LIC 6/12/81

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

PART TWO OF

LICENSEE'S PROPOSED FINDINGS OF FACT AND CONCLUSIONS OF LAW ON PLANT DESIGN AND PROCEDURES ISSUES IN THE FORM OF A PARTIAL INITIAL DECISION

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O. Additional LOCA Analysis

Board Question/UCS Contention No. 8: 10 CFR 50.46 requires analysis of ECCS performance "for a number of postulated loss-of-coolant accidents of different sizes, locations, and other properties sufficient to provide assurance that the entire spectrum of postulated loss-of-coolant accidents is covered." For the spectrum of LOCA's, specific parameters are not to be exceeded. At TMI, certain of these were exceeded. For example, the peak cladding temperature exceeded 2200° fahrenheit (50.46(b)(1)), and more than 1% of the cladding reacted with water or steam to produce hydrogen (50.46(b)(3)). The measures proposed by the staff address primarily the very specific case of a stuck-open power operated relief valve. However, any other small LOCA could lead to the same consequences. Additional analyses to show that there is adequate protection for the entire spectrum of small break locations have not been performed. Therefore, there is no basis for finding compliance with 10 CFR 50.46 and GDC 35. None of the corrective actions to date have fully addressed the demonstrated inadequacy of protection against small LOCA's.[104]

Board Question Regarding UCS Contention 8: The board directs the staff and the licensee to present experts and the

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104 ECNP Contention 1(e) was accepted by the Board to the extent that it relates to a further analysis of the spectrum of small-break LOCAs, and ECNP was permitted to adopt UCS Contention 8. First Special Prehearing Conference Order, LBP-79-34, 10 N.R.C. 828, 844 (1979). Consequently, the ECNP contention was not addressed separately in the hearing and it is not quoted here. See Board Memorandum and Order, September 8, 1980, at 3. We note that ECNP did not appear to participate in any of the evidentiary sessions at which this issue was heard.

> fundamental documents involved in the small break LOCA analysis, and to have very complete testimony on this subject. The recommendations of NUREG-0565 and NUREG-0623 should be addressed.
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> It appears from the small break LOCA analysis that there is a large amount of reliance upon operator action and on non-safety-grade equipment. The board wants that issue explored by testimony, including why such reliance is proper.

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333. The TMI-2 accident was equivalent to a small-break, loss-of-coolant accident. UCS Contention 8, which was not objected to and was admitted by the Board without limitation, challenged the adequacy of the analyses performed to identify appropriate corrective actions for the entire spectrum of small-break LOCAs. Yet, on July 31, 1980, in "Union of Concerned Scientists' Review of Contentions," UCS withdrew its Contention 8, but asked the Board to pursue it. The Board not only adopted UCS Contention 8, but added its own questions (quoted above) on the contention. Consequently, the entire record we are about to address in this section of the Initial Decision was developed only because the Board, in its discretion, elected to explore the issue of small-break LOCA analyses.

334. In response to the Board's interest in the additional small-break LOCA analyses performed since the TMI-2 accident, a very complete and extensive record has been

compiled. The record includes the testimony for Licensee of a Supervisory Engineer of B&W's ECCS Analysis Unit (Jones) and of GPU's Control and Safety Analysis Manager (Broughton), describing the purposes, assumptions and results of the small-break analyses for TMI-1 conducted both before and after the TMI-2 accident, and the development of operator guidelines and procedures for small-break LOCA mitigation on the basis of these analyses. Mr. Jones has performed both large and small break ECCS analyses under AEC and NRC regulatory criteria and is responsible, within B&W, for the calculation of large and small break ECCS evaluations, evaluations of mass and energy releases to the containment during a LOCA, and the performance of best-estimate pre-test predictions of LOCA experiments. He has also been personally involved in the preparation of B&W operator guidelines for small-break LOCAs and inadequate core cooling mitigation. Jones and Broughton, ff. Tr. 5038 (attached statement of qualifications, Robert C. Jones, Jr.).

335. The record includes, as well, the fundamental documentation of the results of Licensee's small-break LOCA analyses, which was thus available for the close scrutiny of the Board and the parties. Licensee Exhibits 3 and 4 constitute the spectrum of small break analyses submitted prior to the TMI-2 accident to demonstrate compliance with the requirements of 10 C.F.R. § 50.46 and Appendix K to 10 C.F.R. Part 50. Licensee Exhibits 5 through 9 and 13 consist of additional analyses of plant response to various small-break

scenarios, which were performed in response to specific NRC directives, orders and requests following the TMI-2 accident. Licensee Exhibit 12 is the B&W "Small Break Operating Guidelines," developed to provide guidance for operator actions based upon the results of the small-break analyses. Licensee Exhibits 47 and 48 are the TMI-1 emergency procedures for small-break LOCAs, which implement the B&W guidelines.[105] Licensee Exhibits 10 and 11 were performed in response to NRC IE Bulletin 79-05C, and address the need for, and the consequences of, a prompt reactor coolant pump trip upon receipt of a low pressure (1600 psig) Engineered Safety Features Actuation System ("ESFAS") signal.

336. Licensee also presented testimony in response to each of the recommendation , applicable to licensees, in NUREG-0565, "Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox Designed 177-FA Operating Plants" (January 1980), and in NUREG-0623, "Generic Assessment of Delayed Reactor Coolant Pump Trip during Small Break Loss-of-Coolant Accidents in Pressurized Water Reactors" (November 1979). Jones and Broughton, ff. Tr. 5039.

337. The NRC Staff provided for the record documentation on the results of its review of the B&W small-break LOCA analyses performed in response to NRC direction and

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105 These revise supercede earlier versions of these procedures, UCS Ex. 8 and 6, respectively.

requests following the TMI-2 accident, including the Staff's own audit calculations used in the review. See NUREG-0565 (Board Ex. 4); Tr. 5006-07 (Jensen). Short-term action 1(d) of the Commission's Order and Notice of Hearing (10 N.R.C. at 144) stated that Licensee shall "[c]omplete analyses for potential small breaks and develop and implement operating instructions to define operator action." The Staff's review of Licensee's compliance with this action is documented in the record in Staff Ex. 1 at C1-12 to C1-16. See also, Staff Ex. 1 at C2-16 (IE Bulletin 79-05C short-term action on small-break LOCA analyses); Staff Ex. 1 at C8-48 and Staff Ex. 14 at 43-44 (NUREG-0578 recommendation 2.1.9.a on analysis, emergency procedures and training to substantially improve operator performance during a small-break LOCA). Long-term action 2 of the Commission's Order and Notice of Hearing (10 N.R.C. at 145) recommended that Licensee should be required to "give continued attention to transient analysis and procedures for management of small breaks by a formal program set up to assure timely action of these matters." The Staff's review finding that Licensee has made reasonable progress toward the satisfactory completion of this action is documented in Staff Ex. 1 at D2-1, and in Staff Ex. 14 at 50.

338. The Staff also presented testimony on its reaction to Licensee's responses to the recommendations made in NUREG-0565 and NUREG-0623 (see paragraph 336, supra), and on the relationship of the implementation of those recommendations

with other TMI-related requirements imposed or recommended by the Staff. Ross and Capra, ff. Tr. 15,806.

339. The Commission has established, by regulation, the standards to be applied in evaluating loss-of-coolant accidents for the purpose of specifying the design of the emergency core cooling system. See 10 C.F.R. § 50.46 (acceptance criteria for emergency core cooling systems for light water nuclear power reactors), and Appendix K to 10 C.F.R. Part 50 (ECCS evaluation models).

340. Prior to the TMI-2 accident, small-break LOCA evaluations had been performed to verify conformance of TMI-1 to 10 C.F.R. § 50.46. In order to perform these analyses, the break location which imposes the most severe requirements on the ECCS was identified. As a result of this identification, an analysis was performed of the core flood line break, which results in only one core flood tank and one high pressure injection train available to mitigate the accident under the worst single failure assumption. An analysis of a spectrum of breaks in the reactor coolant pump discharge piping was also performed, since a break at that location results in the loss of a portion of the HPI fluid. These analyses were performed using the B&W ECCS evaluation model which has been approved by the NRC as meeting the requirements of Appendix K to 10 C.F.R. Part 50. For the worst-case break, the peak cladding temperature was found to be less than 1100°F, and no metal-water reaction nor cladding rupture were calculated to occur.

Therefore, conformance to 10 C.F.R. § 50.46 was demonstrated. Jones and Broughton, ff. Tr. 5038, at 2, 3 and 12; Lic. Exs. 3 and 4, and oral summary at Tr. 5047-64 (Jones); Jensen, ff. Tr. 5496, at 4-6. TMI-1 continues to be in compliance with 10 C.F.R. § 50.46. Jensen, ff. Tr. 15,808, at 3; Tr. 5023 (Jensen); Tr. 5196 (Jones). A principal finding of the Staff's generic review is that the original LOCA analyses for TMI-1 remain valid. Staff Ex. 1 at C1-13.

341. The analyses performed prior to the TMI-2 accident assumed the use of only safety-grade equipment for accident mitigation, except that emergency feedwater was assumed to be available.[106] These analyses assumed no mitigating operator actions within ten minutes of the initiating event, except that operator action to cross-connect the HPI system was determined to be required in the event of a small break in the reactor coolant pump discharge piping and the postulated failure of the HPI train which discharges into the unbroken coolant loop. Subsequent modifications to the HPI lines have been made, however, to add cross connections and flow-limiting devices to ensure sufficient flow without operator action. Jones and Broughton, ff. Tr. 5038, at 3, 4; Jensen, ff. Tr. 5496, at 7; Tr. 5605 (Jensen).

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106 See Section II.Q, supra, for the Board's findings on the reliability of the emergency feedwater system. In the event of a loss of all feedwater, however, the feed-and-bleed mode of emergency cooling is available for LOCA mitigation. Jones and Broughton, ff. Tr. 5038, at 4.

342. We now turn to the additional small-break LOCA analyses performed since the TMI-2 accident, which are the subject of the Board's questions on the former UCS Contention 8. First, however, it is imperative to understand why these additional analyses were performed. Because the severity of the TMI-2 accident was aggravated by operator actions, these analyses were performed for the purpose of providing an improved analytical basis for plant emergency operating procedures for responding to small-break LOCAs. Jones and Broughton, ff. Tr. 5038, at 4, 5; Board Ex. 4 at 1-1. This purpose is evident from the language of the Commission itself in short-term action 1(d). See paragraph 337, supra. These analyses performed after the TMI-2 accident were not done to demonstrate compliance with 10 C.F.R. § 50.46. Jones and Broughton, ff. Tr. 5038, at 5; Tr. 5131, 5194 (Jones). Indeed, in an effort to develop an improved set of operator guidelines, these analyses go beyond the scope of Appendix K to 10 C.F.R. Part 50 (for example, in the types and numbers of failures assumed). Tr. 5194-95 (Jones).

343. The small-break LOCA analyses performed after the TMI-2 accident included an extension of the lower end of the break spectrum previously analyzed, an assessment of the effect of failures in the main and emergency feedwater systems, and an assessment of small-break LOCAs with a delayed reactor coolant pump trip. Jones and Broughton, ff. Tr. 5038, at 5.

344. The generic analyses performed by B&W are applicable to TMI-1. Jones and Broughton, ff. Tr. 5038, at 9,

10. In fact, the analyses generally assumed, however, less HPI flow than the TMI-1 system, as modified, will provide. Tr. 5062, 5127 (Jones). The HPI system at TMI-1 will produce roughly 10% more flow than was assumed in the analysis. Tr. 5143 (Jones).

345. The first case examined in the additional LOCA analyses is a loss of all feedwater without a small-break LOCA. In this scenario, it is assumed that: loss of main feedwater occurs; the anticipatory trip on loss of main feedwater fails and the reactor trips on high reactor coolant system pressure; loss of off-site power occurs coincident with the reactor trip; emergency feedwater is not provided to the steam generators; while reactor coolant system pressure continues to increase, the PORV does not open and the pressurizer safety valves open; there is a single failure in the HPI system. The results of this analysis, which also assumed a core decay heat value of 1.0 times the ANS standard value,[107] are that operator action within 20 minutes either to establish emergency feedwater or to actuate manually the HPI system is sufficient to assure

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107 Appendix K modeling assumptions call for the use of a core decay heat value of 1.2 times the standard ANS value. The number of failures assumed in this evaluation, however, and in the one other case where this departure was made, justifies the use of the more realistic 1.0 times the standard ANS value. Tr. 5072-73 (Jones). It should also be noted that a substantial number of investigations, including core decay heat experiments, have demonstrated that the 1971 ANS value used in Appendix K is conservative, so that a core decay heat value of 1.0 times the ANS standard value is adequate, for a realistic determination, to define properly the core decay heat. Tr. 5208 (Jones).

adequate core cooling. Jones and Broughton, ff. Tr. 5038, at 5 and 13 (Table 2); Lic. Ex. 9, and oral summary at Tr. 5064-73 (Jones).

