GDC 4. The criteria for determining qualification should be those set forth in Regulatory Guide 1.89 or equivalent.

While Licensee and Staff continually refer to this contention as "abandoned" by UCS (See PID 1153), it manifestly was not. UCS moved the Board to adopt the contention as its own which it agreed to do, adding the following questions (none of which were answered in the decision):

> Board Questions Regarding UCS Contention 12:
>
> 1. The TMI-2 accident demonstrated that some safety-related equipment may have been exposed to environmental stresses beyond that for which it was qualified. The board's concern is primarily with such equipment qualification. In addition, environmental stresses to safety-related equipment will be of concern to the extent that such equipment is not included in existing staff requirements.
>
> 2. Which items of Regulatory Guide 1.89 have been grandfathered with respect to TMI-1? Explain any justification for allowing restart without compliance with the grandfathered items.
>
> 3. What are the environmental qualification criteria which equipment inside of containment must meet with respect to radiation levels and length of time of exposure? (Address the Interim Staff Position on Environmental Qualification of Electrical Equipment, NUREG-0588).
>
> (PID 1139)

UCS did extensive cross-examination on this issue and submitted 44 pages of detailed proposed findings of fact. We urge the Appeal Board's attention to these findings, particularly since a reading of PID alone gives little hint of the fact that there is a factual record on UCS Contention 12. This record demonstrates that safety systems in TMI-1 cannot be shown to be qualified to

withstand accident conditions as required by GDC 4 (UCS PF 632-655), that the record does not even establish that safety equipment can withstand a design basis SBLOCA (1% fuel failure) (UCS PF 661-668, 680-702), and even that the new equipment and instrumentation installed as part of the TMI-2 lessons learned has not been reviewed for qualification (UCS PF 669-679).

Moreover, nothing in the record supports a finding that "reasonable progress" has been made toward qualifying safety-related equipment in TMI-1. The only piece of evidence bearing on that issue was the Staff's safety evaluation report offered by UCS which was <u>excluded</u> by the Board on the basis of Staff objections. (PID 1152, 1162; UCS PF 647, 650)

The Staff chose, inexplicably, to present evidence <u>only</u> on the ability of equipment to survive a SBLOCA with a maximum of 1% fuel failure - a small fraction of the severity of the TMI-2 accident. (UCS PF 634-654; PID 1151-1155) The Board found the Staff's evidence to be "useless" (PID 1155) and held that "the Staff has defaulted and the decision must rely chiefly on Licensee's testimony and argument." (PID 1156)

The only Licensee testimony even arguably relevant was that of Mr. Braulke who presented only the vaguest generalities to the effect that the Licensee was making progress toward complying with the Commission's order in CLI-80-21. (PID 1157; UCS PF 720-723) The Licensee offered no specific evidence with respect to the qualification of any real equipment. When UCS offered the Staff's safety evaluation report which has been prepared to demonstrate compliance (or non-compliance) with CLI-80-21 in an attempt to question such general conclusions, it was rejected. As the Board itself admitted, on the basis of this record it was unable even to find out which equipment will not be qualified at restart or to make "at least a qualitative judgment of the risk of allowing interim operation prior to June 1 [sic], 1982." <u>But the Board had "no basis for such a judgment</u>" in either the Staff or Licensee testimony. (PID 1157)

Faced with this total void of evidence in the record to support a finding favorable to Licensee, the Board in effect dismissed the contention on "legal" grounds. While finding at PID 1181 that UCS "demonstrated that all of the safety equipment at TMI-1 will not meet all the criteria of Regulatory Guide 1.89 at time of restart" (a statement which by nature of its generality distorts the record by failing to acknowledge the breadth and nature of the environmental qualification problem demonstrated by the UCS proposed findings), the Board stated its belief that the Commission's decision in Petition for Emergency and Remedial Action, CLI-80-21, 11 NRC 707 (1980) resolves the issues. (PID 1161, 1181) The Board stated at PID 1161:

> With this in mind the Commission's guidance in CLI-80-21 and the DOR guidelines are very appropriate and convenient. They subsume a TMI-2 type accident. We see no basis upon which to treat TMI-1 differently than other operating reactors on the issue of radiation environmental qualification of electrical equipment. By virtue of CLI-80-21, June 30, 1982 is a reasonable time for compliance, and we have the testimony of Mr. Braulke cited above, that the Licensee has made reasonable progress toward meeting this date. (emphasis added)

Thus, the Board has determined that June 30, 1982 is a "reasonable time" for compliance with GDC 4 on the basis of CLI-80-21 and that Mr. Braulke's testimony that Licensee has made reasonable progress is enough on the record for the Board to issue its decision. The latter finding - reliance on the Braulke testimony - is inexplicable and clearly erroneous in light of the fact that, as the Board acknowledged at PID 1157, the witness' statements were so conclusory and without supporting facts as to preclude the Board from making any judgment as to their accuracy. Moreover, UCS was prevented from probing them by the Board's rejection of its proffer of the TMI-1 SER on environmental qualification. At the most, Mr. Braulke's testimony was that upon completion of the effort mandated by IE Bulletin 79-01B, there will be reasonable assurance that equipment used to protect the public health and safety is environmentally

qualified. (UCS PF 722) That is, of course, no more than saying that when we meet the regulations, we will meet the regulations. It provides no basis for a finding that the plant is safe enough to start operation now.

Nor did the Staff make any review whatever of whether the Licensee has made reasonable progress toward achieving full qualification of safety-related equipment in accordance with GDC 4. (UCS PF 648-650) In apparent recognition of the total lack of evidentiary support for a finding of "reasonable progress," the Board adopted an unprecedented and bizarre procedure; it directed the Staff "to certify to the Commission, for review in immediate effectiveness, a report on Licensee's compliance with CLI-80-21 as it relates to equipment functioning in a radiological environment in a TMI-2 type accident." (PID 1161) Since the record manifestly cannot support either a finding that GDC 4 is met or a finding that the Licensee has made reasonable progress toward meeting it, rather than find as it should in favor of UCS and against restart, the Board has permitted the Staff, an adversary party, to "certify" a new record to the Commission - a record which will exclude UCS, be made only by the Staff and Licensee and which will take the place of the hearing record.

This procedure is offensive to the most basic principles of administrative adjudication. A decision must be made on 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 Trans World Airlines v. Civil Aeronautics Board, 254 F.2d 90 (D.C. Cir. 1958); FTC v. Atlantic Richfield Co., 567 F.2d 96, 102 (D.C. Cir. 1977); King v. Caesar Rodney School District, 380 F.Supp. 1112, 1118 (D. Del. 1974). Based on this record, there is only one finding that the Board could make - that the plant is not safe enough to restart.

Finally, the Board's interpretation of CLI-80-21 is wrong. As UCS pointed out, the Commission did not grant therein any general dispensation from meeting GDC 4. (UCS PF 659-660, 717-719) The Commission did not intend the deadline of June 30, 1982 for demonstrating qualification in accordance with new standards to permit Licensees to operate with unqualified equipment until that date: "These deadlines, however, do not excuse a Licensee from the obligation to modify or replace inadequate equipment promptly." (11 NRC 707, at 715, emphasis added) The Commission continued:

> During its review, the Staff will be faced with many situations where qualification documentation is poor or where the existing documentation raises questions about the ability of the equipment to perform its intended function in accident conditions. In such cases, the Staff will make a technical judgment regarding continued operation. (*Id*., emphasis added)

The Staff has made no such "technical judgment" with respect to equipment in TMI-1, despite the fact that even a cursory perusal of the Staff's safety evaluation report (UCS Ex. 40) shows many components which lack qualification. Thus, there is no basis for believing that vital safety equipment would survive tthe accident environment. Under these circumstances, there is not reasonable assurance that the plant can be safely operated. Nothing in CLI-80-21 changes the fact that the Commission is obliged to ensure the safety of the facilities it licenses throughout their lifetime, nor could it.

Nor does the fact that TMI-1 is to be treated as an operating plant relieve the Board of its obligation. The fact that there may be other plants with similar deficiencies does not provide an excuse for TMI-1 nor any justification for permitting it to operate with known deficiencies. The Board was not divested of its responsibility to find assurance that this plant can safely operate simply because other plants might be unsafe. Moreover, such an attitude would be fundamentally inconsistent with the Commission's direction that the June 30, 1982 generic deadline does "not excuse a Licensee from the obligation

to modify or replace inadequate equipment promptly." Petition for Emergency and Remedial Action, CLI-80-21, 11 NRC 707, at 715.

In this case, the Staff and Licensee called the Board's bluff and the Board backed down. (Tr. 21,881 - 21,921) The Staff told the Board that it had artificially limited its review to the ability of TMI-1 safety equipment to withstand a 1% fuel failure SBLOCA - far less severe than the TMI-2 accident - and that if the Board refused to so restrict the contention, it would take "a lot more time" to decide the "full qualification issue." (Tr. 21,888-9, Cutchin) The Board ruled unequivocally that it did not consider the contention to be restricted to the "design basis" SBLOCA. (Tr. 21,912 - 21,922) However, when neither the Staff nor the Licensee presented evidence establishing the ability of safety equipment in TMI-1 to survive a TMI-2 type accident, the Board in effect threw up its hands, tossed out the record, and passed the issue to the Commission.

Exception:

57. The Board erred in limiting consideration of UCS Contention 12 to small break LOCA'S and failing to accept evidence related to high energy line breaks and main steam line breaks.

UCS's contention is that the TMI-2 accident, which happened to be a small break LOCA, demonstrated that the equipment in TMI-1 did not meet the requirements of GDC 4. (Tr. 21,909 - 21,910, Pollard) UCS stated clearly and as early as the Special Prehearing Conference on November 8, 1979:

> The parameters of the relevant accident have not been identified;
>
> The length of time the equipment must operate has been underestimated . . .;
>
> And the methods used to qualify the equipment were not adequate to give reasonable assurance that the equipment will remain operable during the period required.
>
> I think it is clear that what is needed is essentially a reassessment of the environmental qualification of safety-related equipment in light of the Lessons Learned from the accident. I think the primary of those lessons is that we haven't really understood the environment in which safety-related equipment will be called upon to perform its function or we haven't succeeded in bounding it properly. (Tr. 236)

In other words, in UCS's view, the pertinent lesson learned from the accident is that safety-related equipment previously deemed to be qualified cannot, in fact, be demonstrated to be capable of withstanding the effects of potential accident conditions. TMI-2 was a small break LOCA, but there is no logical reason to look only at small break LOCA's, much less design basis small break LOCA's. (Tr. 21,909 - 21,910, Pollard)

The Board did not allow questioning related to the ability of safety equipment in TMI-1 to withstand a high energy line break or a main steam line break on the ground that such accidents had no nexus to the TMI-2 accident. (Tr. 21,903-4, 21,908 - 21,910, Pollard; Tr. 21,920 - 21,923, Smith, Tourtellotte)

