UNITED STATES OF AMERICS
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

UNION OF CONCERNED SCIENTISTS
REPLY FINDINGS ON
UCS CONTENTIONS 1,2,3, AND 5

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UCS REPLY TO LICENSEE'S INTRODUCTION, PARAGRAPH 3

1. The Licensee raises here for the first time an issue which it attempted to pursue throughout the hearing - that TMI-1 is being treated differently than other reactors. It implies that because some of the issues raised by Intervenors admitted to this proceeding do not relate to the "unique concerns" for TMI-1 identified at CLI-79-8, 10 NRC 141, 143-144, they should somehow be accorded less weight by this Board, or that TMI-1 should be permitted to operate if other plants are operating, without respect to what the record in this case may show with regard to TMI-1. We reject such reasoning.

2. On July 2, 1979, the Commission suspended the operating license for TMI-1, finding that the Commission presently lacks the requisite reasonable assurance that the Licensee's Three Mile Island Unit 1 Facility can be operated without endangering the health and safety of the public."

3. On August 9, 1979, the Commission reaffirmed that the above finding "remains valid" CLI-79-8, 10 NRC 141, 142, elaborated on its reasons and set up this proceeding. The bases for the Commission's conclusion that it did not have reasonable assurance that TMI-1 can safely operate, derived from "its evaluation to date" of the "TMI-2 accident." (Id.) They related to a series of factors causing the sensitivity

of the B&W design to transients originating in the secondary system which cause B&W reactors to "place more reliance" on the reliability and performance characteristic of auxiliary (emergency) feedwater and on ECCS to recover from frequent anticipated transients such as loss of offsite power and loss of main feedwater than other PWR designs. This, in turn, places a large burden on plant operators in the event of off-normal system behavior during such transients. (Id. at 142-143)

4. The Commission went on to note that, in addition to the problems common to B&W reactors, certain "unique circumstances" at TMI-1 require the resolution of additional safety concerns prior to allowing the plant to resume operation. These related to 1) potential interactions between TMI-2 and TMI-1 2) questions about the management and technical capability of the Licensee 3) decontamination of TMI-2 4) recognized deficiencies in TMI emergency plans and operator procedures. (Id. at 143-144)

5. The Commission proceeded to establish this proceeding and to order that the issues to be heard should include whether the short and long-term measures identified by the NRC Staff to date are "necessary and sufficient" to provide reasonable assurance that TMI-1 can restart and operate in the long-term without endangering the health and safety of the public. (Id. at 148)

6. Thus, the Commission made no distinction among issues relating to public health and safety on the basis of whether they are unique to TMI-1. Nor did it suggest that the required measures identified at the time of issuance of CLI-79-8 represented a complete list. That is precisely the issue to be resolved by this Board.

7. The Licensee attempted at the outset of this proceeding to convince the Board that it could consider only the individual factual issues expressly mentioned in CLI-79-8 or the documents specifically referenced therein. We rejected that argument. LBP-79-34, 10 NRC 282, 829-831. It has become no more valid by the passage of time.

8. In essence, this proceeding resembles other hearings on individual nuclear plants except that the scope of the issues is limited to those which have a nexus to the TMI-2 accident. We reviewed all contentions and rejected many which did not have such a nexus. Once an issue has passed this test and has fairly been raised by a party, it is this Board's responsibility to resolve that issue for TMI-1. A Board sitting for an operating license or construction permit proceeding could not allow a plant to operate on the grounds that there may be other plants operating with the same safety problem. This Board is no more free to allow TMI-1 to operate in the face of safety problems identified on this

record on the ground that other plants may also exhibit these safety problems.

9. It is worth observing that such a practice would have the effect of precluding consideration of any safety problem if an applicant or licensee could show that it existed in another nuclear plant. This result is both absurd on its face and clearly counterproductive to the goal of protecting public health and safety. It is similar to the now-discredited and prohibited staff practice of licensing plants without considering the so-called generic safety problems. Virginia Electric and Power Co. (North Anna Nuclear Power Station, Units 1 and 2), ALAB-491, 8 NRC 245, 248 (1975):

> Of course, these 'unresolved' issues cannot be disregarded in individual licensing proceedings simply because they also have generic applicability....