346. The second case examined is a small-break LOCA with the loss of all feedwater. In this scenario, it is assumed that: a small-break LOCA occurs; the reactor trips on low reactor coolant system pressure; there is a loss of off-site power and a loss of main feedwater coincident with the reactor trip; emergency feedwater is not provided to the steam generators; core decay heat is 1.2 times the ANS standard value; and both HPI trains function.[108] The results of the analysis show that for break sizes greater than 0.01 ft^2, emergency core cooling is initiated automatically and no operator action is required to assure adequate core cooling. For break sizes equal to or less than 0.01 ft^2, the setpoint for automatic HPI actuation is not reached. Operator action within 20 minutes to initiate emergency feedwater (which, in turn, will subsequently result in high pressure injection) or to initiate HPI will assure adequate core cooling. Jones and

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108 Two HPI pumps are calculated to be required during only portions of the transient and only for a certain range of break sizes below 0.02 square feet and at specific locations. Tr. 4776-77, 4834 (Jones). The number of failures assumed in this evaluation, however, and in the one other case where this assumption is made, leads the Board not to be concerned with this result. The analysis assumes not only a LOCA and the loss of off-site power, but also the unavailability of all main and emergency feedwater. As we find below, the TMI-1 emergency feedwater system is safety-grade for a LOCA. <u>See</u> section II.C, paragraph 406, <u>infra</u>.

Broughton, ff. Tr. 5038, at 5, 6 and 14 (Table 3); Lic. Exs. 5 and 8, and the oral summary at Tr. 5074-85 (Jones).

347. The third case evaluated is a loss of main feedwater event with a pressurizer PORV failure. This basically represents the TMI-2 accident. In this scenario, it was assumed that: a loss of main feedwater occurs; the anticipatory reactor trip on loss of main feedwater fails and reactor coolant system pressure increases; the PORV opens and does not close (an equivalent break area of 0.007 ft^2); reactor trip occurs on high reactor coolant system pressure; emergency feedwater is provided to the steam generators; core decay heat is 1.2 times the ANS standard value; and a single failure occurs in the HPI system. The results of the analysis show that automatic actuation of HPI provides sufficient reactor coolant system inventory to assure adequate core cooling. Jones and Broughton, ff. Tr. 5038, at 6 and 15 (Table 4); Lic. Ex. 5, and the oral summary at Tr. 5087-90 (Jones).

348. The fourth case considered is a pressurizer PORV failure followed by a loss of all feedwater. In this scenario, it is assumed that: the PORV fails open and does not close; the reactor trips on low reactor coolant system pressure; off-site power and main feedwater are lost coincident with the reactor trip; emergency feedwater is not provided to the steam generators; core decay heat is 1.0 times the standard ANS value;[109] and a single failure occurs in the HPI system.

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109 See n.107, supra.

The results of the analysis show that automatic actuation of HPI provides sufficient reactor coolant inventory to assure adequate core cooling. Jones and Broughton, ff. Tr. 5038, at 6 and 16 (Table 5); Lic. Exs. 6 and 7, and the oral summary at Tr. 5090-94 (Jones).

349. The fifth case is a very small-break LOCA with a loss of main feedwater. In this case it is assumed that: a very small-break LOCA (0.005 to 0.01 ft^2) occurs; the reactor trips on low reactor coolant system pressure; off-site power and main feedwater are lost coincident with the reactor trip; emergency feedwater is provided to the steam generators; core decay heat is 1.2 times the ANS standard value; and a single failure occurs in the HPI system. For breaks of this size, which cause a loss of coolant inventory at a rate in excess of the capacity of HPI, the steam generators would normally be used to remove a portion of the energy added to the primary system by core decay heat. The analysis shows that during the transition from natural circulation to the boiler-condenser mode of cooling (i.e., from single-phase to two-phase natural circulation), an interruption of the energy removal process from the primary system will occur due to void formation in the hot legs, and primary system pressure will increase. However, the subsequent establishment of steam condensation by the steam generators as a heat removal mechanism controls the repressurization, and automatic actuation of HPI provides sufficient reactor coolant inventory to assure adequate core cooling.

Jones and Broughton, ff. Tr. 5038, at 6, 7 and 17 (Table 6); Lic. Ex. 5, and the oral summary at Tr. 5094-97 (Jones).

350. The next case examined is a small-break LOCA with a delayed reactor coolant pump trip. Analyses have shown that if the reactor coolant pumps operate continuously throughout the LOCA, or are tripped promptly upon receipt of a low reactor coolant system pressure signal, adequate core cooling is provided for all break sizes. For certain break sizes (between 0.025 and 0.2 ft^2), however, adequate core cooling has not been demonstrated if the reactor coolant pumps remain in operation and are subsequently tripped at certain times in the transient. The system behavior which leads to this result is that while continued pump operation provides forced circulation cooling of the core, it also causes, for certain break sizes, more fluid inventory to be discharged through the break than would otherwise occur. As a result of this increased loss of inventory, the fluid in the reactor coolant system will evolve to a high void fraction. If the pumps are tripped after a high void fraction is reached, the available liquid in the reactor coolant system would not be sufficient to keep the core covered, and the ECCS may not provide reflooding of the core at a rate which assures that cladding temperatures are maintained within the criteria of 10 C.F.R. § 50.46. Jones and Broughton, ff. Tr. 5038, at 7-9 and 18 (Table 7); Lic. Exs. 10 and 11, and the oral summary at Tr. 5098-5103 (Jones); Jensen, ff. Tr. 15,808, at 3; Ross and Capra, ff. Tr. 15,806, at 51, 52.

351. Since all analyses have confirmed that the plant can be maintained in a safe condition (as defined by 10 C.F.R. § 50.46) during a small-break LOCA without the reactor coolant pumps operating during the transient, provision for prompt tripping of the pumps upon indication of a LOCA[110] (receipt of a low reactor coolant system pressure safety injection signal) assures that adequate core cooling is provided. Jones and Broughton, ff. Tr. 5038, at 9. Consequently, the NRC Staff issued IE Bulletin 79-05C to all licensees which, among other things, required the implementation of plant operating procedures directing that all operating reactor coolant pumps be immediately tripped upon reactor trip and initiation of HPI caused by low reactor coolant system pressure. The bulletin also required an additional operator to be in the control room to perform this action. Jensen, ff. Tr. 15,808, at 3; Staff Ex. 1 at C2-16; Ross and Capra, ff. Tr. 15,806, at 52. While other, non-LOCA events may lead to a low pressure safety signal, tripping of the reactor coolant pumps for these events still allows adequate core cooling to be provided. Jones and Broughton, ff. Tr. 5038, at 9.

352. The Staff, Licensee and the rest of the nuclear industry, however, are investigating the design and installation of a system to trip the reactor coolant pumps

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110 The analysis shows that the earliest of the range of required trip times is on the order of 3 minutes. Tr. 5189 (Jones).

automatically. Staff Ex. 1 at C2-18; Jensen, ff. Tr. 15,808, at 4; Jones and Broughton, ff. Tr. 5039, at 13 (citing Lic. Ex. 1, Supplement 1, Part 3, response to Question 11) and 26; Ross and Capra, ff. Tr. 15,806, at 52-56. The pursuit of this issue, including a schedule for its resolution, has been incorporated into the Commission's TMI Action Plan. Ross and Capra, ff. Tr. 15,806, at 55, 56.

353. The last case examined in these post-TMI-2-accident analyses is a small-break LOCA with a loss of all feedwater and a subsequent PORV failure. In this scenario, it was assumed that: a very small-break LOCA (0.01 ft^2) occurs; the reactor trips on low reactor coolant system pressure; off-site power and main feedwater are lost coincident with the reactor trip; emergency feedwater is not provided to the steam generators; core decay heat is 1.2 times the ANS standard value; both HPI trains function[111]; reactor coolant system repressurization results in the pressurizer PORV opening and failing to close. The results of the analysis show that operator action within 20 minutes to initiate emergency feedwater (which will subsequently result in high pressure injection) or to actuate HPI provides sufficient reactor coolant inventory to assure adequate core cooling. Jones and Broughton, ff. Tr. 5038, at 8 and 19 (Table 8); Lic. Ex. 13, and the oral summary at Tr. 5103-04 (Jones).

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111 See n.108, supra.

354. It is clear from these extensive analyses that multiple failures must occur before a loss-of-coolant accident can result in a challenge to the criteria of 10 C.F.R. § 50.46, and that small-break LOCAs can be mitigated within those criteria. Jones and Broughton, ff. Tr. 5038, at 5 and 11. Further, the assumption that the operator manually trips the reactor coolant pumps immediately following a small-break LOCA is the only reliance on non-safety-grade equipment and the only short-term operator action assumed in these analyses of small-break LOCAs.[112] Tr. 5204 (Jones); Jensen, ff. Tr. 15,808, at 4. The operators will be trained to perform this action (tripping the reactor coolant pumps), which is clearly indicated and requires no diagnosis. Tr. 5204-06 (Jones); Tr. 5302-03 (Broughton); Jensen, ff. Tr. 15,808, at 4; Staff Ex. 1 at C1-16. Operational experience to date indicates that operators are able to execute this action successfully. Jensen, ff. Tr. 15,808, at 4; Tr. 5189 (Jones). The Board finds that this reliance on operator action is acceptable.

355. As we have previously noted, the results of the NRC Staff's review of the generic small-break LOCA analyses performed by B&W on behalf of operating plants with B&W

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112 The need for manual HPI actuation in 20 minutes arises only for events which are beyond the design basis of the plant. In any case, the operator has unambiguous indications upon which to take such action. Tr. 4836-38, 4867-73 (Jones, Keaten). See also, section II.B (Detection of Inadequate Core Cooling), supra.

systems, including TMI-1, are presented in Board Exhibit 4 (NUREG-0565). The Staff's main conclusions are stated as follows:

> B&W has performed a sufficient spectrum of small break LOCA analyses to identify the anticipated system performance for breaks in this range. These analyses serve as an adequate basis for developing improved operator guidelines for handling small break LOCAs. In addition, these analyses provide an adequate basis for demonstrating that proper operator action coupled with a combination of heat removal from the primary system through the break, the steam generators and with the HPI system, assure adequate core cooling.

Board Ex. 4 at 4-25.

356. Based upon the analyses described above, B&W has developed operator guidelines for managing small-break LOCAs. These guidelines contain two parts: Part I provides the guidelines which define operator actions during a small-break LOCA; Part II provides a description of plant behavior during a small-break LOCA and discusses the effects of the operator actions given in Part I. Lic. Ex. 12; Jones and Broughton, ff. Tr. 5038, at 10. These guidelines include the immediate action to trip the reactor coolant pumps and the subsequent filling of the steam generators to a higher level following reactor coolant pump trip to enhance natural circulation. Staff Ex. 1 at C2-17. See also, id. at C1-14, 15.

357. TMI-1 plant emergency procedures have been developed to implement these B&W guidelines. Jones and

Broughton, ff. Tr. 5038, at 10, 11. These procedures include instructions on starting and stopping reactor coolant pumps, terminating high pressure injection, verifying proper operation of the emergency feedwater system, and monitoring core cooling. Id.; Staff Ex. 1 at C1-15. Licensee revised the procedures as a result of Staff comments generated during its review, and the Staff concluded that the TMI-1 procedures adequately reflect the B&W guidelines. Staff Ex. 1 at C1-15.

358. In response to long-term action 2 of the Commission's Order and Notice of Hearing, Licensee has submitted to the Staff additional information concerning small-break LOCA analyses. While further efforts in this area will be undertaken as a part of the Commission's TMI Action Plan, the Staff has concluded that Licensee has made reasonable progress toward the satisfactory completion of this action. Staff Ex. 14 at 50. See also, Ross and Capra, ff. Tr. 15,806, at 19-21.

359. The Board finds that, contrary to the concerns expressed in former UCS Contention 8, adequate analyses have been performed to show that adequate protective actions have been taken for the entire spectrum of small-break LOCAs. We also find, on the basis of an extensive evidentiary record which was heavily scrutinized by the Board and the parties, that Licensee has not relied improperly upon operator action or on non-safety-grade equipment. In response to the Commission's Order, which imposed requirements with respect to small-break

LOCAs which we find to be both necessary and sufficient, analyses have been performed which demonstrate adequate core cooling capability and which serve as a basis for appropriate guidance for operator action, which has been developed and provided. Consequently, the Board finds that TMI-1 can safely mitigate small-break loss-of-coolant accidents.[113]

P. Systems Classification and Interaction

UCS Contention No. 14: 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 Learn Task Force Report (Short Term). The following quote from page 16 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

113 We note that since the TMI-2 accident another licensing board, in a special proceeding on the adequacy of NRC-ordered modifications at an operating B&W plant, reviewed the capabilities of natural circulation, these same B&W small-break LOCA analyses and operator guidelines, and concluded that the analyses and guidelines were adequate, and that the plant can safely respond to and mitigate small-break LOCAs. See, Sacramento Municipal Utility District (Rancho Seco Nuclear Generating Station), LBP-81-12, 13 N.R.C. ___, slip op. at 59 (May 15, 1981).

> 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 all safety-grade design criteria.[114]

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360. This contention by intervenor UCS involves a frontal and generic attack on the entire licensing scheme

114 In its First Prehearing Conference Order, dated December 18, 1979, the Board limited UCS Contention No. 14 to the "core cooling system." LBP-79-34, 10 N.R.C. 828, 837 (1979).

employed by the Commission. The concern raised in UCS Contention No. 14 is common to all licensed nuclear power plants in the United States. Tr. 8125 (Pollard). UCS challenges the classification as "non-safety-related" of systems and components which UCS contends can have an adverse effect on the integrity of the core. The relief sought is that "[a]ll 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 all safety-grade design criteria." While this contention is extremely general, UCS more specifically contends elsewhere that identified components and systems should be upgraded to meet safety-grade design criteria. See, e.g., UCS Contentions 2 (reactor coolant pumps), 3 (pressurizer heaters), and 5 (PORV and block valve). The Board will address those contentions elsewhere in this decision, on the basis of the evidentiary record compiled specifically to meet those issues.