The Board's notion of "nexus" is unduly mechanistic. The only limitation on the contention was that it covered only equipment within the containment and auxiliary building. (Tr. 21,885) As UCS argued, the question raised by the TMI accident is not just the ability of safety equipment to survive a SBLOCA, but whether accident parameters in general have been properly identified and whether the methods used to qualify equipment do indeed ensure compliance with the Commission's minimum requirements for safety contained in GDC 4. (Tr. 21,890-3, 21,903-4, 21,909-11, Pollard) The Board's exclusion of questions related to the ability of equipment to survive main steam line break and high energy line break accidents results in the anomaly that while the reliability of emergency feedwater was a central issue in the case - no questioning could be permitted with regard to the ability of EFW components to survive a high energy line break in the auxiliary building. (Tr. 21,893, Pollard)

Exceptions:

58. The Board erred in failing to admit the environmental qualification SER for TMI-1. (UCS Ex. 40)

59. The Board erred in directing the Staff to "certify" to the Commission a report on Licensee's compliance with CLI-80-21. (PID 1162)

As discussed above, the only document which contains any information on the status of environmental qualification for TMI-1 (beyond the design basis SBLOCA) is the environmental qualification SER for TMI-1, UCS Exhibit 40. The Staff successfully objected to the admission of its own official document on the ground that it was irrelevant (Tr. 22,076) to the Staff's definition of the scope of the proceeding; that is, it included information going beyond the question of qualification for a design basis SBLOCA. (Tr. 22,076, 22,078, Cutchin) That was the sole basis of the objection:

> MR. CUTCHIN: My objection is it is a staff SER and the staff has narrowed its review to what it perceives to be the scope of this proceeding and has offered direct evidence on this subject, and I think it'll be confusing.
>
> CHAIRMAN SMITH: Confusing isn't the test.
>
> DR. JORDAN: But there is much in it that is the basis, it seems to me, for the staff's testimony, that I really find out for the first time some of the reasons why staff said the things they did. Without this I would have had a real problem.
>
> CHAIRMAN SMITH: I think Dr, Jordan has resolved that dispute, so your objection is overruled. It is, of course, received in evidence soley to demonstrate that there is such a document. But it is not received in evidence -- we don't by receiving it in evidence, we do not thereby conclude that the items set forth in the SER were within the scope of this proceeding. Our rulings will be the traditional way we've made them. (Tr. 22,078-9)

While the Board states at PID 1152 that the substance of the document was not received because it lacked a sponsoring witness, that ground was not raised in the objection nor in the Board's above-quoted ruling. Nor does the Board need a sponsoring witness to accept an official document of the NRC. [10 CFR 2.743(h) and (i)]

We understood the Board to be ruling that it accepted the relevant portions of the document, relevance to be determined by its ultimate ruling on whether the scope of the contention could be limited to environmental qualification for a design basis SBLOCA. That is consistent with the fact that lack of relevance

was the only objection raised. We believe that the Board has mischaracterized this ruling and thereby allowed the Staff to, in effect, enforce its own restricted view of the scope of the issue despite the fact that the Board had ruled against the Staff on scope.

The Board now considers it "unfortunate" that the SER was not admitted (PID 1162) and, as a substitute for a record on which to base its restart decision, "instead now directs the Staff to certify to the Commission, for review in immediate effectiveness, a report on Licensee's compliance with CLI-80-21 as it relates to safety equipment functioning in a radiological environment in a TMI-2 type accident." (PID 1162) As discussed supra at 79, this remarkable procedure violates the fundamental principles that a decision must be made on the record after parties have been given a fair chance to participate and that an adversary party, the Staff, cannot also be the decisionmaker.

The Board should have accepted the SER and found that it shows on its face numerous instances of failures in environmental qualification. (UCS PF 640, 641) No showing was ever made that the plant is safe enough to operate despite these deficiencies. This record does not justify a decision favorable to Licensee.

Exceptions:

60. The Board erred in misinterpreting CLI-80-21 by ignoring those sections of the Commission's order which state that the June, 1982 deadline does not excuse Licensees from the obligation to modify or replace inadequate equipment promptly and require a technical judgment justifying continued operation in cases where documentation is poor or raises questions about the ability of equipment to perform its intended function in accident conditions. (PID 1161, 1162, 1181)

61. The Board erred in finding that the question of interim operation of TMI-1 has already been addressed and decided by CLI-80-21. (PID 1181)

62. The Board erred in failing to find that the Staff made no "technical judgment" with respect to many components in TMI-1 for which qualification deficiencies have been found.

64. The Board erred in authorizing the operation of TMI-1 when the evidence demonstrates that it fails to meet the requirements of GDC 4, Regulatory Guide 1.89, and the DOR Guidelines.

These exceptions go to the Board's misinterpretation of CLI-80-21, which it used to circumvent the issues raised by UCS and the force of the finding that "all of the safety equipment at TMI-1 will not meet all the criteria of Regulatory Guide 1.89 at the time of restart." (PID 1181) As demonstrated supra at 80 (See also UCS PF 658-660, 717-722), the "deadline" in CLI-80-21 does not excuse Licensees from the obligation to promptly replace equipment that is discovered during review to be deficient. Even in the case of equipment for which documentation is missing or inconclusive (as opposed to cases where the lack of qualification is proven), the Staff was directed unequivocally to make a "technical judgment" as a necessary basis for permitting continued operation. Except for the design basis SBLOCA, no such judgment was made by the Staff here with respect to equipment shown on the face of UCS Exhibit 40 to lack qualification or documentation. The Staff looked only at a 1% fuel failure SBLOCA and even for that case did little more than check to see whether the Licensee claimed that the pertinent equipment was qualified and whether the Licensee claimed that qualification documentation existed. (UCS PF 660-668, 673, 677)

Exception:

63. The Board erred in failing to find that TMI-1 does not meet GDC 4.

UCS requested the following finding (UCS PF 640):

> One undisputed fact that must be acknowledged at the outset: There has been no demonstration on this record that safety-related equipment in TMI-1 is environmentally qualified in accordance with General Design Criterion 4. In fact, the evidence is to the contrary. The Safety Evaluation Report on environmental qualification discussed by Mr. Rosztoczy in his November 26 testimony, issued on March 24, 1981 and supplemented on May 23, 1981, was introduced into evidence by UCS. (UCS Exhibit 40, Tr. 22,086) It identifies literally dozens of safety-related components for which environmental qualification has not been established. Thus, unless the Board is willing to narrow the scope of Contention 12 to conform to the scope of the Staff's testimony, UCS will perforce prevail.*
>
> * The Board does not intend by this statement to indicate that the contention fails if its scope is narrowed. The question of whether the Staff's testimony is sufficient to support affirmative findings even on the qualification of equipment needed for a loss of feedwater and a small break LOCA is treated later.

While the Board found a way to avoid the ultimate conclusion that UCS should prevail by relying on CLI-80-21, that does not vitiate the fact that we are entitled to a finding that, on the basis of this record, TMI-1 does not meet GDC 4. See also UCS PF 648, 649, and 650. The Board does not respond directly to these proposed findings although it did find that UCS has "prevailed to the extent that UCS has demonstrated that all of the safety equipment at TMI-1 will not meet all the criteria of Regulatory Guide 1.89 at the time of restart." (PID 1181)

There is a significant difference between failure to meet a Regulatory Guide and failure to meet a regulation. As the Board recognized in another context, Regulatory Guides do not have the force of law. (See PID 769) Regulations do. Regulations state the minimum requirements for safety. Compliance with GDC 4 is not waivable. UCS is entitled to a finding on this record that GDC 4 has not been met.

H. UCS Contention 14

Exceptions 65-85
PID 198-216
UCS Proposed Findings 472-549

Summary

UCS Contention 14 is as follows:

> The accident demonstrated that there are systems and components presently classified as non-safety-related which can have an adverse effect on the integrity of the core because they can directly or indirectly affect temperature, pressure, flow and/or reactivity. This issue is discussed at length in Section 3.2, "System Design Requirements," of NUREG-0578, the TMI-2 Lessons Learned Task Force Report (Short Term). The following quote from page 18 of the report describes the problem:
>
> "There is another perspective on this question provided by the TMI-2 accident. At TMI-2, operational problems with the condensate purification system led to a loss of feedwater and initiated the sequence of events that eventually resulted in damage to the core. Several nonsafety systems were used at various times in the mitigation of the accident in ways not considered in the safety analysis; for example, long-term maintenance of core flow and cooling with the steam generators and the reactor coolant pumps. The present classification

system does not adequately recognize either of these kinds of effects that nonsafety systems can have on the safety of the plant. Thus, requirements for nonsafety systems may be needed to reduce the frequency of occurrence of events that initiate or adversely affect transients and accidents, and other requirements may be needed to improve the current capability for use of nonsafety systems during transient or accident situations. In its work in this area, the Task Force will include a more realistic assessment of the interaction between operators and systems."

The Staff proposes to study the problem further. This is not a sufficient answer. All systems and components which can either cause or aggravate an accident or can be called upon to mitigate an accident must be identified and classified as components important to safety and required to meet safety-grade design criteria.

In summary, UCS's testimony explained the significance in nuclear safety regulation of the distinction between safety-grade and non-safety-grade systems and components and described how the TMI-2 accident demonstrated three types of shortcomings in past practice: 1) certain systems previously classified as not safety-related are, in fact, important to safety (UCS PF 482); 2) some systems known to be important to safety do not meet all of the criteria applicable to such systems (UCS PF 483); and 3) the design basis for judging the capability of safety systems has not been properly specified (UCS PF 484).

Despite NRC's general requirement that failure of non-safety grade equipment should not initiate or aggravate an accident, there is currently no comprehensive and systematic analysis done to demonstrate that this requirement has been met. In other words, the elaborate structure for ensuring diverse and redundant safety systems to mitigate accidents remains vulnerable to unforeseen failures of non-safety equipment, or adverse systems interactions, just as during the TMI-2 accident. In the aftermath of the accident, no systematic effort has been made to identify and correct this problem. Therefore, UCS proposes that all systems currently classified as non-safety-related which can in fact either cause or aggravate an accident or be called upon to mitigate an accident should be identified and required to meet safety-grade criteria so as to preclude adverse interactions. (Pollard, ff. Tr. 8091, at 14-1 to 14-9)

UCS's testimony described the manner in which the NRC's licensing process depends upon assessing whether the plant's structures, systems and components can be relied upon to protect public health and safety in the event of occurrence of any of the selected design basis accidents or anticipated operational occurrences. The Commission has developed a set of regulations that define the minimum requirements for design, fabrication, construction, testing and performance which must be met if a structure, system or component is relied upon to protect the public. These requirements are set forth in the General Design Criteria of Appendix A to 10 CFR Part 50, industry standards such as IEEE Std. 279, which are incorporated in 10 CFR Section 50.55a, and other sections of 10 CFR Part 50. (Id. at 14-3)

The introduction to the General Design Criteria provides as follows:

> Pursuant to the provisions of Section 50.34, an application for a construction permit must include the principal design criteria for a proposed facility. The principal design criteria establish the necessary design, fabrication, construction, testing, and performance requirements for structures, systems, and components important to safety; that is, structures, systems, and components that provide reasonable assurance that the facility can be operated without undue risk to the health and safety of the public. (App. A, 10 CFR Part 50, Introduction, emphasis added)

As the language quoted above indicates, Commission policy has been to apply the requirements of the GDC to systems variously referred to as safety-related, safety-grade or important to safety. It is assumed that only safety-grade systems are available to function during a design basis event. Non-safety-grade systems are, by contrast, assumed to be unavailable and therefore, their functioning is not credited in evaluating the protection available to mitigate such events. (Pollard, ff. Tr. 8091, at 14-3 to 14-4)

As additional support of this description of the licensing process, UCS cited the language from the NRC's advance notice of proposed rulemaking. quoted at PID 979.