UCS CONTENTIONS 1 and 2

REPLY TO LICENSEE'S PROPOSED FINDINGS, PARAGRAPH NO. 7

10. The statement that "Forced cooling is not needed to establish [single phase] natural circulation," is lifted out of context without proper qualification.

11. The witness was responding to a question as to whether there are some conditions under which one cannot get into natural circulation without forced cooling. The full answer given by the witness was that forced cooling is not needed to reestablish natural circulation, although that was the method utilized during the TMI-2 accident, because the high point vents could be used to bleed out the gas or steam bubble to reestablish natural circulation. (Tr. 4623-24, Jones)

12. The same answer was given more succinctly earlier by the same witness:

"Q. Assuming, then, that a void such as the one you have described exists in the U-bend, is there any way to get natural circulation going?

A. Well, you could get natural circulation going by

a bump* with the reactor coolant pumps, which was performed at Three Mile Island, or you could get it established by opening the high point vents that are being installed in the hot legs at TMI 1."

(Tr. 4167, Jones)

13. The TMI-2 accident demonstrated that single-phase natural circulation cannot be established in the presence of significant voiding. Voiding is predicted to interrupt natural circulation in the majority of small-break LOCA cases analyzed. (Licensee's Proposed Finding Nos. 10 and 17)

14. The high point vents on the reactor coolant system hot legs have not been installed and are not scheduled to be installed until July 1, 1982, at the earliest. (UCS' Proposed Finding No. 19) Thus, contrary to the Licensee's Proposed Finding No. 7, the record in this proceeding shows that forced cooling is the only method available at TMI-1 to establish or reestablish single-phase natural circulation

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* "Bumping the reactor coolant pumps basically just means a turning on of the pumps and a quick turning off in a short time period, something on the order of ten or fifteen seconds. We would consider that a pump bump. That is, instead of completely starting the pump and allowing it to circulate, you get it up to speed to get some circulation established, and you turn it off, which is in effect like a big bump on the system - or shove." (Tr. 4622, Jones)

for the majority of small-break LOCA cases. However, the reactor coolant pumps cannot be run or even "bumped" unless offsite electrical power is available (Tr. 4654, Keaten) and the loss of offsite power is a condition required to be postulated by GDC-17 and GDC-34.

15. Therefore, we find that single-phase natural circulation cannot be relied on for decay heat removal for the majority of accidents in the small break LOCA spectrum.

REPLY TO LICENSEE'S PROPOSED FINDINGS, PARAGRAPH NO. 8

16. The Licensee refers to a "wide range of plant conditions" over which analyses and testing have allegedly demonstrated that (single-phase) natural circulation can be initiated and maintained. It should be noted that these statements refer not to a small-break LOCA, but to loss of all reactor coolant pumps.

17. There may be a wide range of pressure and temperature over which natural circulation can be initiated and maintained, but in all instances, the necessary condition is that the reactor coolant must remain subcooled to prevent voiding sufficient to interrupt natural circulation. (See UCS' Proposed Finding No. 24)

18. Cross examination by UCS established that testing of natural circulation did not encompass the conditions that prevailed during the TMI-2 accident. (See UCS' Proposed Finding Nos. 26 and 27) Furthermore, for the majority of accidents in the small-break LOCA spectrum, Licensee's analyses predict voiding in the reactor coolant system such that natural circulation cannot be maintained throughout the accident. (Licensee's Proposed Finding No. 10)

19. Thus, the wide range of conditions referred to in Licensee's Proposed Finding No. 8 have little or no relevance to accidents with a close nexus to the TMI-2 accident which are the subject of this proceeding.

REPLY TO LICENSEE PROPOSED FINDINGS, PARAGRAPH NO. 15

20. The record does not support the proposed finding that Licensee witnesses Keaten and Jones are obviously familiar with the accident at TMI-2.