361. UCS Contention No. 14 makes use of the terms "non-safety-related," "nonsafety systems," "important to safety" and "safety-grade." UCS witness Pollard testified that Commission policy and practice has been to apply the General Design Criteria (Appendix A to 10 C.F.R. Part 50) to structures, systems and components variously referred to as safety-related, safety-grade or important to safety. Pollard, ff. Tr. 8091, at 14-3, 14-4. In order to understand the implications for safety of the Commission's scheme for classifying systems and components, as well as the concern with

interactions among these systems and components, it quickly became apparent that more careful use of these terms was needed and that the Board would have to attempt to master the meaning of the various classification terms employed in the contention, as well as the general design approach employed at TMI-1.

362. The general design approach used at TMI-1 to assure the safety of the public is to provide multiple levels of control or protection features for expected operational events, expected transient conditions, or severe equipment failures or natural phenomena. The equipment used to provide the greatest assurance of protection for the most severe plant accidents, or to assure safe shutdown despite severe natural phenomena, is designed and constructed to the highest standards. Systems designed to less stringent but still rigorous standards are used to control less severe transients and normal operations. The acceptability of the less stringent standard lies in the reduced consequence if these systems fail during a transient or normal operation, and the fact that the resulting event is less severe than (i.e., bounded by) the design basis events for the systems relied upon to protect the public. In the event that these normal control systems fail to perform their function, they are backed up by the equipment fully capable of meeting the resulting event -- the equipment designed and constructed to the highest standards (i.e., fully safety-grade). Keaten and Brazill, ff. Tr. 7558, at 14.

363. Staff witness Conran, in turn, described how the Staff licensing process employs the classification of

structures, systems and components. The Staff's review appears to recognize the general design approach described by Licensee witness Keaten. See paragraph 362, supra. The first class to be considered includes the structures, systems and components important to safety. This class is defined in the introduction to the General Design Criteria ("GDC") as those "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." From its consistent use throughout the GDC and in other parts of the Commission's regulations, it is clear that the term "important to safety" is meant to apply generally to all structures, systems and components addressed in the GDC. Conran, ff. Tr. 8372, at 4.

364. As the Board noted above (paragraph 361, supra), UCS witness Pollard lumps the terms "important to safety," "safety grade" and "safety related" as equivalent to the definition in the GDC of "important to safety." Pollard, ff. Tr. 8091, at 14-3, 14-4. According to the Staff, this is wrong. While the term "safety-grade" is widely used in the Staff's safety review process, it is not defined explicitly in the regulations and its meaning must be inferred from the language of the regulations. Conran, ff. Tr. 8372, at 4.

365. General Design Criterion 1 introduces the notion of different quality levels for plant features with differing safety roles and varying degrees of importance to safety. Specifically, GDC-1 requires application of ". . .

quality standards commensurate with the importance of the safety function to be performed . . ." for structures, systems and components important to safety. Conran, ff. Tr. 8372, at 4.

366. Appendix A to 10 C.F.R. Part 100 implements the concept established in GDC-1 (i.e., gradations in quality levels corresponding to relative safety importance) by identifying explicitly a select sub-class of structures, systems and components (out of the broad class "important to safety") that are required for the performance of specific, critical safety functions (e.g., safe shutdown, accident prevention and consequence mitigation). Specifically, section III.c of Appendix A to 10 C.F.R. Part 100 defines the Safe Shutdown Earthquake (the most severe seismic event analyzed for a nuclear power plant), and requires that certain structures, systems and components (important to safety) be designed to remain functional for that event. These certain plant features and the critical safety functions they must perform are further identified as those necessary to assure:

> (1) The integrity of the reactor coolant pressure boundary,
>
> (2) The capability to shut down the reactor and maintain it in a safe shutdown condition, or
>
> (3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to the guideline exposures of this part.

Very high quality standards must be applied to plant features required for such purposes, in order to assure their availability when called upon and their very high reliability in service. Such considerations are the origin of the term "safety-grade." The Staff applies that term <u>only</u> to the structures, systems and components <u>required</u> to perform the above-identified critical safety functions. Conran, ff. Tr. 8372, at 4, 5.

367. The Staff reports, then, that "safety-grade" structures, systems, and components are a sub-class of those "important to safety," which is the broad class of all structures, systems and components addressed in the General Design Criteria. While all structures, systems and components encompassed by the term "important to safety," including the "safety-grade" subclass, are necessary to meet the broad safety goal articulated in the GDC (<u>i.e.</u>, to provide reasonable assurance that a facility can be operated without undue risk to the health and safety of the public), only "safety-grade" structures, systems and components are <u>required</u> for the critical accident prevention, safe shutdown, and accident consequence mitigation safety functions identified in 10 C.F.R. Part 100. Conran, ff. Tr. 8372, at 6. <u>See</u> <u>also</u>, Tr. 7573 (Keaten) (the requirements for safety-grade equipment are imposed upon those systems which are <u>required</u> to mitigate the consequences of an accident and protect the health and safety of the public). The Staff has identified these structures, systems and components which must be safety-grade and has listed them in detail in Regulatory Guide 1.29. Conran, ff. Tr. 8372, at 6.

368. UCS witness Pollard countered that his experience did not support the distinction made by the Staff between "safety-grade" and "important to safety." Tr. 8096 (Pollard). He further argued that if the Staff's classification scheme is correct, then there should be references in NRC regulations and documents to "non-safety-grade/unimportant to safety" equipment. Tr. 8094 (Pollard). Leaving aside the weight to be given Mr. Pollard's experience, the Board cannot appreciate why he might expect to see equipment unimportant to safety in the regulations or documents of the Nuclear Regulatory Commission -- an agency whose sole charter is safety.[115] Tr. 8398 (Conran). The Staff's explanation of these terms, which was not even attempted by others, appears well founded from the regulations discussed. It is no less sound because it has not previously been articulated so clearly and concisely. The fact is that the nuclear power plants in operation today generally have been licensed in accordance with the classification scheme described by the Staff. Tr. 8410 (Conran).

369. UCS contends that the TMI-2 accident demonstrated errors made in applying the Commission's classification scheme. The first asserted error is that because some non-safety-grade systems were used to mitigate the accident, this illustrates that those systems were erroneously classified

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115 The NRC Staff does review equipment and components which are important to safety, but not safety-grade. <u>See</u> Tr. 7689-90 (Keaten); Tr. 8394-96 (Conran).

and should be safety-grade. Pollard, ff. 8091, at 14-4 to 14-6. It is acknowledged that non-safety-grade systems and components were used in the mitigation of the TMI-2 accident. It is important to remember, however, that resort was made to use of non-safety-grade systems and components in the accident mitigation role only after improper operation of installed safety systems had resulted in severe core damage and other beyond-design-basis conditions. Conran, ff. Tr. 8372, at 8. The central issue is not whether these systems were used, but whether they are required. The real test is whether it is acceptable to have the subject system or component unavailable. If it is acceptable to have a given system unavailable because there are other systems which can protect the health and safety of the public, but the system in question is used because it is available and perhaps familiar to the operators, it need not be fully safety-grade. Tr. 7573-74, 7867 (Keaten). At the time of the accident, TMI-2 had operable safety-grade systems which were fully capable of preventing core damage. Tr. 7703 (Keaten).

370. Another classification error UCS contends was revealed by the TMI-2 accident is the failure to require that systems classified as important to safety meet all the requirements applicable to safety-grade equipment. Pollard, ff. Tr. 8091, at 14-6. Of course, this position assumes that the classification of a system as "important to safety" dictates the applicability of all "safety-grade" design

criteria. The Board has already rejected this view.[116] We further reject emphatically the idea that the Staff faces an "all-or-nothing" choice of directing the upgrade of non-safety systems to fully safety-grade, or making no improvements whatsoever. In some instances (as has been the case for some of the non-safety components which were involved in the TMI-2 accident sequence and recovery process), even though none of the Staff's decision criteria that would require upgrading are met, the Staff may decide as a prudent measure to require upgrading of the system or component in question, but not to fully safety-grade. This might be done in order to improve the availability of the component in question, and thereby provide increased safety margins or greater flexibility for dealing with potential future accident situations. Conran, ff. Tr. 8372, at 10. Such actions have been taken as to several systems and components since the TMI-2 accident. Id. at 13, 14.

371. Mr. Pollard reports this practice to be unprecedented, in his experience. Tr. 8100 (Pollard). The Staff reports that it has been done often in the past. See Tr.

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116 An example of such a deficiency, offered by Mr. Pollard, is the fact that the protection system signals used to initiate ECCS operation were not derived from direct measurements of reactor vessel water level. Pollard, ff. Tr. 8091, at 14-6. The TMI-2 accident, however, did not demonstrate any deficiency in the design for protection system signals, and reactor vessel water level would be a much inferior signal to use than reactor coolant system pressure. Tr. 7570-71 (Keaten).

8403-04 (Conran). Whether or not it is a new practice is irrelevant. The Board believes it is a wise one. The Staff should have the flexibility of ordering improvements short of fully safety-grade, and we do not understand how anyone sincerely interested in enhancing the safety of nuclear power plant operation could oppose such a policy.

372. UCS also asserts that the TMI-2 accident disclosed errors in the determination of the design basis event for which safety-grade systems must provide protection.[117] Pollard, ff. Tr. 8091, at 14-6, 14-7. In sum, UCS witness Pollard does not believe that the General Design Criteria, which are Commission regulations, are adequate to protect the health and safety of the public. Tr. 8115 (Pollard). There were, however, no failures of safety-grade equipment to perform its intended safety function during the TMI-2 accident.[118] Keaten and Brazill, ff. Tr. 7558, at 15. If operator action had not interfered with the proper functioning of the installed safety systems to their design capability, the safety-grade

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117 This allegation, of course, was the subject of UCS Contention No. 13, which was abandoned by UCS in its letter of January 5, 1981.

118 The example of such a deficiency, cited by UCS witness Pollard, is the decay heat removal system. Pollard, ff. Tr. 8091, at 14-7. The decay heat removal system at TMI-2, however, could have been used; it simply was preferred at that time to use other core cooling modes. Further, steps have been taken at TMI-1 to ensure that the decay heat removal system could be used even if the primary coolant contained very high levels of radioactivity. Tr. 7571-73 (Keaten).

systems could have accommodated the effects of non-safety component failures that occurred, and still have prevented the serious core damage and other outside-design-basis effects that resulted. Conran, ff. Tr. 8372, at 11.

373. In short, the TMI-2 accident did not demonstrate that the inherent design capabilities of safety systems were inadequate to protect against failures in non-safety systems, or that there were unacceptable interactions of non-safety-grade equipment with safety systems. In fact, it provided additional insight into the positive results that can be obtained if non-safety systems are available and utilized. Keaten and Brazill, ff. Tr. 7558, at 15. See also, Conran, ff. Tr. 8372, at 7 (it has not been established that non-safety systems alone can have an adverse effect on the integrity of the core). The TMI-2 accident did not demonstrate any inadequacy in the Commission's scheme for classification of systems. Tr. 4907 (Keaten). Contrary to UCS Contention No. 14, the Board finds that it is not necessary or appropriate that all systems and components which can either cause or aggravate an accident or can be called on to mitigate an accident be identified and classified as components important to safety and required to meet all safety-grade criteria. See Tr. 8673-74 (Conran). We believe that, as the term "important to safety" is used here, such systems and components would already be classified as important to safety, and that not all such systems and components need to be safety-grade. Only

components <u>required</u> for specific critical safety functions need to meet safety-grade design criteria. Conran, ff. Tr. 8372, at 8; Tr. 7747 (Keaten).

374. There is no need, then, for any of the non-safety systems or components that contributed to the TMI-2 accident, or that were called upon in the accident recovery process, to be made safety-grade. Reliance can still be placed at TMI-1 on the capability of safety systems currently provided in the TMI-1 design to assure adequate safety, without resort to the general upgrading of non-safety systems and components which would be required by the contention, if proper operation of installed safety systems is assured such that full credit can be taken for the functioning of those systems to design capability.[119] Conran, ff. Tr. 8372, at 11, 12. The conclusion this Board comes to consistently is that the real lessons learned from the TMI-2 accident are in the plant software -- operator training and procedures -- rather than in the plant hardware, which was capable of preventing core damage. <u>See</u> Tr. 7748 (Keaten). The endorsement of UCS Contention No. 14 would require unknown upgrades to unknown systems and components, with the potential safety disadvantage of adding unnecessary complexities to the plant which would make it more difficult

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119 The Staff has taken a number of corrective measures in the aftermath of the TMI-2 accident to better assure that operators will not interfere with the proper functioning of installed safety systems in the future. <u>See</u> Conran, ff. Tr. 8372, at 12, 13.

for the operator to exercise effective control. Tr. 7712-14 (Keaten); Tr. 8675-77 (Conran).

375. UCS witness Pollard takes the position, stated in UCS Contention No. 14, that the Staff's effort to study the question of safety/non-safety systems interaction is inadequate. Pollard, ff. Tr. 8091, at 14-8. As a part of the Commission's overall TMI Action Plan, the NRC Staff does have plans and programs for evaluating possible safety effects of non-safety systems and components generally, and for reassessing the appropriateness of the current non-safety classifications in view of the lessons learned from the TMI-2 accident. Conran, ff. 8372, at 14, 15. The Staff already has efforts underway in this regard at the Lawrence Livermore, Brookhaven National, and Battelle Northwest Laboratories. Tr. 8375-78 (Conran). There appears to be no disagreement among the parties that such efforts should be pursued. The disagreement lies in the schedule for such studies vis-a-vis the proposed restart of TMI-1. Tr. 8172 (Pollard).