Further confirmation can be found from the Lessons Learned Task Force itself, which described the classification system and noted that the present classification scheme does not adequately recognize that non-safety systems can (and did at TMI-2) cause accidents and can be (and were at TMI-2) used in accident mitigation in ways not considered in the plant's safety analysis. (The full quote from page A-18 of NUREG-0578 is contained in the text of UCS contention 14)

While we may appear to be belaboring this initial question concerning NRC's safety/non-safety classification scheme and its implications, the discussion is necessary because NRC's witness disputed UCS's description of the licensing process and the Board accepted his view. (PID 981)

UCS testified that the licensing review of TMI-1, while based on a fundamental distinction between "safety" and "non-safety" equipment, was not adequate to identify all systems which are important to safety, to define the design basis for such equipment, or to identify and prevent adverse interactions between non-safety and safety equipment which can compromise the ability of safety systems to perform their necessary functions. (Pollard, ff. Tr. 8091, at 14-7 to 14-8) The Lessons Learned Task Force conceded as much:

> The interactions between non-safety grade and safety grade equipment are numerous, varied, and complex and have not been systematically evaluated. Even though there is a general requirement that failure of non-safety grade equipment or structures should not initiate or aggravate an accident, there is no comprehensive and systematic demonstration that this has been accomplished." (NUREG-0585, p.3-3)

Thus, the lessons to be learned from the TMI-2 accident include that the current safety/nonsafety classification scheme does not adequately identify all systems important to safety or identify and correct all potentially adverse systems interactions. Nor do any of the short or long-term measures for TMI-1 address this problem, despite the fact that the ACRS has called for "timely" systems interaction studies for TMI-1. (UCS PF 487-490) While it appeared that

the Board had required a systems interaction study for TMI-1 [PID 1000, 1003(f)], the Staff has since informed the Board that it has no present intention to require such studies and the Board responded by stating that its intent was only that TMI-1 should be included if studies are done. (Memorandum and Order Modifying and Approving NRC Staff's Plan of Implementation, April 5, 1982 at 6-7) See also "Union of Concerned Scientists' Response to the Staff's Proposed Enforcement Plan," February 17, 1982, at 4, 5. Thus, there is no plan or requirement for any systems interaction studies for TMI-1, despite the ACRS recommendations.12/

The Staff responded to this contention by disputing UCS's description of the manner in which the concepts and terms "safety-related", "safety-grade" and "important to safety" have been used in licensing. (UCS PF 501-504, 509-518) According to the Staff witr r. Conran, only systems and equipment which perform "critical safety functions" (a term nowhere used in the regulations, Tr. 8530-1, Conran) need be safety grade, while other equipment "important to safety" need not be. Regulatory Guide 1.29, which deals with protection from earthquakes, is said to contain a list of all "safety grade" equipment. Thus, Mr. Conran challenges UCS's assertion that when a system is determined to be "important to safety", it has been required to meet the applicable GDC which form the definition of "safety grade." (Conran, ff. Tr. 8372 at 4-6)

Mr. Conran went on to testify that there is no need to fully upgrade any non-safety grade equipment which either contributed to or was used in mitigation of the TMI-2 accident. He states that three criteria are used by the Staff in deciding whether such upgrading is required:

12/ It should be noted in this connection that the "probabilistic risk assessment" cited by the Board in its April 5 Memorandum and Order at 6 is not the same thing. The ACRS called for both a reliability assessment and a systems interaction review for TMI-1. (UCS PF 488)

1. Will the failure of the non-safety component in and of itself degrade the capability of safety systems so that they cannot mitigate accidents?

2. Will the effects of failure of the non-safety system alone exceed the capability of properly-operated safety systems?

3. Is the non-safety system actually required to mitigate an accident assuming safety systems are properly operated? (Id., at 8 - 10)

According to Mr. Conran, if "by careful analysis or actual experience," the answer to any of these questions is yes, upgrading may be called for. (Id. at 10) However, he states that none of the TMI-1 non safety systems were used until after improper operation of safety systems had caused core damage. (Id. at 8) Nor did failure of non-safety systems cause the core damage. (Id. at 11) Hence, his criteria for upgrade are not met.

Mr. Conran then states that even though upgrade is not called for by application of his criteria, the Staff may decide to require partial upgrading "as a prudent measure", (Id. at 10) as it did with the PORV, pressurizer heaters and emergency feedwater. (Id. at 13-14) No criteria for the exercise of this prudence were offered.

In finding for the Staff and Licensee, the Board failed to confront substantial evidence showing that the Conran definitions were inconsistent with prior practice, with a fair reading of the rules, and were largely a post hoc attempt for purposes of this litigation to construct a factually logical explanation to support what the Staff has required (and failed to require) for TMI-1. (UCS PF 501-517) The Board also failed to confront evidence demonstrating that, even if one accepts Mr. Conran's definitions and proposed criteria for deciding when to upgrade equipment to safety grade, the evidence did not support a finding that the criteria had been applied properly to TMI-1. (UCS PF 518-524)

Finally, the Board mischaracterized UCS's position in a way which made it seem unreasonable and which dictated the result. The Board found at PID 981:

> The Board is of the opinion after hearing arguments and testimony on all sides for the question that the Staff's interpretation, especially that of Mr. Conran, is the one closest to the system actually used by the Staff. It is also the system which thich the Board feels should be employed. To argue otherwise would in one aspect of the question argue against making improvements in safety which would result in a safer system, without upgrading to a fully 'safety-grade' system. In other words such a viewpoint might discourage safety improvements to existing systems. We agree with Mr. Conran when he states that: 'The language of regulations typically is broadly drawn so as not to be too prescriptive -- to permit flexibility in the implementation of those requirements'. Tr. 8432.

See also PID 1003(b).

This issue came up at the hearing. (UCS PF 492) Indeed, the Licensee pursued almost exclusively this issue in cross-examination: are there circumstances where the Staff can and should mandate a partial upgrade of non-safety equipment without going all the way to full safety grade? While noting that past NRC practice in implementing the GDC prior to this case has not encompassed partial upgrade, Mr. Pollard stated that such partial upgrading might be justified from an engineering standpoint if it were based upon the results of technical analyses assessing the degree of improvement to safety gained by the partial upgrade, comparing that with the degree of improvement to be gained by full upgrade and establishing that the partial upgrade causes no adverse effects on plant safety. (Tr. 8123, Pollard) No such analyses have been done in this case. (Tr. 8613-8621, Conran) See UCS PF 520-524.

Thus, UCS took care to establish on this record that, with respect to the "partial upgrades" of the PORV, pressurizer heater, etc., no analysis was done to determine what degree of improvement has been achieved, what level of reliability has been attained, what would be needed to make the component safety-grade, what the reliability would then be, or what the resultant cost would be. (UCS PF 520-524) Under these circumstances, the platitudes offered by

th Staff and endorsed by the Board with respect to the need for "flexibility" (PID 981) are purely abstract, unrelated to this record, and provide no support for the Staff's position in this case.

Exception:

65. The Board erred in giving significant weight to the testimony of the Staff witness, J. Conran, and in finding that the witness was qualified to present the testimony he presented. (PID 1002)

The Board gives Mr. Conran's basic experience at PID 1002 and finds him qualified. In so doing, it discounts substantial evidence showing that while he has held a number of different positions at AEC/NRC, he has little if any experience in the systems interaction issue, the question pertinent to this contention. (UCS PF 493-503) The great bulk of his regulatory experience is in the safeguards field. His testimony indicated heavy reliance on the conclusions of other people (the accuracy of which he did not have the independent expertise to judge) or work that he assumed had been done by other people. (ucs pf 500; Tr. 8489-8492, 8545, 8547-8549, 8554, 8555-8559, 8607, 8614-8615, 8616-8618, 8620, Conran) He was assigned to present evidence on this contention only two weeks before it was filed. (UCS PF 498) Indeed, the witness conceded that his qualifications to present testimony with respect to the systems interaction issue were no greater than his qualifications for <u>any</u> of the safety issues raised by the TMI accident. (Tr. 8556, Conran)

The Board's ruling in this case can only serve to encourage the Staff to present witnesses with minimal personal experience in the subject at issue since there is not only no penalty for doing so, but a positive litigative advantage: a witness whose testimony is constructed out of whole cloth is not constrained by inconsistent history or experience since he is unfamiliar with history and has no prior experience.

Exceptions:

66. The Board erred in failing to find that NRC practice has consistently used the terms "safety-grade," "safety-related" and "important to safety" interchangeably.

68. The Board erred in failing to find that the definitions of "safety-grade," "important to safety" and "safety-related" presented by the Staff witness constituted a post hoc attempt to construct a factually logical explanation of Staff practice in order to support the Staff's opposition to the UCS contention.

69. The Board erred in finding that the interpretation of NRC's classification presented by Mr. Conran "is the one closest to the system actually used by the Staff." (PID 981)

70. The Board erred in disregarding and failing to confront substantial evidence demonstrating that the Staff witness misconstrued Regulatory Guide 1.29 and its application to the questions raised by UCS Contention 14.

71. The Board erred in finding that the UCS position would argue against or discourage making improvements in safety without upgrading to a fully safety-grade system. (PID 981)

In finding (PID 981) that the Conran interpretation of the terms "important to safety", "safety-grade" and "safety-related" is the appropriate one, the Board simply ignored all of the evidence discussed at UCS PF 509-517 and mischaracterized UCS's position as arguing against improvements to safety. The latter issue is the subject of Exception 71 and is discussed supra at 92. (See UCS PF 492, 520-524)

In fact, Conran's testimony was inconsistent with prior regulatory practice, internally inconsistent and self-serving. Mr. Conran was cross-examined extensively with regard to his assertion that the phrase "structures, systems and components important to safety" in the introduction to Appendix A to 10 CFR Part 50 is an extremely broad category and only equipment with "critical safety functions" need be safety grade and meet the applicable General Design Criteria. He was asked to identify equipment which is in his view not "important to safety." He identified the office building, rest room and water cooler. (Tr. 8404-6, Conran) He admitted that the term "critical safety function" is used nowhere in the regulations, but is rather his own term. (Tr. 8530, Conran)

Conran's definition of the terms "important to safety" and "safety grade" can be found nowhere in any AEC or NRC documents, regulations or regulatory guides. UCS's witness, who served as a member of the AEC and NRC licensing staffs and as a licensing project manager for 6 1/2 total years, has never seen these definitions nor heard them used in any NRC proceeding nor heard them in discussion with any NRC staff member. (Tr. 8099, Pollard) As discussed above, after Mr. Conran developed his testimony, it was circulated to staff members who were directed to conform their testimony to these definitions. (Tr. 8319, Conran) This suggests strongly that Mr. Conran's definitions were developed solely for the purpose of this case and have not appeared before in NRC practice.