21. Mr. Keaten's testimony was that Licensee's attention focussed on the earlier portion of the accident - about the first hour and forty minutes. (Tr. 4605, 4610, Keaten) He was unaware of the extent to which attempts were made to start the reactor pumps. (Tr. 4611, Keaten) The extent of Mr. Keaten's testimony on details of the accident sequence

was merely to confirm that UCS Ex. 1 indicated certain events but that he had not reviewed it in detail. (Tr. 4612-13, Keaten)

22. Mr. Jones refers to the PORV block valve being located downstream of the PORV. (Tr. 4799, 4877, Jones) The block valve is upstream of the PORV between the pressurizer and the PORV. (Lic. Ex. 1, at Figure 302-650, Rev. 18, Zone B3) Mr. Jones also relied on what he had heard concerning operator actions during the accident but did not know the basis for the operator decisions. (Tr. 4650, Jones) Mr. Jones is not an expert on the TMI-2 accident sequence.

23. The Licensee also includes a statement in its Proposed Finding No. 15 that after adequate injection flow was restored, and subsequent to the core damage, the core was effectively cooled even though natural circulation was not occurring in the primary system. The witness actually testified that the first time in the accident sequence that he <u>knew</u> adequate core cooling was established is when a reactor coolant pump was restarted at about 16 hours. (Tr. 4655, Jones)

REPLY TO LICENSEE'S PROPOSED FINDINGS, PARAGRAPH NO. 16

24. Operation of the reactor coolant pumps during the TMI-2 accident was not limited to the period from approximately

16 hours after the start of the accident to approximately one month after the accident. To the extent that the Licensee's findings imply that this is so, they are inaccurate.

25. In fact, all four reactor coolant pumps operated for more than one hour after the start of the accident, the period when decay heat was at its highest level. One reactor coolant pump was operated for about 15 minutes at approximately 3 hours after the accident began. Another pump was "bumped" at approximately 4 hours ten minutes and again at approximately 15 hours thirty minutes after the accident began. (UCS Ex. 1)

REPLY TO LICENSEE'S PROPOSED FINDINGS, PARAGRAPH NO. 22

26. There is no evidence on the record of what will be installed to reduce radiation levels outside containment. At the least, the Board must condition its decision to requiring verification by the Staff of the installation and adequacy of the new shielding.

27. The Board should also address the adequacy of the radiation shielding outside containment should the letdown line be used during an accident as is specified in the TMI-1 emergency procedure for inadequate core cooling. (Lic. Ex. 51, at 4.0, 5.0) Unfortunately, there is nothing in the record to indicate that either the Licensee or the Staff

assessed this aspect of the radiation shielding problem. In fact, the Restart Report states that the shielding review was predicted (sic) (predicated?) on the assumption that letdown of reactor coolant outside containment will not be employed when coolant activity is at unsatisfactory levels. The Restart Report also states that letdown will be automatically terminated and will not be re-established if activity levels are unacceptably high. Earlier versions of the TMI-1 emergency procedure contained such a caution but the current version does not. (Compare UCS Ex. 4, at 6.0 with UCS Ex. 51, at 4.0, 5.0)

28. Furthermore, the Restart Report relies on letdown through the RCS high point vents which have not been installed to compensate for the prohibition against letdown via the normal path to outside containment. (Lic. Ex. 1, at 2.1-38e, Am. 25)

29. In sum, the radiation shielding assessment required by Section 2.1.6.b of NUREG-0578 was based in part on the abandoned commitment to install the RCS high point vents prior to restart and the former prohibition against use of the normal letdown path. Therefore, the shielding review now on the record cannot be used as the basis for an assertion that TMI-1 is safe enough to restart.