376. Board Question 3 inquires in detail into the Staff's Interim Reliability Evaluation Program, and Board Question 2 explores the sufficiency generally of the short-term and long-term actions recommended by the Staff for TMI-1. <u>See</u> section II.T, <u>infra</u>. The Board is convinced, on the basis of its inquiry here and on Board Questions 2 and 3, that it is not necessa. to postpone the restart of TMI-1 for several years until such studies are completed. The existing safety analyses

for the plant and the substantial improvements which have been made and will be made pursuant to this decision are more than adequate to provide reasonable assurance of safe operation until the systems interaction issue is explored in more depth.[120] See Tr. 7574-75 (Keaten). The Board agrees with the Advisory Committee on Reactor Safeguards that a study to examine the plant, from the standpoint of systems interactions that may degrade safety, should be conducted on a timely basis, but that its completion should not be a condition for restart. Staff Ex. 14, Appendix C at 2.

Q. Emergency Feedwater Reliability

Board Question No. 6:

a. Is a loss of emergency feedwater following a main feedwater transient an accident which must be protected against with safety-grade equipment? Would such an accident be caused or aggravated by a loss of non-nuclear instrumentation, such as occurred at Oconee?

b. In what respect is the emergency feedwater system vulnerable to non-safety-grade system failures and to operator errors?

c. What has been the experience in other power plants with failures of safety-grade emergency feedwater systems, if they have such systems in other power plants?

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120 We note that the Commission has chosen not to impose on TMI-1, or any other licensee to date, a requirement to perform the specific evaluation recommended in section 9, NUREG-0585 (Review of Safety Classifications and Qualifications). Tr. 8701 (Conran).

d. What operator action is required to operate in a feed-and-bleed mode following a loss of emergency feedwater?

e. If the emergency feedwater system were to fail, what assurance do we have that the system can be cooled by the feed-and-bleed mode? This is of particular concern if the PORV's and safety valves have not been tested under two-phase mixtures.

f. Can the system be taken to cold shutdown with the feed-and-bleed cooling only? Are both high pressure injection (HPI) pumps required to dissipate the decay heat in the feed-and-bleed mode? The board would like an evaluation of the reliability of the feed-and-bleed system. Has there been any experience using that system?

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

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

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

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

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

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377. Board Question No. 6 was first identified by the Board during the prehearing conference of August 12 and 13, 1980. See Tr. 2394-96. The Board reduced the question to writing in its Memorandum and Order of September 8, 1980 (at A-31 to A-33), and in its Memorandum on Board Questions, dated September 12, 1980. Board Question 6 is entitled, "Emergency Feedwater Reliability," and is divided into parts "a" through "k", which are quoted above.

378. In response to Board Question 6, Licensee filed and presented the following direct evidence:

1. Licensee's Testimony of Gary R. Capodanno, Louis C. Lanese and Joseph A. Torcivia in Response to Board Questions 6.a, 6.b, 6.c, 6.g, 6.h, 6.i, 6.j and 6.k, following Tr. 5642;

2. Licensee's Testimony of Robert C. Jones, Jr. in Response to Board Questions 6.e and 6.f, following Tr. 4588; and,

3. Licensee's Exhibit No. 15, "TMI-1 Emergency Feedwater System."

Part 6.d of the question was answered at page 12 of Licensee's Testimony of Robert W. Keaten and Robert C. Jones in Response to UCS Contention Nos. 1 and 2 (Natural and Forced Circulation), following Tr. 4588.

379. Licensee's Exhibit No. 15, which was prepared especially for this hearing to supplement the written testimony in response to Board Question No. 6, describes the TMI-1 emergency feedwater ("EFW") system as it existed prior to recent modifications, the modifications being made to the system prior to plant restart, and the long-term modifications planned for the TMI-1 EFW system. The exhibit discusses the reliability of the EFW system both before and after these modifications, and compares the system against the NRC General Design Criteria (Appendix A to 10 C.F.R. Part 50) directly applicable to the system design.

380. In response to Board Question No. 6, the NRC Staff filed and presented the "NRC Staff Testimony of J. Wermeil, W. Jensen, E. Lantz, and B. Boger Regarding Emergency Feedwater System Reliability (Board Question 6)," following Tr. 5616 and 6035. No other party filed direct testimony on Board Question 6.

381. At the hearing session of November 5, 1980, before the evidence identified above was offered and presented, the Board, on the basis of its review of the pre-filed testimony, advised Licensee and the Staff that their testimony on Board Question 6 did not address all of the issues the Board intended to be covered by the question. The Board clarified the issues which it intended to be addressed in Board Question No. 6 to include the following:

> How would the emergency feedwater system, if relied upon, bring the plant to cold shutdown?
>
> If emergency feedwater fails, what are the complexities and problems involved in the operation and termination of the feed and bleed cooling mode?
>
> How is an alternative cooling mode, such as restoration of emergency feedwater, initiated in order to bring the plant to cold shutdown?

See Tr. 4812, 4813.

382. In response to this clarification, Licensee filed "Licensee's Supplemental Testimony of Robert W. Keaten, Joseph J. Colitz and Michael J. Ross in Response to Board Question No. 6 (Emergency Feedwater Reliability)," dated November 25, 1980 (following Tr. 16,552).

383. At the hearing session of November 20, 1980, during the examination of NRC Staff witnesses on their initial emergency feedwater reliability testimony, and before the parties had responded to the November 5th clarification of Board Question 6, Administrative Judge Jordan stated his views on the deficiencies he perceived in the evidentiary record on emergency feedwater reliability. The postulates advanced and concerns raised by Administrative Judge Jordan on November 20, 1980, went beyond the Board's November 5th clarification of Board Question 6 and, consequently, were not addressed in the supplemental (Keaten-Colitz-Ross) testimony Licensee filed on November 25, 1980.

384. The concerns raised by Administrative Judge Jordan on November 20, 1980, prompted the filing of still

further written testimony on Board Question No. 6. Licensee filed and presented "Licensee's Second Supplemental Testimony of Robert W. Keaten in Response to Board Question No. 6 (Emergency Feedwater Reliability)," following Tr. 16,612. The Staff filed and presented "NRC Staff Supplemental Testimony of J. Wermeil and J. Curry Regarding Emergency Feedwater System Reliability (Board Question 6)," following Tr. 16,718.

385. The Board's findings of fact on Board Question No. 6 will be divided into the following three parts, to reflect the evolution of the question during the hearing: Part I, the original, written version of Board Question 6; Part II, the oral clarification by the Board at the hearing session of November 5, 1980; and Part III, the issues raised by Administrative Judge Jordan at the hearing session of November 20, 1980. First, however, it is appropriate to discuss the reasons why the Board posed its Question 6, and to establish, in summary fashion, the role and design of the TMI-1 EFW system, and the modifications which have been and will be undertaken.

<u>Emergency Feedwater at TMI-1</u>

386. The Board has already found that the unavailability of emergency feedwater for a short period at the beginning of the TMI-2 accident had no significant effect on its outcome. <u>See</u> paragraph 274, <u>supra</u>. Nevertheless, following the March 28, 1979 accident at TMI-2, the EFW systems for

operating pressurized water reactors were reconsidered to determine where changes might be made in design or operation to improve the likelihood of proper functioning of the system upon demand. This re-examination of PWR EFW systems occurred primarily as part of two post-TMI NRC activities -- the Bulletins and Orders Task Force and the Lessons Learned Task Force. Other post-TMI reviews (such as the Kemeny Commission and the Rogovin Group) did not identify significant modifications specifically related to the EFW system, although many of their general recommendations will tend to improve the reliability of EFW along with other plant systems. Wermeil and Curry, ff. Tr. 16,718, at 2.

387. The NRC Staff's early evaluation of the TMI-2 accident led it to the view that B&W designed reactors appear to be unusually sensitive to certain off-normal transient conditions originating in the secondary system and that, because of features of the B&W design that contribute to this sensitivity, B&W designed reactors place more reliance on the reliability and performance characteristics of, among other systems, the emergency feedwater system, than do other PWR designs. See Commission Order and Notice of Hearing, CLI-79-8, 10 N.R.C 141, 142-143 (1979). Consequently, several of the short and long-term actions recommended by the Director of Nuclear Reactor Regulation go to improvements to the TMI-1 EFW system. Short-term action 1(a) calls for the performance of specified items to upgrade the timeliness and reliability of

the EFW system. Short-term action 1(b) recommends the development and implementation of operating procedures for initiating and controlling EFW independent of integrated control system (ICS) control.[121] Short-term action 2 would require, among other things, IE Bulletin 79-05A items on EFW valve positioning procedures and EFW train operability. See Wermeil and Curry, ff. Tr. 16,718, at 2, 3. Short-term action 8 and long-term action 3, which incorporate the NUREG-0578 recommendations, include NUREG-0578 item 2.1.7.a on automatic initiation of the EFW system, and item 2.1.7.b on EFW flow indication to the steam generators. See, generally, Commission Order and Notice of Hearing, CLI-79-8, 10 N.R.C. 141, 144-145 (1979), and Staff Ex. 1.

388. None of the contentions raised by the intervenors challenge the reliability of the TMI-1 emergency feedwater system.[122] Nevertheless, because of the early concerns voiced by the Staff and the several Commission Order items directed at the EFW system, Board Question 6 was posed to determine whether the TMI-1 emergency feedwater system is sufficiently reliable to permit restart of the plant.

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121 This modification is also discussed in our findings on the integrated control system. See paragraph 182, supra.

122 UCS Contention No. 10 proposes that the design of the EFW system, among others, be modified to prevent operator intervention after automatic initiation. The Board, however, has already rejected this UCS proposal. See section II.D, supra.

389. We proceed, first, to address in a general way the role of the EFW system in plant operation and the design of the system. The primary system reactor coolant normally removes heat from the fuel and transports it through two piping loops (hot legs) to the top of the two steam generators. The cooler fluid then goes out the steam generator cold legs, through four reactor coolant pumps, and back into the reactor vessel and the lower portion of the core. Keaten et al., ff. Tr. 16,552, at 2. See also, id. at Figure 2 (which is also Licensee Exhibit 17) for an illustration of the major plant systems at TMI-1.

390. The two steam generators are large, vertical, tube-in-shell heat exchangers that transfer the primary system heat through tubing walls into the secondary system. The primary coolant passes through the inside of the steam generator tubes. Heat is transferred through the tube surface to the outer, or secondary, side of the tubes where the cooler, secondary fluid is heated. The secondary coolant boils in the steam generators. Keaten et al., ff. Tr. 16,552, at 3.

391. Secondary side makeup water (feedwater) is normally provided by the main feedwater system. The feedwater system contains two main feedwater pumps, three condensate pumps and three condensate booster pumps located in the turbine building which supply the two steam generators. After the reactor has tripped, this system can supply enough feedwater to remove residual heat with only one main feedwater pump, one

condensate pump and one condensate booster pump supplying one steam generator. (<u>See</u> paragraph 422, <u>infra</u>.) The steam produced in the steam generators is normally piped through the containment structure and through the turbine bypass valves to the shell side of a condenser where it is condensed to liquid water. From there the water is returned to the steam generator by the main feedwater system. Keaten et al., ff. Tr. 16,552, at 3.

392. The emergency feedwater system at TMI-1 is an alternate source of steam generator secondary side water supply. In the event main feedwater is not available (for example, the proper combination of the condensate pumps, condensate booster pumps, main feedwater pumps, or the main condenser are not available), the EFW system would supply water from either or both of the condensate storage tanks[123] to the secondary side of the steam generators. The steam produced would be removed through the turbine bypass valves to the main condenser, if available, or through the main steam relief valves or the atmospheric dump valves to the atmosphere. Keaten et al., ff. Tr. 16,552, at 3, 4.

393. The TMI-1 EFW system consists of two feed trains supplied by one turbine-driven pump and two motor-driven

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123 Each of the two interconnected condensate storage tanks has a capacity of 250,000 gallons; and, by Technical Specifications, each is required to contain a minimum of 150,000 gallons of water for EFW use. Another water source is the 165,000-gallon condenser hotwell. A backup source of river water is also available via the Reactor Building emergency cooling pumps. Lic. Ex. 15 at 1, 4.

pumps with common suction sources. Prior to the modifications to the system, it could feed emergency feedwater to either or both steam generators under automatic initiation of the turbine-driven pump or manual initiation of the motor driven pumps.[124] The turbine-driven pump is started automatically either on loss of both main feedwater pumps or on loss of all four reactor coolant pumps. In the case where the turbine-driven EFW pump is not available, prior to the modifications the two motor-driven pumps would be started manually by an operator. Lic. Ex. 15 at 1, 4-5. The two motor-driven EFW pumps can be powered from either on-site or off-site AC power sources. The steam-driven EFW pump requires neither off-site nor on-site AC power sources to operate. Any one of the three EFW pumps supplying water to either of the two steam generators has sufficient capacity to remove residual heat. Keaten et al., ff. Tr. 16,552, at 4. See also, Tr. 5662-72 (Capodanno); paragraphs 415 and 416, infra.