Mr. Conran stated that Regulatory Guide 1.29 contains a list of all safety grade equipment. He derived this from reasoning that, since Regulatory Guide 1.29 lists equipment that is required to perform what he believes are "critical safety functions" after an earthquake, this list of equipment contains, ergo, all equipment that need be safety grade. (Conran, ff. Tr. 8372 at 4-5) While this has a veneer of logic, it does not stand up to scrutiny.

First it must be pointed out that Regulatory Guide 1.29 never states that the listing of systems and equipment contained therein constitues a list of all safety grade equipment. (Tr. 8537-8, Conran) Nor does any other NRC document so state. Indeed, the parties were asked earlier in the proceeding if such a listing existed and stated that it did not.

Moreover, GDC 2, which is the genesis of Regulatory Guide 1.29 explicitly requires that "structures, systems and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes..." (Tr. 8096, Pollard; Tr. 8531-2, Conran) Thus, one must conclude that the listing of equipment in Regulatory Guide 1.29, which bounds the coverage of GDC 2, is a listing of equipment "important to safety." This reinforces the

proposition that "important to safety" and "safety grade" are indeed interchangeable.

There is a circular and self-serving element in Mr. Conran's testimony. According to the witness, a system or component could be "important to safety" within the meaning of the introduction to the GDC, yet not be required to meet any of the specific GDC or Regulatory Guides, including even the quality assurance provisions of Appendix B to 10 CFR Part 50. The explanation offered is that, although the system or component is important to safety within the meaning of GDC 1, its level of importance is not enough to cause any specific requirements to apply. (Tr. 8409-8426, particularly 8419, Conran) Such an interpretation renders the phrase "important to safety" virtually meaningless as a regulatory concept since no regulatory consequences whatever flow from it. This provides an additional reason for the Board to discount the testimony.

Finally, the witness stated that his understanding and definitions had been applied during the licensing of TMI-1. (Tr. 8411, Conran) Yet there is equipment listed or covered by listings in Regulatory Guide 1.29 which is <u>not</u> safety grade for TMI-1, including the PORV and emergency feedwater system. (Tr. 8537-42, Conran) Mr. Conran testified that he doesn't know enough about the "details of the system" to know whether non-safety components are listed in Regulatory Guide 1.29. (Tr. 8633, Conran; See also Tr. 8692-6, Conran) Since this goes to the heart of his testimony, it cannot be treated lightly. If Regulatory Guide 1.29 is not even a listing of all safety-grade equipment (and none other), the "logical" construct built by Mr. Conran falls completely.

Because a system or component is important to safety, that does not mean that <u>all</u> the GDC apply, nor does UCS so argue. Certain of the criteria apply only to certain types of systems, e.g., the ECCS need not meet the criteria for containment heat removal. (Tr. 8096-7, Pollard) Moreover, the design basis for certain systems will determine whether particular GDC apply. For example,

systems needed only after an earthquake need not meet fire protection requirements. (Id.) However, once a system is determined to be important to safety, and its design basis established, it must meet the applicable GDC. That is what makes a system safety grade. (Tr. 8096-8101, Pollard)

The Board simply failed to confront any of this evidence.

Exceptions:

> 73. The Board erred in failing to confront substantial evidence demonstrating that, even assuming the correctness of the Staff's stated approach to determining the need for upgrading safety systems, the record does not support the Staff's conclusion that upgrading is not required.
>
> 74. The Board erred in failing to find that the Staff witness had no basis for claiming that, during the TMI-2 accident, non-safety systems were used only after improper operation of safety systems resulted in core damage.
>
> 75. The Board erred in failing to find that no "careful analysis" was done by the Staff to determine whether any non-safety-grade equipment should be upgraded.
>
> 76. The Board erred in failing to find that no reasoned "judgment" was ever exercised by the Staff to determine whether some systems or components should be partially upgraded as an exercise of "prudence."
>
> 77. The Board erred in failing to find that the Staff does not know whether the additional reliability to be gained by making the PORV or other equipment safety grade is "necessary" or desirable.
>
> 82. The Board erred in giving important weight to "Staff judgment" concerning the need to upgrade systems to safety-grade without considering substantial evidence demonstrating that the Staff's judgment in this case was not properly exercised. (PID 993)

These exceptions do not question the correctness of the Staff's definitions, nor the criteria which it put forward for determining whether a system previously classified non-safety should be upgraded, wholly or partially. They raise issues challenging the applicability of these definitions and criteria in this case.

The criteria proposed by Mr. Conran would require as a requisite to upgrading a showing that failure of a non-safety system by itself would cause

core damage or that use of a non-safety system was required to mitigate an accident assuming properly-operated safety stems. (Conran, ff. Tr. 8372, at 8-10) Since the witness believes that non-safety equipment was used only after improper operation of safety systems resulted in core damage, 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 outset of the accident before core damage occurred. (UCS PF 15) It was apparent that the witness had no basis 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 implies that "careful analysis" was done by the Staff to determine whether any non-safety grade equipment should 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, what GDC should be applied and which ignored in the partial upgrade? He stated that a "judgment had to be struck as to whether the additional reliability that might be gained by that was necessary." (Tr. 8613, Conran)

However, there is no indication that anyone on the Staff ever did the review necessary to 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, 8619-20, Conran) Mr. Conran knew of no such analysis. He testified that this is because of the "circumstances 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 analyses (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. Although it claims to have exercised judgment, the Staff is not in possession of the basis 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.

Once again, the Board nowhere discusses any of this evidence or suggests why it has rejected UCS PF 518-524, although these go to the heart of Mr. Conran's testimony.

Exception:

78. The Board erred in finding that partial upgrading of the PORV is a significant improvement to safety. (PID 992)

The pertinent portion of the PID at 992 implies that the partial upgrade of the PORV is a significant improvement to safety and applauds this approach. In fact, the witness did not know whether the reliability of the PORV had been significantly improved. (Tr. 8614, 8619-20, Conran; UCS PF 522-524) Nor did the witness know in what way the PORV is currently non-safety grade. He never looked at the current design because the Staff had already decided not to require full upgrading. (Tr. 8684-8687, Conran) This approach is curiously backward. The Staff could not possibly decide what measures were needed to improve the reliability of non-safety components such as the PORV without first trying to determine the ways in which they are vulnerable to failure.

Exception:

79. The Board erred in failing to find that instrumentation relied upon by the operator in order to determine the need for initiating or terminating safety systems should be safety grade.

See UCS PF 535-540. The evidence establishes that both the incore thermocouples and the pressurizer level instruments are relied upon to indicate to the operator when HPI can be throttled. (Tr. 7592, 7654) In the case of the incore thermocouples, no other instrumentation could be identified which the operator can use during a LOCA to determine the temperature in the downcomer. (Tr. 7623, Keaten) The pressurizer level instruments are used to tell the operator when HPI should be throttled to avoid exceeding the temperature-pressure limits on the reactor vessel (Tr. 7654, Keaten), a function which even the Licensee agreed is important to safety. (Tr. 7596, Keaten)

Thus, the TMI-1 operators are directed to perform important (or "critical") safety functions by relying upon non-safety grade instrumentation. Such

instrumentation should be safety-grade. Equipment used to detect the need for a safety function, to accomplish the safety function, and to confirm that the safety function has been accomplished is required to be safety grade if the safety function is performed automatically. There is no logical reason to relax or dispense with those requirements simply because the safety function is to be performed by the operator.

Exceptions:

83. The Board erred in failing to require systems interaction studies as a prerequisite for restart.

84. The Board erred in authorizing the operation of TMI-1 without even a commitment to perform plant-specific systems interaction studies.

These exceptions go to the remedy which is appropriate. The Board has now removed any requirement that systems interaction analyses be done for TMI-1. Memorandum and Order Modifying and Approving NRC Staff's Plan of Implementation, April 5, 1982, Sl. op. at 6-7.

We have argued above that the analyses of the TMI-2 accident clearly show that systems presently classified as non-safety, and hence receiving little or no NRC review, can cause accidents and be used to mitigate accidents in ways not originally considered in the plant's safety analysis. The present NRC classification system does not adequately recognize either of these kinds of effects that non-safety systems can have on the safety of the plant. (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 or plan whatever to take the first required step in this process - the undertaking of a comprehensive study to identify potential adverse systems interactions at TMI-1.

Even as to the non-safety equipment specifically involved in the TMI-2 accident (e.g. PORV, pressurizer level instruments) the Staff made only a hasty and ill-documented effort to determine whether and to what extent they need to be upgraded. Insufficient basis was presented to justify the Staff's decisions. (UCS PF 520-524)

It is simply unacceptable to acknowledge that an unresolved safety problem exists and then to act as if this plant can be operated without restriction having taken no steps nor even committed to any future steps directed toward resolving that problem.

There is a direct analogy between this situation and that presented in Virginia Electric Power Co. (North Anna Nuclear Power Station, Units 1 and 2), ALAB-491, 8 NRC 245 (1978). There, the unresolved safety safety issues in question were those identified by the ACRS and the Staff in its Task Action Plans. The Appeal Board stated:

> Of course, these 'unresolved' issues cannot be disregarded in individual licensing proceedings simply because they also have generic applicability; rather, for an applicant to succeed, there must be some explanation why construction or operation can proceed even though an overall solution has not been found.
>
> * * *
>
> Where operation of a facility is involved, similar analysis is necessary; but as to certain issues, the justification for giving an applicant the green light can obviously be more difficult to come by. For example, the reason often given for allowing construction activity is that there is still time to find a solution and build it into the plant's design. At the operating license stage, that reason is not available. But there may be one or more other justifications for permitting the plant to operate. The most common are that a solution satisfactory for the particular facility has been implemented, a restriction on the level or nature of operation adequate to eliminate the problem has been imposed, or the safety issue does not arise until the later years of plant operation. (8 NRC 245 at 248)

No such justification has been suggested sufficient to allow unrestricted operation of TMI-1 despite the existence of this safety problem. Nor, as the Appeal Board indicated, does the problem go away because it is generic. The

fact that other plants are subject to same problems and uncertainties is no reason to ignore the issue when it comes to the Board in a case within its jurisdiction.

It is also significant that the ACRS recommended "timely completion" of systems interaction studies for TMI-1, a recommendation that the Staff chose to reject without apparent justification.

TMI-1 should not be permitted to operate until the completion of a comprehensive engineering analysis which identifies potential adverse interaction between non-safety and safety systems and

1) non-safety systems which can cause or aggravate an accident are either upgraded or their potential adverse effects are effectively isolated from safety systems, and

2) non-safety components and systems (including instrumentation) which are called upon in the mitigation of accidents and transients are upgraded to safety grade.