REPLY TO STAFF'S PROPOSED FINDING, PARAGRAPH NO. 11

30. To the Staff's proposed finding should be added the following:

31. The Licensee's witnesses testified that the first time following the start of the accident when adequate core cooling is known to have been established is at about 16 hours when a reactor coolant pump was restarted. (Tr. 4655, Jones)

32. The witnesses also testified that for about the first three days following restart of one reactor coolant pump, natural circulation might not have been established if the pump had stopped because of the amount of noncondensible gas in the primary system. (Tr 4654-55, Keaten)

REPLY TO STAFF'S PROPOSED FINDINGS, PARAGRAPH NO. 12

33. The staff asserts that further evaluation found that there was less risk in operating in the natural circulation mode than with the reactor coolant pumps running. The appropriate qualifications for that statement are: 1) the primary system was taken solid (i.e., no steam bubble in the pressurizer) because of action taken following failure of the pressurizer level instruments and 2) the time period that the statement applies to began about one month after the start accident and thus the decay heat rate was substantially lower than

earlier in the accident.

REPLY TO STAFF'S PROPOSED FINDINGS, PARAGRAPHS NOS. 13 and 14

34. The Staff asserts that natural circulation can be established or reestablished either by "bumping" a reactor coolant pump or by opening the RCS high point vents. This assertion is not supported by the record.

35. The RCPs cannot be operated or even "bumped" in the event of loss of offsite power and the RCS high point vents have not been installed. (See Reply to Licensee's Proposed Finding No. 7, supra)

REPLY TO STAFF'S PROPOSED FINDINGS, PARAGRAPHS NOS. 16 and 17

36. The Staff refers to Mr. Jones testimony that during the TMI-2 accident sequence there were time periods during which the core was adequately cooled without the RCP running.

37. It is not obvious why the Staff cites this testimony since on its face it means that there were also time periods when the core was <u>not</u> adequately cooled when the reactor coolant pumps were not operating. In fact, the first period of core damage occurred shortly after the second two reactor coolant pumps were shut off and was terminated in part by

restarting a reactor coolant pump. (Tr. 4678, 4681, Jones)

38. In any event, the testimony cited by the Staff cannot be accorded any weight. The basis for the witness' "belief" that the core was adequately cooled with the RCP was that there was "no substantial increases in radiation which were indicative of substantially increasing damage within the core." (Tr. 4655, Jones, emphasis added) There is no evidence on the record to identify the radiation detectors providing the information the witness relied upon. There is no evidence to determine whether those detectors were capable of detecting a substantial increase in radiation and whether those detectors were qualified to operate accurately in the accident environment.

39. We therefore rely on the testimony of the witness that the first time in the accident sequence when he knows that the core was adequately cooled is after a reactor coolant pump was restarted at about 16 hours in the accident sequence. (Tr. 4655, Jones)

REPLY TO STAFF'S PROPOSED FINDINGS, PARAGRAPH NO. 18

40. It is unstated in the Staff's proposed finding, but it is important to recall that the analyses, testing, and unplanned occurrences alleged to verify natural circulation

capability of the TMI-1 design refer solely to single-phase natural circulation. (Tr. 4684-85, Jones)

41. The testing of natural circulation capability involved eight to ten tests conducted prior to the TMI-2 accident. None of the tests involved sufficient voiding to interrupt single-phase natural circulation, although the record is clear that for the majority of small break LOCA's, sufficient voiding will occur to interrupt natural circulation. None of the tests simulated flow blockage. None of the tests involved opening the PORV or in any other way simulated a small break LOCA. At least one of the tests was conducted at a raised loop plant; TMI-1 is a lowered loop plant. (Tr. 4702-03, Jones)

42. The Staff cites four unplanned occurrences alleged to demonstrate the natural circulation capability of the TMI-1 design. Only two of those are applicable to the stated purpose. The Davis-Besse occurrence is not applicable because it is a raised loop plant. (Tr. 4704, Jones) The Crystal River event was not an occurrence predominately involving natural circulation. The witness described the early part of the sequence as essentially a feed and bleed cooling mode.* Later in the sequence, after electrical

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* We do not agree that this event demonstrated the adequacy of the feed and bleed cooling mode as discussed elsewhere in this decision. (UCS Proposed Finding No. 31)

power and feedwater were restored, HPI was throttled, a bubble was reestablished in the pressurizer, the reactor coolant pumps were restarted and a normal cool-down was conducted. While natural circulation may have occurred for some portion of the sequence, it certainly cannot be claimed to have demonstrated the natural circulation capability of the TMI-1 design to any useful degree.