394. The flow of emergency feedwater to each steam generator is controlled by air-operated modulating flow control valves. Positioning of these valves is via electric to pneumatic converters that receive control signals from the ICS. The valves are modulated to maintain the desired steam generator water levels.[125] The valves are also interlocked

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124 A common discharge crosstie permits any of the three pumps to feed either or both of the steam generators through a piping system that is independent of the normal main feedwater system. Lic. Ex. 15 at 1, 2.

125 If all four reactor coolant pumps are tripped, the valves will open and control to the setpoint for reactor coolant (continued next page)

with pressure switches so that emergency feedwater (and main feedwater) is cut off to a given steam generator if a low pressure (less than 600 psig) is detected within that generator. Lic. Ex. 15 at 2.

395. A number of modifications will be made to the TMI-1 emergency feedwater system prior to plant restart. See, generally, Staff Ex. 1 at C1-1 to C1-12, C2-6 and 7, C8-34 to C8-40; Staff Ex. 14 at 13-14, 38-39; Tr. 5672-81 (Capodanno). An important modification is the installation of a safety-grade auto-start for the EFW pumps. The EFW system, as modified for restart, will automatically start the turbine-driven pump and both motor-driven pumps upon:

(a) loss of both main feedwater pumps, or

(b) loss of four reactor coolant pumps.

This auto-start capability will exist with a loss of off-site power, and with a concurrent ESFAS actuation with or without a loss of off-site power.[126] Lic. Ex. 15 at 6; Tr. 5823-26 (Capodanno, Lanese). The EFW pump automatic initiation signals

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(continued)
pump trip. If at least one reactor coolant pump is operating, but both main feedwater pumps have tripped, the valves will open and control to a lower setpoint. If at least one reactor coolant pump and one main feedwater pump are operating, both valves are directed to remain closed. Manual control of valve position is also available in the ICS in the control room. Lic. Ex. 15 at 5.

126 Previously, the loss of off-site power concurrent with ESFAS actuation would have inhibited starting of the motor-driven EFW pumps. Lic. Ex. 15 at 6.

are independent of the ICS. Staff Ex. 1 at C8-35. Licensee has committed to modify the EFW system to provide, prior to restart, control room annunciation for all automatic start conditions of the EFW system. Staff Ex. 1 at C1-7, 8. Prior to restart, Licensee will perform a functional test to verify that all EFW pumps automatically start on loss of feedwater or loss of four reactor coolant pumps. Staff Ex. 1 at C1-1. In addition, all EFW pumps can be started manually from the control room. With these modifications, a single failure will not result in the loss of the EFW system function during a loss-of-coolant accident. Lic. Ex. 15 at 6. Further, a single failure in the automatic initiation system will not result in the inability to actuate the emergency feedwater pumps on a loss of main feedwater or loss of off-site power. Staff Ex. 1 at C8-35.

396. The original EFW system design did not have any provision for indication in the control room of emergency feedwater flow. Safety-grade, redundant indication of EFW flow to each steam generator will be provided in the control room prior to restart. Lic. Ex. 15 at 6; Staff Ex. 1 at C8-39. Licensee has committed to perform a functional test of the new EFW flow instrumentation prior to restart. Staff Ex. 1 at C1-5. Based upon the Staff's review of Licensee's design for providing safety-grade EFW flow indication in the control room and on the information that the flow transducers are qualified for operation in the assumed environment from a postulated main

steam lire break in the Intermediate Building, the Staff has concluded that Licensee is in compliance with the NUREG-0578 recommendation, in item 2.1.7.b, for emergency feedwater flow indication to the steam generators. Staff Ex. 1 at C8-40; Staff Ex. 14 at 39. The Staff will verify that the flow devices are installed and suitably qualified prior to restart. Staff Ex. 14 at 39.

397. Prior to restart, the failure mode of the EFW flow control valves will be changed in order to assure that emergency feedwater can be delivered when required. In the original system design, these valves failed half open on loss of control power, and failed "as is" on loss of instrument air. As a result of the modification, the valve will fail in the open position on loss of instrument air, and will remain in that position. Lic. Ex. 15 at 6; Staff Ex. 1 at C1-1, 2. The modification we disucss next (paragraph 398, infra) will enable the operator to switch to manual control in the event of a loss of control power.

398. Short-term Commission Order item 1(b) requires that Licensee develop and implement operating procedures for initiating and controlling EFW independent of ICS control. In addition to providing automatic initiation of the EFW pumps independent of the ICS, Licensee will provide, in the control room, a separate manual EFW control station independent of ICS for each control valve. When this manual control is selected, all active components of the ICS are bypassed. Power for each

control valve from the backup control station will be derived from the redundant emergency power supplies. Lic. Ex. 15 at 6, 7; Staff Ex. 1 at C1-11. The Staff has reviewed Licensee's conceptual design for this modification, as well as the revised emergency procedures which include operating instructions on the use of the new EFW manual control station. The Staff has concluded that Licensee is in compliance with this part of the Commission order. Staff Ex. 1 at C1-11, 12.

399. A support system which affects EFW system reliability is the air supply for certain air-operated valves. The TMI-1 air sypply system consists of two 60 hp compressors. Lic. Ex. 15 at 3. One of the restart modifications for the EFW system will be the provision of a redundant, two-hour air supply system that will supply instrument quality air to the pressure control valve that regulates steam supply to the turbine, and to the two EFW flow control valves, for a two-hour period in the event of a loss of all AC power. Lic. Ex. 15 at 7. The Staff has verified that EFW system initiation and operation is assured independent of any AC power source for at least two hours. Staff Ex. 1 at C1-9, 10.

400. Prior to restart, the low-low level condition at each of the two condensate storage tanks will be annunciated in the control room. The alarm setpoint will be such that the operator will have a minimum of twenty minutes before either of the tanks is pumped dry. This will provide ample time for the operator to realign the EFW pumps' suction to an alternate

water source. Lic. Ex. 15 at 7; Staff Ex. 1 at C1-8. Separate power supplies for each level transmitter loop will be provided as a longer-term modification. Lic. Ex. 15 at 7; Staff Ex. 14 at 13.

401. Another restart modification is the provision of redundant, single-failure-proof indication in the control room, independent of the ICS, of the level in each steam generator. All hardware used in this modification will be safety-grade. This level indication will assure that the operator can properly control steam generator level, using the new manual loaders added for the EFW control valves, in the event of an ICS/NNI malfunction. Lic. Ex. 15 at 7; Staff Ex. 14 at 38.

402. Licensee has committed, for the long-term, to modify the TMI-1 emergency feedwater system to achieve a single-failure-proof, safety-grade design. Included within the scope of that effort will be:

a. Safety-grade automatic system start;[127]

b. Safety-grade system flow indication in the control room;[128]

c. Safety-grade EFW flow control system for each steam generator;

d. Addition of cavitating venturi in each EFW line;

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127 Safety-grade automatic EFW pump start is being installed prior to restart. See paragraph 395, supra.

128 This is being accomplished as a restart modification. See paragraph 396, supra.

e. Safety-grade condensate storage tank low-low level alarm;

f. Safety-grade steam generator high level alarm;

g. Safety-grade isolation of main feedwater on overfill of an affected steam generator;

h. Upgrade Main Steam Rupture Detection System to safety-grade.

Lic. Ex. 15 at 10, 11.

403. Item 2.1.7.a of NUREG-0578 recommends, as a long-term action, the installation of a safety-grade automatic initiation of the emergency feedwater system. While the NRC's TMI Action Plan (NUREG-0737, Item II.E.1.2) presently calls for installation by July 1, 1981, it has become apparent that Licensee will be unable to meet this schedule as to the safety-grade EFW flow control system. In response to the Staff's request to provide the detailed design of this long-term modification, Licensee has documented, and the Staff has reviewed, the status and major problems being encountered in finalizing the complete, safety-grade EFW system design.[129] While the Staff still will require submittal of the final detailed design for its review prior to system installation, the Staff has concluded, based upon Licensee's good-faith

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129 A number of additional long-term modifications to the EFW system, derived from the analyses and evaluations performed pursuant to item II.E.1.1 of NUREG-0737, are recommended for implementation prior to January 1, 1982. However, it is probable that Licensee will be unable to implement certain aspects of this long-term upgrade until the Cycle 6 refueling outage, due to procurement delays. Ross, ff. Tr. 15,555, at Table 2; Tr. 15,563-65, 15,577-81 (D. Ross, Capra).

effort to procure the required equipment and the similarity of problems encountered by other operating plants in making these modifications, that Licensee has demonstrated reasonable progress toward the satisfactory completion of this long-term action. Staff Ex. 14 at 36-38. The Board agrees.

Part I of Board Question 6

404. Subpart "a" of Board Question 6 asks whether loss of emergency feedwater following a main feedwater transient is an accident which must be protected against with safety-grade equipment, and whether such an accident could be caused or aggravated by a loss of non-nuclear instrumentation. The Staff's position is that the loss of emergency feedwater following a main feedwater transient is not an accident which must be protected against with safety-grade equipment. Wermeil et al., ff. Tr. 6035, at 1. Because the TMI-1 emergency feedwater system will be safety-grade for a loss of main feedwater transient at the time of restart, the loss of both feedwater systems is an accident which is beyond the design basis. Tr. 6082, 6200-01 (Wermeil). In addition, the feed-and-bleed cooling mode, using safety-grade equipment, is available. Capodanno et al., ff. Tr. 5642, at 2.

405. The long-term, safety-grade modification of the EFW system will completely eliminate any intertie between the ICS/NNI and EFW systems. Wermeil et al., ff. Tr. 6035, at 1, 2. Licensee has not been able to identify any single failure

in the ICS that will cause a loss of both main and emergency feedwater. Tr. 5712 (Lanese). One of the authors of the B&W ICS failure modes and effects analysis testified that there is no single failure in the ICS that would prevent both EFW control valves from providing feedwater to the steam generator(s). Tr. 7038-40 (Joyner). However, Licensee will provide, prior to restart, steam generator level and EFW flow indication independent of the ICS, control room indication of failed power supplies, and a manual switch, operable from the control room, to transfer the ICS supply bus from the inverter bus to the regulated AC supply. In addition, TMI-1 will have the capability, prior to restart, to operate the EFW system independent of the ICS. Capodanno et al., ff. Tr. 5642, at 2, 3. Consequently, the operator can take the necessary manual action in the control room to restore EFW flow. Wermeil et al., ff. Tr. 6035, at 2.

406. Question subpart "b" asks the respect in which the EFW system is vulnerable to non-safety-grade system failures and to operator errors. Prior to implementation of the fully safety-grade modification, the EFW system is not safety-grade with respect to a postulated main steam/main feedwater line break, and may not be fully safety-grade with respect to seismic qualification and protection against pipe breaks in other high energy systems. Lic. Ex. 15 at Table 1; Wermeil et al., ff. Tr. 6035, at 2, 3. The TMI-1 EFW system will be safety-grade at restart, however, for a small-break

LOCA and a feedwater transient. Tr. 5691, 5780 (Lanese); Tr. 6200-01 (Wermeil). The EFW system is vulnerable to operator errors, as are all plant systems. However, operational errors that might affect the functioning of the EFW system have been evaluated, and procedural changes, coupled with operator training, have been instituted to assure proper surveillance and operation of the system to preclude loss of function. Capodanno et al., ff. Tr. 5642, at 4; Wermeil et al., ff. Tr. 6035, at 3.

407. Board Question 6.c asks about the experience in other power plants with failures of safety-grade emergency feedwater systems. The NRC Staff reviewed the available data, in Licensee Event Reports, for plants in commercial operation that have safety-grade EFW systems, and found that in the vast majority of cases the failures that occurred did not defeat the functional capability of the system. The Staff reported four cases where sufficient emergency feedwater was not available, although EFW was not required at the time to cool the reactor (plant in startup operations or testing).[130] Wermeil et al., ff. Tr. 6035, at 3, 4. In addition, the Staff explained that at least some of the EFW failures it reported could not occur at TMI-1. Tr. 6136-37 (Wermeil). All plants perfo[illegible] routine periodic EFW system surveillance testing. It is important to

130 Losses of function due to misalignment and operator errors were not included in this list. Wermeil et al., ff. Tr. 6035, at 4.

note, however, that data on EFW system success on demand is not maintained. Wermeil et al., ff. Tr. 6035, at 3, 4.

408. Subpart "d" of Board Question 6 asks for an identification of the operator action required to operate in a feed-and-bleed mode following a loss of emergency feedwater. Licensee testified that the only manual actions required are: (1) for certain scenarios, manual actuation of high pressure injection; (2) if it is utilized, manual opening of the PORV; and (3) if a low level in the Borated Water Storage Tank is reached, switchover of the HPI suction to the containment sump via the low pressure injection system.[131] Keaten and Jones, ff. Tr. 4588, at 12; Tr. 4859-62 (Jones). <u>See also</u>, Wermeil et al., ff. Tr. 6035, at 5; Keaten et al., ff. Tr. 16,522, at 10, 11.

409. Question subpart "e" asks for any assurance that the feed-and-bleed mode can cool the system if EFW fails. The Board's findings of fact on Natural and Forced Circulation describe the basic energy removal processes associated with assuring adequate core cooling and how these related to feed-and-bleed operation. <u>See</u> paragraphs 11-14, <u>supra</u>. Our findings of fact on Additional LOCA Analysis present the results of analyses performed which verify the capability of

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131 Any complexities and problems involved in the operation and termination of the feed and bleed cooling mode are discussed below in response to Part II of Board Question 6. <u>See</u> paragraphs 426-429, <u>infra</u>.

the feed-and-bleed mode to provide adequate core cooling. See paragraphs 345, 346, 348 and 353, supra. The only action required of the PORV and safety valves in feed-and-bleed cooling is that one or more of these valves open to provide a fluid discharge path. Jones, ff. Tr. 4588,[132] at 1, 2. The bases for the Board's conclusion that these valves can be expected to open upon such a demand are presented in section II.R (Valve Testing), infra. Based upon all of these findings, and our findings immediately below on Question 6.f, the Board concludes that there is sufficient assurance that the feed-and-bleed operation can provide adequate core cooling in the event of a loss of all main and emergency feedwater. See Jones, ff. Tr. 4588, at 2.