I. Board Question 6

Exceptions 103-109

PID 1005-1067

UCS Proposed Findings 380-471

UCS Reply Findings 78-114

Summary

Board Question 6, subparts (a) - (k) were directed at determining whether the decay heat removal systems at TMI-1 are sufficiently reliable to permit restart (and long-term operation) without undue risk to public health and safety. UCS believes that the record shows that the TMI-1 emergency feedwater system ("EFW") is not a sufficiently reliable mode of decay heat removal. The

Board agreed that "the reliability of the EFW system has not been demonstrated to be adequate by itself." (PID 1050) However, the Board ruled further that, when EFW is backed up by the bleed-and-feed mode of cooling using the high pressure injection ("HPI") system, the two systems combined are adequately reliable.

The Board's reliance on bleed-and-feed is misplaced. This record does not permit a finding that bleed-and-feed can be relied upon. The Board essentially picked a number out of the air for the reliability of bleed-and-feed (PID 1056, 1065) that has no support whatever in the record. (See discussion in connection with exceptions 9-11, 15-19, supra, at 9-13, 18-24) Moreover, to use bleed-and-feed for decay heat removal transforms an anticipated operational occurrence, such as loss of main feedwater or loss of offsite electrical power, into a loss-of-coolant accident. This turns GDC 14 on its head. (UCS PF 434) In addition, the Board has permitted restart despite acknowledging that a single failure can cause loss of all feedwater to both steam generators. (PID 1060-1064)

Exception:

> 103. The Board erred in finding that automatic initiation of the emergency feedwater system will be safety-grade at restart for small break LOCA's and main feedwater transients. (PID 1057) 13/

The Board found that the automatic initiation of the EFW system will be safety grade at restart for small-break LOCA's and main feedwater transients. (PID 1036, 1057) This is a factual error. As the Board noted at PID 1028, the automatic initiation of the EFW pumps will be safety grade at restart. However, this is not true of the EFW flow control valves. The existing automatic

13/ The PID reference originally cited in UCS's exceptions was PID 1050. That was a typographical error.

circuits for opening, and subsequently controlling, the EFW flow control valves are derived from the non-safety grade Integrated Control System (PID 1031) The Board apparently overlooked the fact that the ICS both initiates opening of the EFW flow control valves and, thereafter, controls the flow rate. This latter function is clearly recognized as being performed by the non-safety grade ICS. (PID 1036) Thus, the automatic intiation of the EFW system is <u>not</u> safety grade for any accident.

The automatic circuits therefore do not meet the single failure criterion. (Staff Ex. 1, at C8-36) This violates the <u>short term</u> requirement that the automatic initiation circuits shall be designed so that a single failure will not result in the loss of emergency feedwater function. (Staff Ex. 1, at C8-34; NUREG-0578, at A-31; UCS PF 456) NUREG-0578 clearly provides at page A-31:

> 2. The automatic initiation signals and circuits shall be designed so that a single failure will not result in the loss of auxiliary feedwater system function.

<u>Exception:</u>

> 104. The Board erred in recommending restart without requiring design modifications to eliminate the hazard of a single failure causing iolation of all feedwater to both steam generators prior to restart. (PID 1064)

The Board found correctly that a single failure in the steam generator rupture detection system could cause loss of all feedwater to both steam generators. (PID 1060-1064) Its response was to direct the Licensee to propose a long-term "solution" and to have the staff "certify" to the Commission prior to restart that the Licensee has made "reasonable progress" in initiating its program for the long-term solution.

The single failure in question can cause the EFW system to completely lose its capability to perform as intended. This violates the same <u>short-term</u> requirement quoted in connection with the previous contention. The plant should therefore not be permitted to restart until this problem is corrected. The Board is not free to waive this requirement.

Exception:

107. The Board erred in relying on HPI in the feed-and bleed mode a compoensation for an inadequately reliable EFW system. (PID 1057, 1065)

UCS has extensively briefed the issues related to the propriety of reliance on the bleed-and-feed cooling mode to perform functions important to safety. It will not be repeated here, but is incorporated by reference. (See the discussion in connection with exceptions 9-11, 26-19, supra). Bleed-and-feed has assumed a role of central importance in this proceeding. It is relied on to compensate for unreliability in the non-safety equipment needed to maintain natural circulation (pressurizer heaters, UCS Contention 3), for the absence of a reliable method of providing forced cooling in the event of loss of natural circulation (UCS Contentions 1 and 2), and for the unreliability of the emergency feedwater system.

The Board make its "own estimate" (PID 1065) of the reliability of bleed-and-feed; it assumed a failure rate of 1x10-2 (1 in 100) per reactor year. (PID 1056) There is no support whatever in the record for this figure; it is literally picked from the air. The Licensee did no quantitative reliability analysis for feed-and bleed cooling. (Jones, ff. Tr. 4589 at 3) The Staff did not request nor did the Licensee provide it with either procedures or analyses for cooldown of the RCS by feed-and-bleed, nor did the Staff perform its own evaluations. (Wermiel et al., ff. Tr. 6035 at 6)

Neither did the Board provide an independent, technically supportable basis for its conclusion at PID 1056 that it can add a safety factor of 100 to the EFW reliability as credit for bleed-and-feed. It gives only two bases for its conclusion: 1) that the HPI sytem is safety-grade, and 2) that there has been "no demonstration" that the failure probabilities are "grossly in error." (PID 1056) The latter is no basis at all. The fact is that there is no evidence in the record of a failure rate for bleed-and-feed. This lack of evidence cannot

be used as support for a number selected by the Board. Morever, there was no opportunity to make a "demonstration" that the number is erroneous, since no party put it forth in the hearing.

The former statement assumes use of the safety valves for the bleeding function, (PID 1051) a function for which they are untested and unqualified. (See supra at p. 21-23) For the purpose of performing the bleeding function, the safety valves cannot be said to be safety-grade. (Id.) This record does not support a finding that the safety valves are highly reliable for the bleeding function.

Moreover, even if one were to assume that the HPI system for bleed-and-feed is safety-grade, that does not lead automatically to the conclusion that it has a failure rate as low as the Board assumes. Indeed, the EFW system being reviewed will also be safety-grade at some point in the future, yet as the Board finds, it will not even then be highly reliable. (PID 1050)

The Board's heavy reliance on bleed-and-feed is an error. We believe that the Appeal Board should also be aware of a document casting further serious doubt on the propriety of placing heavy reliance on the bleed-and-feed cooling mode. This document, SECY-81-513, "Plan for Early Resolution of Safety Issues," Enclosure 3, Appendix: Some Numerical Examples, is not part of the hearing record. It was not brought forward by the Staff and was only discovered by UCS in the past few weeks. We believe that it contains the type of information which the Staff should have brought to the attention of the Licensing Board under the direction of such cases as Duke Power Company (William B. McGuire Nuclear Station, Units 1 and 2), ALAB-143, 6 AEC 623 (1973) and Virginia Electric and Power Company (North Anna Power Station, Units 1 and 2), CLI-76-22, 4 NRC 480, 491 (1976). We do not assert misfeasance on the Staff's part; it is certainly possible that this information never came to the attention of the appropriate persons. However, if it had, it should have gone to the Board and would not now be outside the record.

The document, which is attached, presents the results of a reliability assessment for the Davis-Besse EFW System. It contains the following conclusion at page 4: "In particular, there is a large uncertainty in the probability of successful cooling using bleed-and-feed."

This must be read in context. TMI-1 does not have the same design as Davis-Besse. Therefore, the uncertainty in the viability of bleed-and-feed related to the low shut-off head of the HPI pumps is not applicable to TMI-1 (Id. at 3). However, the "considerable uncertainty in the viability of feed-and bleed..." related to the fact that "no thorough analysis has been performed of the use of the ECCS in this mode" is applicable to TMI-1. (Id.) It is clear in context that the probability of successful cooling using bleed-and-feed was analyzed independently from the plant-specific problem at Davis-Besse. That is, part of the analysis assumed that bleed-and-feed was established, but the probability of it successfully cooling the core was still very uncertain. The document is attached for the Board's attention.

Exception:

108. The Board erred in rejecting the lesson learned from TMI-2 that EFW systems shuld be adequate to remove decay heat without causing opening of reactor coolant system relief and/or safety valves. (PID 1026, rejecting UCS PF 390-92)

See UCS PF 390-392. The pertinent lesson learned requirement was that EFW initiation time and capacity and reactor scram time should be such that steam generator water level should remain high enough to remove decay heat "without causing opening of the primary coolant relief and code safety valves." (NUREG-0578 at A-30) The Board found that a valve will possibly open. (PID 1026) This is inconsistent with the requirement quoted above.

The Board discounts this on the basis that the overall improvements will "greatly reduce challenges to the relief valves and thereby satisfy the concerns..." contained in the pertinent lessons learned requirement. However,

this record clearly does not support a conclusion that challenges to the valves have been "greatly" reduced. All of the changes at restart will only increase the reliability of EFW from $8x10^{-3}$ to $2x10^{-3}$ (Wermiel/Curry Chart, PID P. 239). This is less than a factor of three for an analysis which has an inherent uncertainty of a factor of 10. (Tr. 16, 965, Curry)

Exception:

109. The Board erred in failing to give substantial weight to evidence showing that heat removal from the steam generators, which is necessary if EFW is to be effective in removing decay heat from the reactor coolant system, relies on non-safety-grade equipment. (PID 1023, 1024; compare with UCS PF 395)

In evaluating the EFW system, the Board failed to take into account that use of EFW 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.

In addition, 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)

The effectiveness of heat removal via the steam generators cannot be evaluated solely by considering the probability of delivering feedwater to the steam generators. One must also examine the oprobability of successfully removing heat - i.e. steam-from the steam generator. If the heat is not successfully removed from the steam generator, the temperature and pressure in the primary and secondary systems will increase until there is opening of the PORV and/or safety valves. In other words, delivery of water to the steam

generator is only one of the functions needed to successfully cool the primary system. The Board ignored the other necessary function of removing heat from the steam generator and thereby from the core. For this function, the system is non-safety grade.

J. Conclusion

For the above stated reasons, UCS's exceptipns should be sustained.

Respectfully submitted,

Ellyn R. Weiss

HARMON & WEISS
1725 I Street, N.W.
Suite 506
Washington, D.C. 20006
(202) 833-9070

Counsel for UCS

DATED: April 14, 1982

APPENDIX: SOME NUMERICAL EXAMPLES

1. Introduction

Several examples of the application of the proposed priority scheme are described below. The examples chosen were recently evaluated in the SPEB Branch. They are issues for which the estimates of frequency, consequences, and cost are new and which have not received extensive peer review and discussion.

The derivation of a priority score for any issue is of course only as "good" as the accuracy of the three most important parameters-frequency, consequences and costs. Estimates of these three parameters are necessarily rough, since firm data and/or sophisticated risk analyses are usually not available. Moreover, as the issue progresses toward resolution, additional information obtained during the course of the work will lead to updates of the estimates and thus to the priority score.

In general, factor-of-five estimates of the three parameters are desirable, but factor-of-ten estimates are sufficiently accurate to produce useful priority scores. Thus, estimates of the three parameters need not be very sophisticated.