43. Finally, the Staff cites testimony by Licensee's witness Keaten for the apparent purpose of attempting to establish that although the PORV opened during the unplanned occurrence at Crystal River in February 1980, the PORV would not open during a similar occurrence at TMI-1. The Staff fails in that attempt.

44. The initiating event at Crystal River was a loss of the non-nuclear instrumentation power supply. The PORV was opened as a <u>result</u> of that loss of power. (Tr. 4705-06, Jones) The testimony by Mr. Jones was <u>not</u> that power was lost to the PORV nor that the PORV at Crystal River fails open upon loss of its power supply. Mr. Jones was very precise. Power was lost to the non-nuclear instrumentation and that power loss resulted in the opening of the PORV.

45. When specifically asked whether that was unique to Crystal River - whether the PORV normally opens on loss of power, Mr. Jones apparently interpreted the question to

apply to loss of power to the non-nuclear instrumentation, the situation he was discussing. He could not answer the question because of his unfamiliarity with the integrated control system design. Whether the PORV opens as a result of the loss of the non-nuclear instrumentation power supply depends on the specific design of the integrated control system. (Tr. 4707, Jones)

46. Mr. Keaten's testimony, which is cited by the Staff, clearly pertains to an entirely different situation - loss of power to the PORV itself.

47. At TMI-1, the PORV is powered from a 250-volt direct current power source. The TMI-1 integrated control system and non-nuclear instrumentation are powered from a 120-volt alternating current power supply. (Lic. Ex. 1, at 2.1-7c, paragraphs 2.1.1.3.2 and 2.1.1.3.4) Thus, a loss of power to the non-nuclear instrumentation at TMI-1 does not necessarily result in loss of power to the PORV and may result in opening the PORV at TMI-1, just as occurred at Crystal River.

REPLY TO STAFF'S PROPOSED FINDINGS, PARAGRAPH NO. 22

48. The Staff alleges that for small break LOCA's, voiding would only "temporarily" block natural circulation because two-phase natural circulation would be established. This

is not supported by the record.

49. For the boiler-condenser or two-phase mode of natural circulation to be effective, the primary water level must be lower than the secondary water level in order to expose a condensing surface in the steam generator for the steam in the primary system. During the TMI-2 accident, the boiler-condenser mode was not established because the high pressure injection system was used to refill the primary system. This blocked the heat removal surface in the steam generator and trapped a void up in the 180° bend of the primary system hot legs. (Tr. 4616, Jones)

50. TMI-1 emergency procedures instruct the operator not to terminate high pressure injection flow until the reactor system becomes subcooled. (Tr. 4968, Jensen) The majority of small-break LOCA's will result in sufficient voiding to interrupt single-phase natural circulation. (Licensee's proposed finding No. 10) Thus, even if the primary water level drops below the secondary level where EFW flow enters the steam generator, continued operation of the HPI pumps will refill the primary system blocking the boiler-condenser mode by covering the condensing surface and blocking the single phase mode of natural circulation by trapping a steam void in the hot legs.

REPLY TO STAFF'S PROPOSED FINDING NO. 23

51. The record does not support Mr. Jensen's testimony that the improvements to the EFW system provide "assurance" that EFW will be available for natural circulation. To the contrary, the record indicates that EFW reliability has not been significantly improved and that even after implementation of the long-term improvements, EFW will not be a highly reliable system. (See UCS Proposed Findings Nos. 380 to 471)

REPLY TO STAFF'S PROPOSED FINDING NO. 24

52. The Staff's witness had no basis for his professed agreement with the Licensee's allegations concerning the extent to which natural circulation provided adequate cooling during the first month of the TMI-2 accident. (See UCS Proposed Finding No. 21) Further, the Staff's witness testified that he had not done an investigation of the TMI-2 accident and was not attempting by his testimony to explain in detail what happened at TMI-2. Mr. Jensen's testimony was not based on an investigation of the accident at TMI-2. (Tr. 4966-67, Jensen)