410. Board Question 6.f probes, in a general way, the reliability of the feed-and-bleed cooling operation, and asks specifically whether it alone can take the plant to cold shutdown and whether two HPI pumps are required to dissipate the decay heat in the feed-and-bleed mode. Feed-and-bleed operation would not directly take the primary system to a cold shutdown condition.[133] Jones, ff. Tr. 4588, at 2; Tr. 4774-75

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132 "Licensee's Testimony of Robert C. Jones, Jr., in Response to Board Questions 6.e and 6.f"; and not Keaten and Jones, which also follows Tr. 4588.

133 It may be possible, however, to use the PORV to depressurize the system down to the point where the normal decay heat removal system could be used, without ever regaining EFW flow. Tr. 4864-65 (Keaten). See also, paragraph 425, n.136, infra.

(Jones). However, feed-and-bleed operation can be continued, as required, to assure adequate core cooling until secondary side cooling is available and/or the primary system can be depressurized to allow the Low Pressure Injection system to provide core cooling directly.[134] Jones, ff. Tr. 4588, at 2, 3.

411. One or two HPI pumps are calculated to be required for adequate feed-and-bleed cooling, depending on the specific scenario postulated. Jones, ff. Tr. 4588, at 3; Wermeil et al., ff. Tr. 6035, at 6. See also, paragraph 346, n.108, supra. For a loss of all feedwater event without a small-break LOCA, however, only one HPI pump is required to assure adequate core cooling. Jones, ff. Tr. 4588, at 3; paragraph 345, supra.

412. A quantitative assessment of the reliability of the feed-and-bleed mode of operation has not been performed.

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134 Sufficient water is available in the Borated Water Storage Tank for at least 19 hours of feed-and-bleed operation, assuming two HPI pumps are used. After the BWST has been emptied, feed-and-bleed could be continued for an indefinite period by reinjection of the water "bled" from the system and stored in the containment sump. A primary objective of the operators throughout this time would be to re-establish either main or emergency feedwater flow to the steam generators. The majority of the components of these systems are located outside containment and would be available for service. Once feedwater flow was established, the primary system would be cooled and depressurized utilizing the steam generators. Wermeil et al., ff. Tr. 6035, at 6, 7. There is sufficient water in the BWST, including use of the recirculation mode, to get down to the Low Pressure Injection system. Tr. 16,576-77 (Keaten, M. Ross).

Feed-and-bleed cooling is not required, however, except for an extended loss of all main and emergency feedwater or for certain accident conditions in conjunction with an extended loss of all feedwater, which are beyond the design basis. Jones, ff. Tr. 4588, at 3; Tr. 5201 (Jones); paragraphs 345, 346, 348 and 353, <u>supra</u>. Further, while the PORV may be used as the fluid discharge path from the reactor coolant system, feed and bleed can be accomplished with only safety-grade systems and components -- <u>i.e.</u>, the pressurizer safety valve(s) in conjunction with the borated water storage tank, high pressure injection, containment and low pressure injection. Keaten and Jones, ff. Tr. 4588, at 12.

413. There is some experience which shows that feed-and-bleed operation can provide adequate core cooling. During the February 26, 1980 event at Crystal River 3, the HPI system injected water into the primary system and fluid was discharged initially by the PORV and then by a safety valve. Therefore, the incident was a demonstration of the operability of feed-ard-bleed cooling. It should also be noted that during a portion of the Crystal River transient, secondary side cooling was significantly reduced or non-existent. Throughout the scenario, however, the core was adequately cooled. Jones, ff. Tr. 4588, at 3, 4. <u>See also</u>, Jensen-1, ff. Tr. 4913, at 9, 10.

414. The Board is also aware that the individual systems and components required for feed-and-bleed cooling

(e.g., HPI, LPI and the safety valves) are routinely operated and/or tested to assure their functionability. Jones, ff. Tr. 4588, at 4; Tr. 4886-87 (Jones). For all of these reasons, and because the operator actions required are not complex, the Board finds that feed-and-bleed operation is adequately reliable to perform its potential function. See Jones, ff. Tr. 4588, at 3; Tr. 4778-79 (Jones).

415. Question subpart "g" asks whether both motor-driven EFW pumps will be required to supply the requisite amount of emergency feedwater if there is a loss of steam in the secondary system which results in failure of the turbine-driven pump, and whether this meets the single failure criterion. Licensee's witness explained the flow capacity of the EFW pumps and the system flow requirements for the most severe plant heatup transient (loss of main feedwater) and for small-break LOCAs. The record shows that the EFW system can protect the plant within the established safety limits even with the assumed failure of the turbine-driven pump and one motor-driven EFW pump, even though this situation is beyond the single failure criterion. Capodanno et al., ff. Tr. 5642, at 8, 9. The Staff also testified that one motor-driven EFW pump can supply adequate feedwater for decay heat removal for all postulated accidents and transients, so that the single failure criterion is satisfied. Wermeil et al., ff. Tr. 6035, at 7, 8.

416. Board Question 5.h asks whether the turbine-driven pumps and valves can be operated on direct current, or

whether they are dependent upon the alternating current safety buses. The record shows that the TMI-1 turbine-driven emergency feedwater train can operate to supply feedwater on direct current power sources, and is not dependent upon the alternating current safety buses. Capodanno et al., ff. Tr. 5642, at 9, 10; Wermeil et al., ff. Tr. 6035, at 8.

417. Subpart "i" of Board Question 6 asks whether the reliability of the EFW system will be greatly improved upon conversion to safety-grade, and whether it is the position of Licensee and the Staff that the improvement is enough such that the feed-and-bleed back-up is not required. Licensee's witnesses testified that the ability of the TMI-1 emergency feedwater system to respond to anticipated transients, and many other accidents, will not be substantially improved upon conversion to safety-grade, because the principal deficiencies in the existing EFW system are in the environmental qualification of equipment for relatively improbable, non-LOCA events. Capodanno et al., ff. Tr. 5642, at 11. Licensee also testified, however, that the restart upgrading of the EFW system to safety-grade for small-break LOCAs and loss of main feedwater transients sufficiently improves the reliability of the EFW system that reliance need not be placed on the feed-and-bleed cooling mode. Tr. 5786 (Lanese). See also, Tr. 4816-18 (Keaten). Based on knowledge of the improvement in reliability gained by eliminating first order failure sources, it is the Staff's judgment that the reliability of the EFW

system will be improved when it is fully safety-grade. Further, while the Staff does not require the feed-and-bleed back-up, it is recognized as additional defense-in-depth for providing core cooling in the very unlikely event that both main and emergency feedwater are lost. The HPI pumps and primary safety valves which comprise the feed-and-bleed mode are required, by Technical Specifications, to be available. Wermeil et al., ff. Tr. 6035, at 8, 9.

418. Board Question 6.j asks whether the short-term actions proposed will improve the reliability of the emergency feedwater system to the point where restart can be permitted. Because this raises the ultimate issue which underlies all of Board Question No. 6, we will defer our finding on this subpart of the question until we complete our findings on all of the evidence presented.

419. Subpart "k" advises the parties to address Board Question 6 with reference to the decision in Florida Power and Light Company (St. Lucie Nuclear Power Plant, Unit No. 2), ALAB-603, 12 N.R.C. 30 (1980), i.e., whether loss of EFW is a design basis event notwithstanding whether design criteria are met. Since this Appeal Board decision and its applicability to the TMI-1 EFW system were the subject of additional evidence presented in Part III of Board Question No. 6, we will defer our finding on this aspect of the question until we address that additional evidence.

Part II of Board Question 6

420. The oral clarification of Board Question 6 provided by the Board at the hearing session of November 5, 1980, posed three questions. See paragraph 381, supra. The first question asks how the emergency feedwater system, if relied upon, would bring the plant to cold shutdown.

421. Several methods are available to proceed to cold shutdown from the condition immediately following reactor trip (while the system is still at or near normal system temperature and pressure), depending on the remaining operable equipment. It is important to note, however, as Licensee points out, that the plant can remain in the hot condition for extended periods with any of these methods if the decision to transition to cold shutdown is deferred. Keaten et al., ff. Tr. 16,552, at 8. See also, id. at Figure 1 for an illustration of the TMI-1 core cooling and heat removal paths.

422. In the case of a normal reactor trip, the process of removing the decay or residual heat from the primary or reactor coolant system would be through the steam generators to secondary coolant provided by either of the feedwater supply systems. Assuming an end of life, equilibrium full power history before the time of trip, the decay heat level is approximately 7% of full power at the time of trip. This heat level quickly decays to 4% within 40 seconds and roughly to 1% in an hour. An equivalent percentage of main feedwater flow

would be required to maintain equilibrium reactor coolant system temperature, or approximately 720 gpm of emergency feedwater 40 seconds after trip. The flow requirements and capabilities of the main feedwater pumps are above 50% of full rated power. Consequently, there is abundant capacity in either of the two main feedwater pumps to provide feedwater flow for residual heat removal. Keaten et al., ff. Tr. 16,552, at 6.

423. The normal method for cooldown from operating pressure and temperature is to remove steam from the steam generators at a rate greater than the decay heat generation rate, using the main feedwater system, the turbine bypass valves, and the main condenser. This is accomplished by taking manual control of the turbine bypass valves and opening the valves to a position where the resulting steam flow to the condenser yields the desired cooldown rate of the reactor coolant system. This method can be maintained despite single active failures in the process train including single failures in off-site power feeds. The reactor coolant system can be cooled by this method to the point that the decay heat removal system is put into operation (about 250°F/320 psig). The decay heat removal system can then continue the normal shutdown cooling process until the conditions of cold shutdown are reached (reactor coolant system temperature less than 200°F). Keaten et al., ff. Tr. 16,552, at 9.

424. If main feedwater is unavailable, the EFW system will provide sufficient secondary coolant. As we have

noted (paragraph 393, <u>supra</u>), the EFW system at TMI-1 has two flow paths, supplied by one turbine-driven pump and two motor-driven pumps, which can supply emergency feedwater to either or both of the steam generators. The turbine-driven pump has a rated capacity of 920 gpm, and each motor-driven pump has a rated capacity of 460 gpm. Either one turbine-driven pump or both motor-driven pumps exceed the requirements to remove the 7% residual heat that exists at the time of reactor trip. By two and one-half minutes after trip, one motor-driven pump has enough capacity to remove the decay heat. Even if only one motor-driven pump were available initially, adequate heat removal would also be provided.[135] Keaten et al., ff. Tr. 16,552, at 7; paragraph 415, <u>supra</u>.

425. Where the main feedwater system is lost and the condenser is available, the secondary system will function as a closed loop by steaming through the turbine bypass valves to the condenser and water drawn from the condenser by the emergency feedwater pumps and returned to the steam generators. If the condenser is not available, steam can be released to the atmosphere via the atmospheric dump valves. These valves can be controlled in the same manner described above for the

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135 In this case, reactor coolant system temperature and pressure would initially increase, possibly resulting in lifting a relief valve. As decay heat drops, however, the single EFW pump would supply enough water to overcome the temperature/pressure rise and restore normal conditions. Keaten et al., ff. Tr. 16,552, at 7.

turbine bypass valves in order to achieve the desired cooldown rate. In this cooling mode water from the condensate storage tanks is fed to the steam generators by the emergency feedwater systems and then released to the atmosphere. The condensate storage tanks are required by the Technical Specifications to have 150,000 gallons in each tank during reactor operation. This amount of water is more than adequate to allow the reactor coolant system to be cooled to the temperature and pressure where the decay heat removal system can be placed in operation, prior to the depletion of inventory in the condensate storage tanks.[136] Keaten et al., ff. Tr. 16,552, at 9, 10. See also, paragraph 410, n.134, supra.

426. The second clarification question posed by the Board asks what complexities and problems are involved in the operation and termination of the feed-and-bleed cooling mode. Initiation of the feed-and-bleed cooling mode is a very simple operation. If neither main nor emergency feedwater is available, the operator will initiate and maintain full high pressure injection until feedwater is restored. The operator can open the PORV and its block valve, or allow the code safety valves to open to provide a flow path. Keaten et al., ff. Tr. 16,552, at 10.

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136 It is also possible to take the reactor to cold shutdown, to the point where the LPI system is operable, without emergency feedwater, by depressurizing with the PORV. Tr. 16,575 (M. Ross); Tr. 16,685 (Keaten). See also, Tr. 16,725 (Wermeil), and paragraph 410, n.133, supra.

427. Once initiated, the feed-and-bleed cooling mode will automatically continue without need for additional short-term operator actions. In the long term, the operator must transfer the suction of the high pressure injection pumps from the borated water storage tank to the containment building sump via the low pressure injection pumps. If ESFAS has automatically initiated, this transfer requires opening four valves and closing four valves, all of which can be done at the main control console. If ESFAS has not automatically initiated, the LPI pumps must be started manually, but this also can be accomplished from the main control console. Keaten et al., ff. Tr. 16,552, at 10.