The range estimates must reflect the uncertainty in the three parameters. When possible, statistically-derived limits (95% confidence) are used. However, it is usually necessary to use judgement to set these limits. In such cases, the objective is to set the limits widely enough to be reasonably certain of bounding the parameter.

2. BWR Scram Air Dump

This issue concerns the slow loss of control air pressure in the scram system of BWRs.[1,2] Air pressure dropping at a certain rate will first allow some of the CRD scram outlet valves to open slightly, thus filling the scram discharge volume with water, but allowing little or no control rod movement. Eventually the rods will try to scram, but the scram will be impaired. Meanwhile, the dropping air pressure can cause a transient (e.g., via feedwater controller lockup) which would normally call for a scram. One non-damaging event of this nature has already happened.

A proposed fix for this problem is an air header dump valve, which will force air pressure to drop rapidly once it reaches a critical setpoint.

Frequency is estimated as follows. One non-damaging event has happened in about 175 GE BWR-years of experience. If we assume that one tenth of these events lead to an ATWS scenario, the frequency becomes:

$$F = \frac{.1 \text{ events}}{175 \text{ r-y}} = 5.7 \times 10^{-4} \text{ events/r-y}$$

The estimation that one-tenth of these events could lead to an ATWS is based on the assumption that the true number is less than one, but probably more than one hundredth.

Likely consequences of such an event are those of a loss-of-feedwater ATWS, i.e., a core melt.[3] The release is on the order of WASH-1400 Release Category BWR-4:

$$R = 2.1 \times 10^{8} \text{ Ci/event}$$

Cost to a typical BWR plant owner is about one week's downtime to install the air header dump valve system.

$$I = 7 \text{ days} \times \$300{,}000\text{/day} = 2.1 \times 10^{6} \text{ dollars}$$

Nineteen BWRs have scram discharge volumes susceptible to this problem

$$N = 19$$

The cost to NRC is approximately one staff year. At $80,000 per staff year, this is:

$$C = 80{,}000 \text{ dollars}$$

These numbers are simply plugged into the formula:

$$S = \frac{N_{\Delta}[FR^{1.2}]}{C + NI}$$

$$S = \frac{(19)(5.7 \times 10^{-4})(2.1 \times 10^{8})^{1.2}}{(.08) + (19)(2.1)}$$

$$S = 3 \times 10^{6} \ \frac{\text{Ci}^{1.2} \text{ avoided per year}}{\text{million dollars spent}}$$

It is now necessary to estimate the range of uncertainty in this numerical score. To do this, we must estimate the uncertainty of each of the principal parameters: frequency, release, and costs. In practice, these uncertainty ranges are quite large, and are generally expressed at factors of 2, 3, 5 or 10.

In this example, the frequency F is composed of two components: the frequency of an air-failure-induced scram failure and the probability that this event will lead to the release of radioactivity. The former is based on operating experience. We estimate that this figure (5.7×10^{-3} /r-y) could be a factor of 10 higher or lower.

The probability of the event leading to a radioactivity-releasing ATWS could, in our judgement, be a factor of 5 higher (0.5) or a factor of 10 lower (0.01). Here, the asymmetry in range results partly from the fact that probabilities cannot exceed 1.0.

The release estimate could be off a factor of 5 in either direction, in our judgement.

The cost estimates (here dominated by licensee costs) could be a factor of 5 higher or a factor of 2 lower. Cost estimates are generally regarded as being more often too low than too high, and the range here reflects this.

Calculation of the range of the priority score, S, is now straightforward since all of the component factors for S are multiplicative and errors in these factors are independent of one another:

	Initiating event Frequency	Release Probability	Curies Released	Costs	S
High estimate	10	5	$5^{1.2}$	2	13
Low estimate	10	10	$5^{1.2}$	5	17

The factors for S are the square roots of the sum of the squares of the component factors, except that the factor for curies released must be raised to the 1.2 power first. Note that the low estimate for cost is used for the high estimate of S and vice versa; this is because costs appear in the denominator.

If S = 3 x 10 , we divide by 17 and multiply by 13 to get a range of:

$$2 \times 10^5 \quad \text{to} \quad 4 \times 10^7.$$

This range is an estimate based primarily on judgement, and should be considered as such. For a "true" score (a hypothetical figure based on exact values of the component frequencies, releases and costs) to lie outside this range, either an individual parameter would have to be out of its individual range or several paramenters would have to simultaneously be near the extremes of their ranges. It is possible, therefore, for the "true" score to lie outside this range. The range is only intended as a guideline. However, it is expected that most "true" scores would lie within the estimated range.

3. Diesel-driven Auxiliary Feed Pump at Davis-Besse

The auxiliary feedwater system at Davis-Besse has only steam-driven auxiliary feedwater pumps.[4] The safety concerns are (1) there is no diversity in the auxiliary feedwater system (other B&W plants also have a motor-driven auxiliary feedwater pump) and (2) if the steam generators dry out, no steam will be available to drive the auxiliary feed pump turbines. The only way to remove decay heat would then be using the ECCS in a feed-and-bleed mode of operation.

There is considerable uncertainty in the viability of feed-and-bleed operation at Davis-Besse since the high pressure safety injection pumps have a shut-off head below the safety valve and PORV setpoints, and no thorough analysis has been performed of the use of the ECCS in this mode.

In regard to steam generator dryout, four such events have occurred at B&W plants during a total B&W experience of about 41 reactor-years. For a dryout event to occur at Davis-Besse, a special automatic steam generator isolation on low steam line pressure would have to fail. In such a case, the licensee claims that a special small motor-driven pump could be aligned to pump water into the steam generators. This would not provide enough water to remove decay heat, but would generate enough steam to enable the auxiliary feedwater turbines to start.

An event tree was constructed which considered the frequency of initiating events that would require use of the auxiliary feedwater system, the ability to isolate the steam generators and use the special motor driven pump so that there would be steam available for the pumps, the reliability of the auxiliary feedwater pumps, and the probability of successful cooling using feed and bleed. The initiating event frequency was based on the actual frequency of the occurrence of steam generator dryout events at B&W plants, and on EPRI data for other events that resulted in the need for the auxiliary feed water system. The success/failure probabilities of the various sequences shown on the event tree were estimated based on a comparison of reliabilities reported in WASH-1400 for similar systems and components. Evaluation of the event tree indicated that the point value of the probability of the failure to adequately cool the core was 10^{-5} per reactor-year.

The consequences of such an event include a core melt but not direct containment failure. We approximate the release using WASH-1400 Release Category PWR-7:[3]

$$R = 2.1 \times 10^{6} \text{ Ci}$$

The proposed fix to reduce the probability of inadequate core cooling is to install a diesel-driven auxiliary feedwater pump at a cost of about $2,000,000. NRC cost is estimated to be 1/2 staff year, or $40,000. The rest of the calculation is straightforward.

$$S = \frac{N \Delta [FR^{1.2}]}{C + NI}$$

$$S = \frac{(1)(10^{-5})(2.1 \times 10^{6})^{1.2}}{(0.04) + (1)(2.0)}$$

$$S = 2 \times 10^{2}$$

As noted above, there is a considerable uncertainty in the value of some of the probabilities used to quantify the event tree. In particular, there is a large uncertainty in the probability of successful cooling using feed-and-bleed. Also the reliability of the systems that isolate the steam generator to maintain a source of steam to the steam driven auxiliary feedwater system is not known with a great deal of certainty. Bounding values of the success/failure probabilities of these two systems were estimated. The event tree was then quantified using these extreme values. This provided an estimated uncertainty band about the point value of the frequency of the failure to adequately cool the core. The band ranged from a factor of 10 low to a factor of 10 high.

The uncertainty in frequency was then combined with the estimated uncertainties in consequences and cost to determine the estimate range in priority score. The consequences were estimated to possibly be a factor of 5 higher or lower.

The actual costs were judged to be no more factor of two lower or a factor of 5 higher (factor of - 2 to + 5). These factors were combined, as described above, to yield a range of 2×10^{1} to 2×10^{3}.

4. Steam Generator Tube Rupture Detectors

This issue deals with the installation of special radioactivity detectors on PWR main steam lines.[5] They would be installed to provide additional diagnostic capability in the event of a steam generator tube rupture, enabling the operator to determine which steam generator is leaking. The operator could then isolate the leaking steam generator (after ramping down to no-load temperature) and cool down using only the intact steam generator(s). Note that the proposed fix does not change the frequency of the event. It reduces the magnitude of the radioactive release, i.e., mitigates the event.

The frequency of Steam Generator Tube Rupture is 10^{-2} per reactor-year. The release depends on the availability of offsite power (steam may be vented via atmospheric dump valves rather than by turbine bypass to condenser) and on the coolant activity (which can increase by a factor of 60 during an iodine spike). Details of this calculation can be found in Reference 5. The result is a point estimate of:

$$\Delta[FR^{1.2}] = 1.4\ Ci^{1.2}/r\text{-}y$$

There are 43 PWRs affected by this issue. The cost per plant is estimated at $600,000. The forward-looking cost to NRC is about two staff weeks per plant plus another two staff months of generic work, or about $150,000.

$$S = \frac{N\ \Delta[FR^{1.2}]}{C + NI}$$

$$S = \frac{(43)(1.4)}{(0.15) + (43)(0.6)}$$

$$S = 2 \times 10^{0}$$

Because the frequency and release calculations are rather lengthy (in Reference 5), the details of the range calculation will not be given here. It was assumed that the frequency range was plus or minus a factor of 5. The release (which was based on a detailed computer calculation) was estimated to be accurate to within a factor of 3. The cost was estimated to possibly be too low by a factor of 5 or too high by a factor of two. This gave overall uncertainty factors for S of - 8 and + 7. That is, the range would extend from one eighth of the point value to seven times the point value, or:

$$3 \times 10^{-1} \quad \text{to} \quad 2 \times 10^{1}.$$

BWR Misloaded Assembly

This issue deals with an unexpectedly high number of BWR fuel loading errors in recent operating experience.[6] Misloading of a BWR bundle, either by rotation or by loading into the wrong location, can result in the bundle being overdriven, possibly into departure from nucleate boiling (DNB).

The safety analysis for each BWR reload assumes that one assembly is misloaded. Thus, normal operation should not drive a misloaded bundle into DNB. However, the misloaded bundle can be driven into DNB in two ways:

a. The GEXL correlation assures that boiling transition will be avoided 99.9% of the time. Therefore, there is a probability of 0.001 that sustained DNB will occur for the worst-case misloaded assembly during normal operation, if the misloading ΔCPR is reasonably close to the limiting ΔCPR (and it usually is). With a 1/3 core reload, there is roughly a 1/3 chance of a given misloaded assembly being near a worst case during the course of a cycle.

   Ten fuel-loading-error (FLE) occurrences are presently on file for about 190 BWR-years experience. The frequency of bundle failures during steady-state operation is therefore:

   $$F_a = \frac{10 \text{ FLEs}}{190 \text{ r-y}} \times \frac{1 \text{ worst-case FLE}}{3 \text{ FLEs}} \times \frac{.001 \text{ DNB-induced failures}}{1 \text{ worst-case FLE}}$$

   $$F_a = 1.8 \times 10^{-5} \text{ bundle failures/r-y}$$

b. If a transient occurs, the core may be driven closer to DNB. The misloaded assembly starts out closer to DNB than assumed in the Safety Analyses, so it will be driven momentarily into DNB for worst-case misloadings.