REPLY TO STAFF'S PROPOSED FINDING NO. 26

53. The Staff's "concurrence" with the Licensee regarding the capability to cool the core using the bleed and feed mode cannot be accorded any weight since it was nothing more than a hollow echo of the Licensee's position. The Staff has neither performed nor reviewed an analysis of the decay heat removal capability of the bleed and feed mode. (Tr. 16,848, 16,873, Wermiel)

REPLY TO STAFF'S PROPOSED FINDINGS NOS. 27 and 28

54. The Staff study referenced in this proposed finding is NUREG-0565. (Jensen, Natural Circulation, ff. Tr. 4913, at 10). The sentence cited in the direct testimony and repeated in the proposed finding appears at page 4-72 of NUREG-0565. Reviewing the entire section 4.2.11 of NUREG-0565 which contains this statement discloses the qualifications that must be attached to this statement.

55. The statement applies only to non-condensible gases which are already in or could be introduced into the primary system: dissolved hydrogen in the primary coolant; dissolved air in the borated water storage tank; hydrogen gas (free

and dissolved in the makeup tank); pressurizer steam space gas; and fission and fill gases assuming one percent failed fuel. Other sources of non-condensible gases were not included. (NUREG-0565 at 4-69)

56. For example, the free nitrogen used to pressurize the core flood tanks was assumed not to be injected into the primary system on the assumption that the operator would shut off the reactor coolant pumps promptly in the event of a main steam line break and the operator would isolate the core flood tanks during a cooldown following a small break LOCA. (NUREG-0565, at 4-67, 4-68) Thus, failure of the operator to follow emergency procedures may result in more than about 22 cubic feet (Tr. 5019, Jensen) of non-condensible gas in the primary system. The record does not contain evidence on which we can determine whether this additional gas could block natural circulation.* Furthermore, this testimony was based on the assumption that core damage of the extent that occurred at TMI-2 does not occur. (Tr. 4991-92, Jensen)

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* The Staff's witness testified that a void of 360 cubic feet is required to block natural circulation in one loop. (Tr. 5019, Jensen) However, NUREG-0565 states that a void of 125 cubic feet will block both loops. (NUREG-0565, at 4-70)

57. Perhaps, more importantly, even if the Staff's proposed findings concerning the extent to which non-condensible gases could interfere with natural circulation did not suffer from the deficiencies discussed above, they still have little import. The evidence is that the majority of accidents within the small break LOCA spectrum will cause sufficient steam voiding to interrupt natural circulation. (Licensee's Proposed Finding No. 10) The steam will be trapped in the hot leg U-bends and will not be condensed. (Tr. 4619-21, Jones) Thus, the same result occurs - voiding sufficient to interrupt natural circulation - whether the source is non-condensible gas or steam.

58. Staff's proposed finding No. 27 also states that the RCS high point vents will be installed at TMI-1 prior to restart. It is inconceivable that anyone who sat through these hearings could be unaware that this is not the case. The vents will not be installed at restart. (See, as just one example, Staff Ex. 14, at 53)

UCS CONTENTION 3

REPLY TO LICENSEE'S FINDINGS, PARAGRAPH NO. 135

59. In note 51, Licensee attempts to argue that Mr. Pollard was incorrect in stating that loss of offsite power is a "relatively frequent anticipated operational occurrence." In fact, these are precisely the Commission's own words:

> ...B&W designed reactors place more reliance on the reliability and performance characteristics of the auxiliary feedwater system, the integrated control system and the emergency core cooling system (ECCS) performance to recover <u>from frequent anticipated transients, such as loss of offsite power</u> and loss of normal feedwater, than do other PWR designs.
>
> CLI 79-8, 10NRC 141, 143 (Emphasis added)

60. In addition, as is noted in NUREG-0578 at A-1:

> <u>Pursuant to NRC regulations in 10 CFR Part 50, Appendix A, the loss of offsite power is considered to be an anticipated operational occurrence (AOO), since it is expected to occur one or more times during the life of a nuclear plant</u>.