428. Termination of the feed-and-bleed cooling mode is also very simple. Once the appropriate criteria are met the HPI discharge valves are throttled and eventually the HPI pumps are turned off. These actions are also performed from the main control console. Such throttling and/or termination of high pressure injection, however, is only permissible when specific criteria regarding reactor coolant system conditions are met. (<u>See</u> paragraph 38, <u>supra</u>.) Keaten et al., ff. Tr. 16,552, at 11.

429. It should be noted that the simple actions associated with initiation, continuation and termination of feed-and-bleed cooling would be performed by an operator assigned to this portion of the control panel. Any parallel actions being taken in an attempt to restore main or emergency

feedwater would be taken by a different operator assigned to the feedwater control panel. The TMI-1 Technical Specifications require that two licensed reactor operators be in the control room during startup, shutdown, and recovery from a reactor trip. The normal control room practice is that immediately upon reactor trip one operator goes to the portion of the console from which HPI and LPI are controlled, and the other operator goes to the feedwater control portion of the panel. This allows actions to be carried out in parallel under the supervision of the senior watchstanders. Keaten et al., ff. Tr. 16,552, at 11.

430. The third clarification question posed by the Board asks how an alternative cooling mode, such as restoration of emergency feedwater, is initiated in order to bring the plant to cold shutdown. If no feedwater is available, and the plant is operating in the feed-and-bleed mode, the normal steps taken would be directed at restoring emergency feedwater flow, as described in the follow-up action section of TMI-1 plant Emergency Procedure 1202-26A (Lic. Ex. 49). The exact steps depend upon the reason why no feedwater is available, and generally consist of verifying that valves are in the correct position, verifying that the pumps have started and taking manual actions where pump or valve actuation have not occurred correctly. Keaten et al., ff. Tr. 16,552, at 11, 12.

431. Assuming emergency feedwater is made available, the steam generator can be restored as a heat sink by adding

emergency feedwater to the steam generator(s), and relieving steam through one or both atmospheric dump valves or through the turbine bypass valves to the condenser. These pumps and valves are normally operated from the control room but the valves can also be operated locally, and the steam-driven emergency feedwater pumps can be started locally. With the steam generator in operation, primary system temperature can be reduced below system saturation temperature and a 50°F sub-cooling margin will be maintained or reestablished. High pressure injection can then be throttled, and a bubble can be formed in the pressurizer by energizing pressurizer heaters and reducing high pressure injection flow to allow the PORV or primary safety valve(s) to close. The normal makeup system can be used. Once the bubble has been reformed in the pressurizer, the plant has been returned to a normal shutdown condition and cooldown may continue using normal plant cooldown procedures. Keaten et al., ff. Tr. 16,552, at 12.

Part III of Board Question 6

432. The original Board Question 6 concluded as follows:

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

The Board's concerns with emergency feedwater reliability were inspired to a great extent by the Atomic Safety and Licensing

Appeal Board's decision in Florida Power and Light Company (St. Lucie Nuclear Power Plant, Unit No. 2), ALAB-603, 12 N.R.C. 30 (1980), review pending, CLI-80-41, 12 N.R.C. ____ (December 12, 1980).

433. As a part of its review of licensing board decisions authorizing the issuance of a construction permit for St. Lucie-2, the Appeal Board conducted an evidentiary hearing on the adequacy of electric power systems:

> Because of Florida's peninsular shape the applicant's electrical distribution system (grid) can be connected with the grids of other utilities only to the north. This suggested -- and the applicant's operating history tended to confirm -- that FP&L's grid might be less reliable than ones interconnected with multiple grids. There was no indication, however, that the onsite emergency power system at St. Lucie had been designed to compensate for a lesser degree of grid stability and the Licensing Board had no occasion to explore the matter.

ALAB-603, supra, 12 N.R.C. at 31. Consequently, the Appeal Board sought from the parties certain information and advice as to whether further proceedings were necessary. Id. at 33.

> The substantial amount of information submitted by the parties convinced us that an evidentiary hearing was needed to explore our questions about the stability of Florida Power and Light's electrical grid and the reliability of AC power for St. Lucie Unit 2. We had several particular concerns: (a) the implications of then recent grid disturbances (including a complete loss of offsite power on May 14, 1978); (b) the staff's opinion that offsite power was less assured for St. Lucie than for nuclear plants in nonpeninsular areas, and (c) the lack of compensation for that situation in the design of the onsite power

> system. We therefore ordered a hearing held before us on those concerns and directed the parties to answer additional questions in preparation for it.

Id. at 34 (footnotes omitted).

434. The Appeal Board, on the basis of the evidence presented, found that the likelihood of the loss of all AC power at St. Lucie-2 is the product of two factors: (1) the probability of an off-site power failure (found to be between 0.1 and 1.0 pe. year) and (2) the probability of a simultaneous failure of both diesel generators to start on demand (found to be 10^{-4}, at best, assuming true independence of the two diesel generator systems). This yielded a combined probability in the range of 10^{-4} to 10^{-5} per year. Id. at 45.

435. Rejecting arguments that the assumed simultaneous failure of both diesel generators challenges the "single failure criterion," the Appeal Board found, on the basis of failure rate data presented in the Reactor Safety Study, WASH-1400 (the Rasmussen Report), that diesel generators are relatively unreliable pieces of equipment, compared to other equipment to which the single failure criterion is commonly applied, and that "[b]lind reliance on the single failure criterion (that is, simple redundancy) does not provide an adequate degree of plant safety and public protection in this state of affairs." Id. at 48-52.

436. The Appeal Board compared the probability range for station blackout at St. Lucie-2 (10^{-4} to 10^{-5} per year)

with certain guidelines in the NRC Staff's Standard Review Plan ("SRP") for determining whether particular accidents should be considered in designing a plant, even though the Staff testified that it had no numerical reliability goals for station blackout. The SRP, according to the Appeal Board, provides that events must be considered in the design where they have: (1) a realistically calculated probability of occurrence of at least 10^{-7} per year, or (2) a conservatively calculated probability of 10^{-6}. Accordingly, the Appeal Board found that the probability of a loss of all AC power is unacceptably high relative to accidents and other events considered incredible for design purposes.[137] Id. at 45-46, and 52.

437. As the Appeal Board acknowledged, and as the Commission subsequently observed, SRP section 2.2.3, used in ALAB-603 as some sort of benchmark for assessing whether events are "design basis,"[138] deals specifically with Staff reviews of

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137 With respect to the specific event postulated in St. Lucie -- loss of all AC power -- the testimony shows that the TMI-1 EFW system will perform during a two-hour station blackout. In addition, it should be noted that the high probability of the loss of off-site power at St. Lucie, 0.1 to 1.0 per year, does not exist at Three Mile Island. TMI has not experienced any loss of off-site power, and, based upon its multiple power feeds, it is not expected that such an event will occur during the life of the plant. Even in the highly improbable event that a loss of off-site power would occur, and that both diesel generators failed, AC power for Three Mile Island can be obtained from off-site combustion turbines within two hours. Capodanno et al., ff. Tr. 5642, at 13, 14.

138 "Design bases" means that information which identifies the specific functions to be performed by a structure, system, or component of a facility, and the specific values or ranges (continued next page)

certain off-site hazards and the need for any protective measures. See id. at 45, n.53; CLI-80-41, supra, slip op. at 1. One of the generic issues in ALAB-603 which the Commission has set for review is the following:

> What are the generic implications of using the threshold probabilities in Section 2.2.3 of the Standard Review Plan as guidelines in determining the design basis events to be used for plant design and operation?

CLI-80-41, supra, slip op. at 3 (footnote omitted).

438. In a memorandum issued after CLI-80-41, supra, the Appeal Board advanced its view that the question posed above by the Commission is not presented by, and is inspired by a misconstruction of, ALAB-603. The Appeal Board stated that it was the very magnitude of the probability values, independently assessed by the Appeal Board from the evidentiary record, which served as the basis for its ultimate determination that the station blackout sequence must be considered as a design basis event. Florida Power and Light Company (St. Lucie Nuclear Power Plant, Unit 2), Appeal Board Memorandum (December 22, 1980), Docket No. 50-389.

439. Consequently, ALAB-603 now appears to have limited precedential value for other plants and other plant

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(continued)
of values chosen for controlling parameters as reference bounds for design. These values may be (1) restraints derived from generally accepted "state of the art" practices for achieving functional goals, or (2) requirements derived from analysis (based on calculations and/or experiments) of the effects of a postulated accident for which a structure, system, or component must meet its functional goals. 10 C.F.R. § 50.2(u).

systems. The Appeal Board's investigation was inspired by a very unique circumstance -- operating experience which confirmed a suspicion that St. Lucie is more vulnerable to loss of off-site power than nuclear power plants in non-peninsular areas. This was compounded by a second unusual circumstance, a finding based on WASH-1400 estimates that diesel generators were sufficiently unreliable to warrant a deviation from the single failure criterion. ALAB-603 apparently does not stand for the proposition that a failure probability of 10^{-6} per year should be used generically to classify a scenario as a design basis event to be used for plant design and operation. Indeed, the NRC Staff has testified in this proceeding that SRP section 2.2.3, referred to in ALAB-603 as the criterion for acceptability of the plant design to mitigate the assumed event, was intended by the Staff to be applied only to external plant hazards such as nearby transportation of toxic gases or explosives, and not to events within the plant such as a postulated loss of emergency feedwater. Wermeil et al., Tr. 6035, at 10. See also, Rosenthal and Check, ff. Tr. 11,158, at 21. The NRC has not yet established a numerical safety goal. Wermeil et al., ff. Tr. 6035, at 10; Rosenthal and Check, ff. Tr. 11,158, at 27.

440. The Board, however, was concerned at the time some of the witnesses appeared (November, 1980; prior to the Commission's expression of concern with, and the Appeal Board's clarification of, the St. Lucie decision), with the testimony

that neither Licensee nor the NRC Staff has a quantitative reliability goal against which the TMI-1 emergency feedwater system has been compared. Tr. 5789-98, 5948 (Capodanno); Tr. 6168, 6178 (Wermeil). Applying, by way of analogy, the Appeal Board's analysis in ALAB-603 of the St. Lucie electrical power system, Administrative Judge Jordan postulated, at the hearing session of November 20, 1980, that:

a. B&W plants are more sensitive because of the once-through steam generator design and they experience an unusually high EFW challenge rate of three per year. Tr. 6150, 6175, 6179-80 (Administrative Judge Jordan). This could be viewed to be analogous to St. Lucie's vulnerability to loss of off-site power.[139]

b. Emergency feedwater systems, on an industry-wide basis, have experienced a failure rate of 1 in 25 per reactor-year -- which is so high that reliance on safety-grade criteria should be rejected. Tr. 6169, 6179-80, 6182-83 (Administrative Judge Jordan). This could be viewed to be analogous to the Appeal Board's findings, in St. Lucie, on diesel generator reliability and its deviation from the single failure criterion.

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139 The Appeal Board's concern at St. Lucie, however, was inspired by evidence of actual experience at that specific plant with instabilities in off-site power supplies. We now know from the record here that there is no evidence that TMI-1 has experienced an unusual challenge rate to its EFW system. See paragraph 446, infra. Consequently, the Board now recognizes that the postulated analogy is not valid.

c. Consequently, it should be demonstrated that the overall reliability of the TMI-1 decay heat removal systems is such that the probability of failure is less than 10^{-6} per year. Tr. 6184, 6186-87 (Administrative Judge Jordan).

441. While it was prudent for the Board to have expressed its concerns, the evidence subsequently presented convinces us now that the St. Lucie analogy, to the extent that Appeal Board decision even stands for the proposition we read into it in November, 1980, is not valid for the TMI-1 emergency feedwater (or overall decay heat removal) system. First, the linkage postulated between the B&W design sensitivity and EFW challenge rates is not sound. The primary difference between the B&W nuclear steam supply system ("NSSS") design and other PWR designs is the B&W once-through steam generator ("OTSG"), which results in a more rapid effect (compared to the U-tube steam generator) on primary system performance from any large change in secondary system inventory. In addition, the volume of water on the secondary side of a plant with the U-tube design is larger than the comparable inventory in a B&W plant. The close coupling of the primary and secondary systems in the B&W design, combined with the relatively small liquid volume in the secondary side, creates the characteristic of the OTSG referred to as "sensitivity" or "responsiveness," which has been considered in the safety analyses for TMI-1. While the design and operating characteristics of the OTSG are important and must be understood, they have little bearing on the

question of how often main feedwater is lost or the emergency feedwater system is challenged. Keaten, ff. Tr. 16,612, at 4, 5. In other words, the OTSG design affects the dynamics of primary system response to a secondary system upset, but it has absolutely no bearing on the frequency of secondary system upsets. See Tr. 15,772-73 (Capra).

442. Further, it is now clear to the Board that there are serious limitations to a comparison of the operating experience with the TMI-1 EFW system to the experience at other B&W plants, or on an even broader basis. The design of the main and emergency feedwater systems are normally the responsibility of the architect/engineer ("A/E"), rather than the NSSS supplier. These designs vary widely, reflecting the different views of the A/E or owner, the NSSS supplier control system interface needs, and the type of main turbine-generator chosen. There are several significant issues that impact the number of loss of main feedwater transients reported, and the significance of partial losses of main feedwater. These, in turn, are important in considering the challenges to the emergency feedwater system. Keaten, ff. Tr. 16,612, at 5.