   Although a reactor scram is not an unusual occurrence, significant transients (i.e., transients that bring the core near DNB limits) are rare. In the absence of firm data, we will assume that a significant transient occurs about 10 times in a 40-year plant life, based on engineering judgment.

   Studies have shown that momentary DNB usually does not cause fuel failure.[7] We will assume that one tenth of the transient DNB events result in fuel failure, based on judgment and the data in Reference 7.

   $$\text{Worst-case FLEs/cycle} = \frac{10 \text{ FLEs}}{190 \text{ r-y}} \times \frac{1.5 \text{ r-y}}{\text{fuel cycle}} \times \frac{1 \text{ worst case}}{3 \text{ FLEs}}$$

   $$= 2.6 \times 10^{-2}$$

   $$\text{significant transients/cycle} = \frac{10 \text{ transients}}{40 \text{ r-y}} \times \frac{1.5 \text{ r-y}}{\text{cycle}}$$

   $$= 3.8 \times 10^{-1}$$

A bundle failure requires a worst-case FLE and a significant transient in the same cycle

$$F_b = [(2.6 \times 10^{-2})(3.8 \times 10^{-1}) \frac{\text{DNB events}]}{\text{cycle}} \times \frac{1 \text{ cycle}}{1.5 \text{ r-y}} \times \frac{1 \text{ failure}}{10 \text{ DNB events}}$$

$$F_b = 6.6 \times 10^{-4}$$

Based on rod-drop-induced fuel failure calculations, one failed fuel bundle will release roughly 100 Ci to the environment.

The proposed fix is for licensees to pay stricter attention to core loading QA procedures. This should not involve more than one staff day per plant, since it is actually a return to commitments already in place, not a true upgrade of procedures.

$$I = 3.1 \times 10^{2} \text{ dollars} = 3.1 \times 10^{-4} \text{ million}$$

$$N = 24 \text{ reactors affected}$$

The anticipated cost to NRC is estimated at one staff week of generic effort plus one staff day per plant of enforcement activity.

$$C = 8.9 \times 10^{3} \text{ dollars} = 8.9 \times 10^{-3} \text{ million}$$

$$S = \frac{N\Delta[FR^{1.2}]}{C + NI}$$

$$S = \frac{(24)[(1.8 \times 10^{-5})(100)^{1.2} + (6.6 \times 10^{-4})(100)^{1.2}]}{8.9 \times 10 + (24)(3.1 \times 10)}$$

$$S = 3 \times 10^{2}$$

F_a is composed of the frequency of misloading (a factor of 2 was assumed), the probability of a worst case (factors of +2 and -5), and the number of failures per worst case (factor of 3). Again, these figures are based primarily on judgment.

In addition, F_b is composed of the frequency of significant transients (factor of 3), and failures per transient DNB (+2 and -5).

Release is judged to be within a factor of 5 and costs to be within +5 or -2. (Note that NRC costs are significant in this example). Since the two frequencies are additive, the range calculation is somewhat more complicated than a simple sum of squares. The results is a factor of +8 or -11 for S, giving a range of:

$$3 \times 10^{1} \text{ to } 2 \times 10^{3}$$

2. Recirculation Pump Seal Failure

This issue deals with an unexpectedly high rate of failures in PWR recirculation pump seals.[8] A seal failure results in a primary coolant leak, i.e., a very small LOCA.

The results reported in WASH-1400 indicated that a break in the reactor-coolant pressure boundary having an equivalent diameter in the range of 0.5 to 2 inches was a significant cause of a core melt. Since the current study shows that comparable break flow rates have resulted from RCP seal falures at a frequency about an order of magnitude greater than the pipe break frequency used in WASH-1400, the overall probability of core melt due to these small-size breaks could be dominated by events such as pump seal failures if the WASH-1400 assessment is correct. Using the current estimates of seal failure rates and WASH-1400 scenarios for core melts induced by small LOCAs, we estimate a core melt frequency of approximately 10^{-4} per reactor year.

For ranking purposes, we are interested primarily in the frequency of seal failures which result in the release of radioactivity. Seal failure is involved in many accident sequences, which lead to a spectrum of releases.

Therefore, to calculate $\Delta[FR^{1.2}]$, we must calculate the change of probability of each WASH-1400 PWR release category due to changing S2 (the very small LOCA probability) from 1.1×10^{-2} back to 10^{-3} per reactor-year. (10^{-3} /R-Y is the S2 probability presently assumed in WASH-1400. Consideration of seal failures adds 10^{-2} to this). The appropriate probability estimates are in Table V 3-14 of WASH-1400.[3] The results are:

Release Category	Probability	Curies Released	$\Delta F_n R_n^{1.2}$
PWR-1	1×10^{-6}	1.2×10^{9}	7.9×10^{4}
PWR-2	3×10^{-6}	9.3×10^{8}	1.7×10^{5}
PWR-3	3×10^{-5}	5.2×10^{8}	8.6×10^{5}
PWR-4	3×10^{-6}	2.8×10^{8}	4.1×10^{4}
PWR-5	3×10^{-6}	1.3×10^{8}	1.6×10^{4}
PWR-6	2×10^{-5}	1.0×10^{8}	1.9×10^{5}
PWR-7	2×10^{-4}	2.1×10^{6}	7.7×10^{3}

Total: 1.4×10^{6} $Ci^{1.2}$ /R-Y .

There are 43 PWRs affected by this problem. Unfortunately, the solution is not known. Possible solutions include special detectors that signal high leakage, more frequent seal replacement, new seal designs, and more smoothly running pumps that take longer to mechanically degrade the seals. In the absence of a firm solution, we choose the least expensive solution (detectors), which are estimated to cost $500,000 per reactor. This choice is not to be considered a recommendation for a particular technical solution. It is merely a first figure, appropriate for a temporary fix. The figure will be revised as more information becomes available.

NRC costs are estimated at perhaps four months of generic effort plus one staff week per reactor, or about $93,000.

The tentative priority score is now easily calculated:

$$S = \frac{N\Delta\,[FR^{1.2}]}{C + NI}$$

$$S = \frac{(43)(1.4 \times 10^6)}{(0.093) + (43)(0.5)}$$

$$S = 3 \times 10^6$$

It should be noted that both the seal failure rates and the likelihood of such small LOCAs leading to core melt according to WASH-1400 are conservatively estimated. These conservatisms are introduced in the analysis by (1) including seal leakage rates of 20 gallons per minute or greater in the failure data base, and (2) taking the conclusions of WASH-1400 regarding the probability of core melt given a small LOCA. The WASH-1400 estimate is believed to be conservative because the more probable sequences are not likely to proceed to core melt if credit is given for operator action or if fuel failure is realistically calculated. In any case, seal failures would still remain significant contributors to core melt if their frequency is not reduced.

Therefore, to calculate the range, we assume that the frequencies could be a factor of 5 higher or a factor of 10 lower. In addition, we assume that releases are within a factor of 5 and costs (which are very uncertain in this example) are within -2 to +20. The results are factors of -23 and +9, giving a range of:

1×10^5 to 3×10^7

7. PWR Diesel Generator Reliability

Diesel generators have not proved to be uniformly reliable. Although some installations have had very good experience, others have had a rather poor operational record. Because many safety-related systems need AC power to operate, and because offsite power is lost occasionally, poor diesel generator reliability is an important safety concern.[9]

Safety consequences were determined using Table V 3-14 of WASH-1400.[3] Loss-of-onsite-power appears in several sequences in Release Categories 1, 2 and 6. The calculated probabilities and consequences are as follows:

Release Category	PWR-1	PWR-2	PWR-6
Estimated Probability	3×10^{-8}	2.7×10^{-6}	6.1×10^{-7}
Curies Released	1.2×10^9	9.3×10^8	1.0×10^8

These numbers assume a diesel failure rate of 0.03 failures per start attempt for each of two diesels (no common mode failures). Field experience indicates an average failure rate of roughly 0.05 : failures per start attempt, with about 5% of these being common mode. (The numbers are also based one one plant design. Other plants may differ.)

It is estimated that this rate can be improved to .02 failures/attempted start. It is further estimated that about 22 plants will need to install the hardware modifications cited in Reference 9 to achieve this target rate.

Removing the assumption of 0.03 failures/attempt from the WASH-1400 figures, and inserting the new probabilities with a 5% common mode assumption, the results are:

Release Category	PWR-1	PWR-2	PWR-6
Present Probability	1.6×10^{-7}	1.4×10^{-5}	3.2×10^{-6}
Improved Probability	4.5×10^{-8}	4.1×10^{-6}	9.2×10^{-7}
Curies Released	1.2×10^{9}	9.3×10^{8}	1.0×10^{8}
$\Delta F_n R_n^{1.2}$	9.0×10^{3}	5.7×10^{5}	9.1×10^{3}

Total $\Delta F_n R_n^{1.2} = 5.9 \times 10^5$

It is estimated that the cost to NRC will be $400,000 and the cost to a licensee will be $830,000. The priority score is then:

$$S = \frac{N\Delta[FR^{1.2}]}{C + NI}$$

$$S = \frac{(22)(5.9 \times 10^5)}{(.4) + (22)(.83)}$$

$$S = 7 \times 10^5$$

To calculate the range, we assume a factor of two range in both the initial and final failure rates. The transient frequencies and radioactive releases are assumed to be within a factor of 5, and costs to be within +5 or -2. The result is a range of :

7×10^4 to 6×10^6

8. Boron Dilution

Many PWRs have no positive means of detecting boron dilution during cold shutdown.[10] Some operations carried out during outages (e.g., steam generator decontamination) reduce the RCS volume, thus speeding up dilution. Boron dilution has taken place during such operations, although thus far criticality has not occurred.[11]

The frequency is difficult to estimate since no actual radioactivity-releasing events have occurred in the 280 PWR-years of current experience. Moreover, the event under consideration occurs during shutdown conditions, for which no appropriate event tree studies exist.