61. Thus, Licensee is wrong when it asserts that the reliability of off-site power at TMI-1 "is sufficiently high that such an event is not expected to occur during the lifetime of the plant." The regulations stipulate otherwise and the Licensee has suggested no reason why they should be waived or deemed inapplicable for TMI-1.

REPLY TO LICENSEE'S FINDINGS PARAGRAPH NO. 136

62. In its attempt to downplay the significance of UCS's testimony on this subject, the Licensee seriously misrepresents the depth of consideration given by Mr. Pollard to the issues. The pages cited by the Licensee actually contain the following questions and answers:

> My question is: Is your evaluation presented in your testimony here on the importance of the pressurizer heaters and associated controls based upon any independent analysis you have performed of the role of pressurizer heaters in accident mitigation, or is it based exclusively on the staff's Lessons Learned Report?
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> A Certainly the development of the contention originally was based upon the staff's Short-term Lessons Learned Report. Between that time and this deposition and my preparation for this testimony, I have given thought to the use of the pressurizer heaters in their role of maintaining natural circulation. I have also given thought to the use of the pressurizer heaters for other functions other than natural circulation, and I have paid particular attention to the use of the pressurizer heaters in Three Mile Island Unit 1 emergency procedures.
>
> The best way I can answer your question is that my testimony is based in part on my own evaluation, but I have,

of course, no way to be sure that my evaluation has not been influenced by the technical documents which the staff has produced in evaluating the accident or by the technical documents produced by the various investigative bodies from the accident.

Q You state that you have thought about the role of pressurizer heaters in natural circulation and about their use in the plant's procedures. What is the thought process involved beyond reading the procedures?

A Well, it involves considering the phenomenon that occurred and the events that occurred at the Three Mile Island Unit 2. It involves recollection of my own personal experiences of the difficulties encountered in solid water operation of naval nuclear power plants. It involves my own personal knowledge of what can happen to operators, even well-trained operators, when faced with unusual circumstances based upon my Navy experience.

It involves considering previous accidents or incidents in commercial nuclear power plants of essentially uncontrolled pressure excursions, certainly not controlled within a permissible range. It involves an evaluation of what reasoning I may find in the staff documents assessing the accident and the need for upgrading of the heaters. Considerations like that.

There may be others, but that gives you some idea of my thought processes.

63. Mr. Pollard did not determine whether B&W took credit for the operation of the pressurizer heaters because that is entirely irrelevant to the question of whether, in the light of the lessons learned from the TMI-2 accident, the heaters should be classified as safety grade. He was well aware that the heaters were considered by B&W to be non-safety-grade. That is not dispasitive of the issue of whether they do, in fact,

have functions which are "important to safety" within the meaning of the NRC regulations.

64. We note that Mr. Pollard uses the words "important" to safety rather than "required" as the Licensee uses, precisely because the regulations use the phrase "important to safety". It is the Licensee which has developed its own terminology-i.e. "essential" and "required"-in order to support its arguments. These phrases do not appear in the regulations.

65. The Licensee also appears to believe that it is somehow significant that UCS has not sought relief against other plants which may exhibit similar safety deficiencies. (Licensee's Proposed Findings, note 53) The intended point of this escapes the Board entirely. UCS is not charged with a responsibility to seek remedial action against all plants with safety problems, nor can any negative references be drawn from its failure (or inability) to do so.

UCS CONTENTION 5

REPLY TO LICENSEE'S PROPOSED FINDINGS, PARAGRAPH NO. 160

66. The record does not contain evidence to support a conclusion that the PORV and safety valves are qualified to relieve liquid on two-phase water.

67. Footnote 55 is purely gratuitous and we reject it. Mr. Pollard thoroughly explained the aspect of his testimony which Licensee cites. Furthermore, NUREG-0578 also refers to PORVs and block valves.

REPLY TO LICENSEE'S PROPOSED FINDINGS, PARAGRAPH NO. 163

68. This proposed finding constructs an argument that is both purely semantic and irrelevant. By artificially defining "failure" of the safety valves as solely failure to perform the overpressure protection function, the Licensee evades addressing the issue of failure of the safety valves to reclose, thereby causing or aggravating a LOCA.

REPLY TO LICENSEE'S PROPOSED FINDINGS, PARAGRAPH NO. 164

69. This proposed finding implies that failures of PORVS

to close which occurred during plant conditions other than high power levels are somehow insignificant. The record does not support such an interpretation of the failure data. Failures to operate properly at any time are fully part of the data base for the PORV failure rate.

REPLY TO LICENSEE'S PROPOSED FINDINGS, PARAGRAPH NO. 167

70. The Licensee urges the Board to reject Mr. Pollard's testimony on the grounds that it is not also attributable to those responsible for the modifications designed to reduce the frequency of challenges to the PORV without increasing the frequency of safety valve operation. Licensee urges us to instead adopt Mr. Jones testimony, but makes no attempt to demonstrate that that testimony is also attributable to those responsible for the modifications. We reject the Licensee's proposition.

71. IE Bulletin 79-05B states that one way of reducing the possibility of void formation in the reactor coolant system is to "Minimize the operation of the...(PORV)... and thereby reduce the possibility of pressure reduction by a PORV that was stuck open." Thus, Mr. Pollard's testimony is fairly attributable to those responsible for the modification. The testimony cited by Licensee actually belongs to witness Jones alone.

REPLY TO LICENSEE'S PROPOSED FINDINGS, PARAGRAPH NO. 168

72. The Licensee alleges that the "PORV is fully qualified (i.e., to GDC 1, 14, 15 and 30) as a reactor coolant system pressure boundary device." However, the testimony cited by Licensee appears to have given no consideration to the electrical circuits of the PORV or the instrumentation and controls. The testimony of Mr. Pollard clearly demonstrates that the PORV does not meet the criteria applicable to safety grade electrical equipment. The Licensee makes no attempt to address the testimony comparing the qualification of the PORV to the requirements which must be met by the high point vents to be installed at TMI-1. (See Pollard, ff. Tr. 9027, at 5-8 to 5-9; UCS' Proposed Findings Nos. 186)

73. The Licensee asserts that the statement that the PORV contributes significantly to the probability of a small break LOCA is irrelevant today. There is no evidence from the Licensee that the change in setpoints has so reduced the probability of the PORV opening that the probability of a stuck open PORV is significantly less than the probability of a small break LOCA from other causes. The Staff has not completed its evaluation of this point. (Staff Ex. 12, at II.k.2.14-1 to 3) Furthermore, NUREG-0578 states that the

modifications ordered are only the first step. "[T]here is a need to consider the upgrading of the PORVs, block valves, and the associated control and power equipment to a safety-grade classification to achieve greater valve reliability and <u>to minimize the number of challenges to the operation of the emergency core cooling components and systems</u>." (NUREG-0578, at A-3 to A-4)

REPLY TO LICENSEE'S PROPOSED FINDINGS, PARAGRAPH 120

74. This proposed finding rests on the basic premise that a stuck open PORV is acceptable because ECCS is available. We reject that premise as inconsistent with the Commission's defense-in-depth approach. (See UCS' Proposed Findings No. 175)

75. In addition, for this finding to be given any weight the PORV position indication instrumentation and the PORV block valve would have to be safety grade components, including conformance with the environmental qualification requirements of GDC-4. No such evidence is on the record.

REPLY TO LICENSEE'S PROPOSED FINDINGS, PARAGRAPH 171

76. Reducing the frequency of challenges to the ECCS so

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REPLY TO LICENSEE'S PROPOSED FINDINGS, PARAGRAPH 120

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REPLY TO LICENSEE'S PROPOSED FINDINGS, PARAGRAPH 171

76. Reducing the frequency of challenges to the ECCS so

that it is within the design basis of the ECCS is more than an operational concern. The probability of a LOCA times the probability of failure of the ECCS must be less than about 10^{-6} per reactor year.

77. As for the proposition that a stuck-open PORV which is not isolated by closing the block valve poses an "unnecessary" challenge to the ECCS, it is absurd on its face considering nothing more than the TMI-2 accident itself. (See, UCS' Proposed Findigns Nos. 179-181)