An upper limit on frequency can be calculated based on statistical considerations. If we assume a Poisson process, having zero occurrences in the 280 reactor-years implies that, to 95% probability.

$$F \leq 1.1 \times 10^{-2} \text{ events/R-Y}$$

To get a lower bound, we make the analogous pragmatic assumption (based also on our judgement) that the probability of one event in the same period is at least 5%. This implies that:

$$F \geq 1.9 \times 10^{-4} \text{ events/R-Y}$$

A "best estimate" value is needed for the priority score formula. The statistical methods above cannot give this, so we take the pragmatic approach of using the geometric mean of the two limits:

$$F = 1.4 \times 10^{-3} \text{ events/R-Y}$$

criticality caused by boron dilution during shutdown (with all rods in) .ll take time to reverse. We make the reasonable but purely judgmental assumption that 1% of the fuel pins release gap activity. If 180 assemblies are present, 1% of the total gap activity would be 4.3×10^5 Ci. Since the reactor head is removed, about 10% of this activity would escape from the containment ventilating system. (See dropped fuel assembly accident analysis, Surry FSAR.)

$$R = 4.3 \times 10^4 \text{ Ci}$$

All 43 operating PWRs are affected by this condition. The fix is to install instrumentation to detect the event and stop the dilution, either automatically or, if the detection is sufficiently early, by alerting the operator. Cost to the licensee is estimated to be $200,000. NRC costs are estimated to be about one month of generic effort plus one week per reactor, or $73,000. The priority score is:

$$S = \frac{N \Delta [FR^{1.2}]}{C + NI}$$

$$S = \frac{(43)(1.4 \times 10^{-3})(4.3 \times 10^4)^{1.2}}{(.073) + (43)(.2)}$$

$$S = 3 \times 10^3$$

.o estimate the frequency range, we use the 95% statistical bounds given above. The release is also quite uncertain, and a factor of 10 is used here Costs are assumed to be within -2 to +5. The resulting range is:

$$2 \times 10^2 \text{ to } 5 \times 10^4$$

PWR Resistance Temperature Detector Time Responses

Some of the inputs to PWR scram systems are derived from resistance temperature detectors (RTDs). Operational experience with these detectors has shown that the thermal time constant has degraded in service by as much as a factor of three.[12] Thus, scram time could be delayed over that assumed in the safety analysis.

Coolant temperatures are only one input to the scram system. They provide the "first in" signal only during boron dilution transients and certain control rod bank withdrawal transients.

The average delay observed is about 0.86 seconds.[12] We estimate that this would cause a minimum departure from nuclear boiling ratio (DNBR) of 1.285 during either of the two transients mentioned above. A simplified hand calculation indicates that such an event would release roughly five curies to the environment.

Based on actual experience, the frequency of boron dilution events is 0.03 per R-Y, and the frequency of uncontrolled rod withdrawal events is 0.01 per R-Y.[13]

There are 43 reactors affected by this condition. Specialized equipment is commercially available to fix this problem, at a total cost of $26,000 per installation.

The NRC cost is estimated to be one week of generic effort, plus two days per plant, totalling $29,000. The priority score is:

$$S = \frac{N\Delta[FR^{1.2}]}{C + NI}$$

$$S = \frac{(43)(.03 + .01)(5)^{1.2}}{(.029) + (43)(.026)}$$

$$S = 1 \times 10^{1}$$

To calculate a range, a factor of 5 was assumed for frequency. For the release, factors of +5 and -10 were assumed since the number of pins failing due to such a small excursion into transition boiling may be quite small.

Costs in this example are more accurate than for the other cases treated previously since there is some plant experience to draw on. A factor of two was estimated. The resulting range is:

6×10^{-1} to 7×10^{1}

10. References

All of the information cited in this section is available for examination and copying for a fee at the NRC Public Document Room, 1717 H Street, N. W., Washington, D. C. 20555.

1. Memo, P. S. Check to G. C. Lainas, T. H. Novak, and R. L. Tedesco, December 1, 1980.

2. Memo, M. H. Ernst to H. R. Denton thru T. E. Murley, December 8, 1980.

3. "Reactor Safety Study, An Assessment of Accident Risks in U. S. Commercial Nuclear Power Plants," WASH-1400 (NUREG-75/014), October 1975.

4. Memo, D. Garner to T. M. Novak, September 11, 1980.

5. Memo, R. Lobel and H. VanderMolen to R. Baer, January 7, 1981.

6. Memo, T. Murley to D. Eisenhut, January 7, 1981.

7. "Fuel Rod Behavior During Tests PCM 8-1 RS, CHF Scoping, and PCM 8-1 RF," NUREG/CR-1715, November 1980.

8. Memo, A. C. Thadani to R. Baer, December 12, 1980.

9. Memo, D. Eisenhut to F. Schroeder, September 25, 1980.

10. Memo, T. Novak to F. Schroeder, December 12, 1980.

11. IE Information Notice No. 80-34: "Boron Dilution of Reactor Coolant During Steam Generator Decontamination," September 26, 1980.

12. Memo, P. S. Check to F. Schroeder, December 3, 1980.

13. "ATWS: A Reappraisal, Part III, "Frequency of Anticipated Transients," EPRI NP-801, July 1978.

TABLE 1. EXAMPLES OF ESTABLISHING PRIORITIES

	NUMBER REACTORS	NRC COST (THOUSANDS)	COST PER LICENSEE (THOUSANDS)	WEIGHTED SAFETY BENEFIT	EFFECTIVENESS OF FIX	MGT SCORE	PRIORITY SCORE	RANGE
BWR SCRAM AIR DUMP	19	80	2100	5.5×10^6	98%	1×10^3	3×10^6	2×10^5 - $4 \times 10^{[illegible]}$
PWR RCP EAL LEAKAGE	43	93	500	1.4×10^6	90%	6×10^2	3×10^6	1×10^5 - 3×10^7
PWR DIESEL ERATOR RELIABILITY	22	400	830	5.9×10^5	50%	3×10^1	7×10^5	7×10^4 - 6×10^6
BORON DILUTION	43	73	200	5.1×10^2	80%	4×10^1	3×10^3	2×10^2 - 5×10^4
BWR FUEL MISLOADING	24	8.9	0.3	1.7×10^{-1}	80%	5×10^{-4}	3×10^2	3×10^1 - 2×10^3
DAVIS-BESSE DIESEL-DRIVEN AUX. FEED	1	40	2000	7.7×10^2	95%	1×10^{-2}	2×10^2	2×10^1 - $2 \times 10^{[illegible]}$
PWR RTD IME RESPONSE	43	29	26	2.8×10^{-1}	95%	4×10^{-4}	1×10^1	6×10^{-1} - 7×10
TEAM GENERATOR TUBE RUPTURE DETECTORS	43	150	600	1.4×10^0	55%	4×10^{-4}	2×10^0	3×10^{-1} - 2×10^1

TABLE 2. EFFECT OF WEIGHTING FACTORS

	MGT Score $Ci^{1.2}$/yr/$	Priority Score $Ci^{1.2}$/yr/Million	Unweighted MGT Score Ci/yr/$	Unweighted Priority Score Ci/yr/Million
BWR Scram Air Dump	1×10^{3}	3×10^{6}	3×10^{1}	6×10^{4}
PWR RCP Seal Leakage	6×10^{2}	3×10^{6}	1×10^{1}	5×10^{4}
PWR Diesel Generator Reliability	3×10^{1}	7×10^{5}	5×10^{-1}	1×10^{4}
Boron Dilution	4×10^{1}	3×10^{3}	4×10^{-2}	3×10^{2}
BWR Fuel Misloading	5×10^{-4}	3×10^{2}	2×10^{-4}	1×10^{2}
Davis-Besse Diesel-Driven Aux. Feed	1×10^{-2}	2×10^{2}	5×10^{-4}	1×10^{1}
PWR RTD Time Response	4×10^{-4}	1×10^{1}	3×10^{-4}	8×10^{0}
Steam Generator Tube Rupture Detectors	4×10^{-4}	2×10^{0}	2×10^{-4}	1×10^{0}

TABLE OF CONTENTS

TABLE OF AUTHORITIES

Statutes

Regulations

DOCKETED
USNRC

'82 APR 14 P4:18

OFFICE OF SECRETARY
DOCKETING & SERVICE
BRANCH

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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 EXCEPTIONS TO THE PARTIAL INITIAL DECISION OF DECEMBER 14, 1981 - PART 2" have been served on the following persons by deposit in the United States mail, first class postage prepaid, this 14th day of April 1982.

* Nunzio Palladino, Chairman
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

* Victor Gilinsky, Commissioner
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

* John Ahearne, Commissioner
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

* Thomas Roberts, Commissioner
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

** Ivan W. Smith, Chairman
Atomic Safety and Licensing Board Panel
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dr. Walter H. Jordan
Atomic Safety and Licensing Board Panel
881 West Outer Drive
Oak Ridge, Tennessee 37830

Dr. Linda W. Little
Atomic Safety and Licensing Board Panel
5000 Hermitage Drive
Raleigh, North Carolina 27612

Professor Gary L. Milhollin
1815 Jefferson Street
Madison, Wisconsin 53711

** Judge Gary J. Edles, Chairman
Atomic Safety and Licensing Appeal Board
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

** Judge John H. Buck
Atomic Safety and Licensing Appeal Board Panel
U. S. Nuclear Regulatory Commission
Washington D.C. 20555

** Judge Christine N. Kohl
Atomic Safety and Licensing Appeal Board Panel
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

** Judge Reginald L. Gotchy
Atomic Safety and Licensing
Appeal Board Panel
U. S. Nuclear Regulatory
Commission
Washington D.C. 20555

Mrs. Marjorie Aamodt
R.D. #5
Coatesville, PA 19320

Robert Adler, Esq.
Assistant Attorney General
505 Executive House
P.O. Box 2357
Harrisburg, Pennsylvania 17120

Louise Bradford
Three Mile Island Alert
325 Peffer Street
Harrisburg, Pennsylvania 17102

Walter W. Cohen, Esq.
Office of Consumer Advocate
1425 Strawberry Square
Harrisburg, Pennsylvania 17127

Jordan D. Cunningham, Esq.
Fox, Farr & Cunningham
2320 North Second Street
Harrisburg, Pennsylvania 17110

Thomas J. Germaine, Esq.
Deputy Attorney General
Division of Law - Room 316
1100 Raymond Boulevard
Newark, New Jersey 07102

Dr. Judith H. Johnsrud
Dr. Chauncey Kepford
Environmental Coalition on
Nuclear Power
433 Orlando Averue
State College, PA 16801

**** William S. Jordan, III
Harmon & Weiss
1725 Eye St., N.W., Suite 506
Washington, D.C. 20006

John A. Levin, Esq.
Assistant Counsel
Pennsylvania Public Utility
Commission
P.O. Box 3265
Harrisburg, Pennsylvania 17120

Mr. Marvin I. Lewis
6504 Bradford Terrace
Philadelphia, PA 19149

Ms. Gail B. Phelps
245 West Philadelphia Street
York, Pennsylvania 17404

Mr. Robert Q. Pollard
609 Montpelier Street
Baltimore, Maryland 21218

**** Mr.Steven C. Sholly
Union of Concerned Scientists
1346 Connecticut Ave., NW
Washington, DC 20036

*** Counsel for NRC Staff
Office of Executive Legal
Director
U. S. Nuclear Regulatory
Commission
Washington, D.C. 20555

**** George F. Trowbridge, Esq.
Shaw, Pittman, Potts &
Trowbridge
1800 M Street, N.W.
Washington, D.C. 20036

* Docketing and Service Section
Office of the Secretary
U. S. Nuclear Regulatory
Commission
Washington, D.C. 20555

Robert Q. Pollard

* Hand delivered to 1717 H St., NW, Washington, D.C.

** Hand delivered to 4350 East-West Highway, Bethesda, MD

*** Hand delivered to Maryland National Bank Bldg., Bethesda, MD

**** Hand delivered to indicated address.