443. The degree of redundancy built into the feedwater-condensate train and the normal controls of redundant components directly influence the number of transients. Systems that have no redundancy in components will either trip or runback if any of the major components in the train are lost. TMI-1 has redundant components (three) in the condensate, condensate booster and heater drain pumps such that the

standby pump should start if one of the two operating pumps fail, without a resulting trip or runback. If one of the two main feedwater pumps trip, the plant will runback or trip. Keaten, ff. Tr. 16,612, at 5, 6.

444. In the event of a trip, the residual heat of the reactor is normally removed via the steam generators. The method of feedwater makeup, post-trip, is also a function of the A/E, owner, NSSS supplier interfaces. Many (if not most) non-B&W NSSSs are designed such that on a turbine trip or reactor trip, the post-trip supply is usually from the emergency (or auxiliary) feedwater system. Additionally, these systems operate for normal plant startups and shutdowns. The TMI-1 design is capable of providing the necessary low flow requirements during post-trip, startup and shutdown with the main feedwater system. Only in the event of a total loss of the main feedwater system is the emergency feedwater system required. Therefore, the number of actual demands on the TMI-1 system is substantially lower than for other designs. Keaten, ff. Tr. 16,612, at 6.

445. Nevertheless, in November, 1980, the Board used NUREG-0560, "Staff Report on the Generic Assessment of Feedwater Transients in Pressurized Water Reactors Designed by the Babcock & Wilcox Company" (May 7, 1979), as a basis for the postulation that the challenge rate to EFW systems at B&W plants is approximately three per year. Tr. 5971 (Administrative Judge Jordan). NUREG-0560 documents an NRC Staff study,

conducted shortly after the accident at TMI-2, to assess the effect of feedwater transients on B&W reactors. For the one-year period from March, 1978, to March, 1979, the Staff reports that there were 9 B&W plants that had 27 "feedwater transients" -- or 3 per year, per plant. While on closer scrutiny the Board cannot confirm this figure from the incident chronology in NUREG-0560, the important error in our use of this data was to construe "feedwater transient" as a challenge to the EFW system. The record shows that in NUREG-0560 the Staff was not reporting an EFW challenge (or demand) rate of 3 per year at B&W plants. Rather, it was reporting cases where forced plant shutdown resulted from some feedwater system malfunction. Thus, the feedwater transient frequency of 3 per B&W plant per year, reported in NUREG-0560, does not represent the frequency of demands upon the EFW system at TMI-1, or at B&W plants generally.[140] Keaten, ff. Tr. 16,612, at 7, 8.

446. Further, the record shows that for the calendar years 1978 and 1979, the frequency of main feedwater losses

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140 The study documented in NUREG-0560 has been described by Staff personnel as "cursory in nature," designed to see if "a vast difference" in feedwater related malfunctions existed for the various vendors. While the Staff found a somewhat larger number of transient events for B&W plants, it was not felt to be an appreciably higher frequency than for the other vendors. The Staff also expressed the thought that the greater number of feedwater transients may have been due to the generally younger age of B&W plants. The somewhat greater frequency of feedwater related transients was not by itself, however, considered by the Staff to be a safety concern. Keaten, ff. Tr. 16,612, at 8.

reported at B&W plants was substantially less than 3 per year per plant (i.e., 0.3). Keaten, ff. Tr. 16,612, at 9. From January, 1979, through August, 1980, there were no instances where the emergency (or auxiliary) feedwater system at any pressurized water reactor was incapable of performing its essential functions. Koppe, ff. Tr. 13,335, at 40. In addition, Staff witness Ross informed us that a more recent Staff survey shows that the arrival rate of feedwater transients is not dependent upon the NSSS design -- i.e., B&W plants are no more likely to have a feedwater transient than Westinghouse or Combustion Engineering plants. Tr. 15,769-70 (D. Ross). We now view that the EFW system experience at TMI-1 is a much more appropriate tool to use here than the statistically temperamental data on systems at other plants. The unavailability of the TMI-1 emergency feedwater system was zero for five years of operation.[141] Koppe, ff. Tr. 13,335, at 41. TMI-1 has not experienced a loss of main feedwater transient, or a total loss of feedwater, during its operating history. Keaten, ff. Tr. 16,612, at 9; Tr. 6175-76 (Wermeil).

447. The basis for the Board's postulation that the probability of failure of an EFW system is 1 in 25 per reactor-year was testimony by NRC Staff witness Lantz, during

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141 The reliability of the TMI-1 EFW system also has been demonstrated by ten manual EFW initiations which exhibited no component failures, and by surveillance testing of individual components which did not reveal conditions in excess of allowable technical specification limits. Capodanno et al., ff. Tr. 5642, at 12.

cross-examination, that Licensee Event Reports indicate 8 failures of safety-grade EFW systems in 200 reactor-years. Tr. 6093-94 (Lantz). Mr. Lantz also testified that for all plants there were 9 EFW failures in 280 reactor-years. Tr. 6106-08 (Lantz).

448. There are several reasons why this data is not useful and cannot be tied to an EFW demand rate. First, Mr. Lantz was not reporting EFW system failures upon demand. He was providing routine LER data on system availability. This includes testing experience. Keaten, ff. Tr. 16,612, at 10. In fact, NRC Staff witnesses have testified that data on EFW system success on demand is not available. Wermeil et al., ff. Tr. 6035, at 3, 4. Consequently, the data tells us nothing about the probability that the TMI-1 EFW system will fail if it is called upon or demanded. Keaten, ff. Tr. 16,612, at 10. In short, failure data without accompanying demand data cannot yield a number for EFW failures upon demand. Tr. 16,693-94 (Keaten).

449. Second, the data is not applicable to TMI-1. Four of the eight failures reported by Mr. Lantz involved normal start-up operations. Tr. 6095-96 (Lantz). In contrast, the EFW system at TMI-1 is not normally used for plant startup or shutdown. The data also includes failures of associated systems which are not found at TMI-1. Keaten, ff. Tr. 16,612, at 10; Tr. 6136-37 (Wermeil). This illustrates the severe limitations on the application of industry-wide EFW system

experience to a specific plant. Just as EFW system demand frequency is a function of plant specific factors, EFW system availability and operation upon demand is dependent upon plant specific EFW components, EFW support services, component and system testing, and maintenance. Keaten, ff. Tr. 16,612, at 10. The Staff testified that the historical data would indicate that the TMI-1 EFW system is more reliable than the average. Tr. 6219 (Wermeil). For all of these reasons the Board now finds that there is no sound basis upon which to attribute to the TMI-1 emergency feedwater system a failure rate of 1 in 25 per reactor-year.[142]

450. The NRC has used principally deterministic criteria, supplemented by elements of probabilistic analysis, in licensing nuclear power plants and in judging the acceptability of a plant system. Rosenthal and Check, ff. Tr. 11,158, at 17-18, 20-26; Tr. 11,200-02 (Check, Rosenthal); Tr. 11,253 (Check). See also, section II.S (Accident Design Bases), infra. The Board acknowledged early on that use of a probabilistic analysis in conjunction with some numerical acceptance criterion, in answer to Board Question 6, would not be necessary in the absence of a special situation. Tr. 6187-88 (Administrative Judge Jordan). See also, paragraph

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142 While at this point we are not terribly interested in pursuing the futile exercise of sorting out numbers, the Board notes that others have reported availability figures of 10 to the minus 5 (Koppe) and 10 to the minus 3 (B&W). Tr. 16,674-76 (Keaten). See paragraph 459, infra.

513, infra (probabilistic techniques should not be relied upon as the sole basis for regulatory decisions). We now see that there is no special situation with the emergency feedwater system at TMI-1 -- either in terms of the rate of challenge to the system or in terms of the likelihood that it will fail.[143] See Keaten, ff. Tr. 16,612, at 11. Consequently, the Board now finds that the St. Lucie decision, ALAB-603, supra, to the extent that it has any precedential value beyond the plant specific record developed there, should not be applied to require a quantitative assessment of the reliability of the TMI-1 emergency feedwater or decay heat removal systems, or to require that loss of emergency feedwater be considered to be a design basis event at TMI-1.

451. Nevertheless, in response to the concerns expressed by the Board early in the proceeding, the Staff presented testimony which reports the results of a quantitative estimate of EFW system reliability at TMI-1. Estimates provided were for the TMI-1 EFW system as it existed in mid-1979, at the time of restart, and after the full safety-grade modifications are completed. See, generally, Wermeil and Curry, ff. Tr. 16,718, at 31-42; Tr. 16,732-34 (Curry). We understand that the Staff provided these numbers specifically

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143 At TMI-1, EFW system reliability at restart will be comparable with some other operating plants. Indeed, it is not inconsistent with the industry average estimate based on a Licensee Event Report survey. Tr. 16,722 (Curry).

in response to the Board's request. Tr. 16,740 (Curry). At the same time, the Staff witness candidly acknowledged the serious limitations on the exercise he had performed, and warned us of the danger in misusing, or placing too much reliance upon, his results.

452. First, the analysis was conducted to estimate the reliability of the EFW system in a five-minute period after the occurrence of the transient, and mission success was defined as delivery of required EFW flow to the steam generator(s) within five minutes. The Staff chose a five-minute time period because of the estimated time for steam generator dryout in a B&W plant if no feedwater is provided. The Staff believes that, in terms of plant operation, steam generator dryout is significant due to the unstable system condition it induces. Wermeil and Curry, ff. Tr. 16,718, at 32, 33. The Staff also acknowledged that it chose this success criterion (i.e., avoidance of steam generator dryout) in order to make the study consistent with those for other PWRs, even though it is a slightly more severe criterion for B&W plants. Tr. 17,068 (Curry). Licensee points out, however, that the absence of EFW flow for five minutes does not result in core damage, and that the significance of dryout at B&W plants is not necessarily the same as for other plants since EFW is sprayed into the steam generator at a very high point and immediately starts to cool the primary system when it is re-established. Tr. 16,613-15 (Keaten).

453. A major implication of the reliability estimate for a five-minute period is that the number and type of operator actions that may be expected to be accomplished to rectify an EFW system fault is very limited. Consequently, system reliability becomes largely a function of the probability of the system being in the proper configuration at the time of demand, and the inherent reliability of mechanical and electrical components to function on demand. Wermeil and Curry, ff. Tr. 16,718, at 33. This analysis really just characterizes the EFW system's innate reliability as a function of its hardware reliability. It does not recognize (or gives essentially no credit for) improved operating procedures and operator training. Tr. 16,744-46 (Curry); Tr. 16,700-02 (Keaten). Neither does it credit the hardware changes made at TMI-1 to facilitate operator action to recover feedwater. Tr. 17,016 (Curry). Yet, the need to improve procedures and training is the main lesson learned from the TMI-2 accident. Tr. 16,701 (Keaten). As a general rule, the Staff believes that consideration of operator recovery actions would certainly improve the reliability. Tr. 16,940 (Curry).

454. Because of the smaller inventory of B&W steam generators, dryout would occur much sooner if all feedwater were lost than would occur under similar circumstances for a Westinghouse steam generator. This results in a more stringent response requirement for an emergency feedwater system associated with a B&W NSSS than one associated with a Westinghouse

NSSS because significantly less reliance on operator intervention to rectify system faults can be credited for the B&W response than the Westinghouse. Thus, again, the selection of steam generator dryout prevention as the benchmark for successful EFW system operation influenced the analysis, and places some bias against the B&W design.[144] Wermeil and Curry, ff. Tr. 16,718, at 41; Tr. 16,741, 17,075-76 (Curry).

455. A Staff witness testified that if the Staff had used a more realistic mission success criterion -- such as the capability of the EFW system to deliver minimum feedwater flow for mitigating a transient -- the potential bias associated with the criterion could have been corrected. He expressed confidence that if the Staff had used this sounder basis for comparison, the upgraded TMI-1 EFW system would have looked very similar to the Westinghouse plants. Tr. 17,080 (Wermeil). Accord, Tr. 17,068, 17,095 (Curry).

456. Second, as with any other such analysis, the construction of the fault tree here was limited by the resolution of available data or by the level of system detail understood by the analyst. Wermeil and Curry, ff. Tr. 16,718, at 33. The Staff used industry-wide averages for component

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144 The analysis for a five-minute period is also influenced by the fact that B&W systems utilize only two steam generators, rather than the three or four present in Westinghouse designs. If the analysis had been conducted to consider prevention of core uncovery, sizes or numbers of differing steam generators may have taken on less importance. Wermeil and Curry, ff. Tr. 16,718, at 41, 42; Tr. 17,080-81 (Wermeil).

failure and human error rates to estimate the reliability of the TMI-1 emergency feedwater system since plant specific data is limited. Id. at 38. When the fault tree development was limited by the level of detail available to the Staff, conservative reliability assumptions were made. Id. at 34. See also, paragraphs 442-444, supra (limitations on industry-wide comparisons of EFW systems).

457. The third limitation of the analysis, and the most important one to keep in mind, is that it represents a rough assessment of the potential of the EFW system to accomplish a given mission under given conditions. See Wermeil and Curry, ff. Tr. 16,718, at 39. There is not necessarily a perfect correlation between the comparative reliabilities of various plant auxiliary systems and the comparative risk associated with the operation of those plants. To draw conclusions about the comparative risks of operating various nuclear plants, consideration needs to be given to the integrated response of all plant systems to cope with potential transients and loss-of-coolant accidents. Id. at 39, 40; Tr. 16,722 (Curry). This point was consistently pressed upon us by a number of witnesses. See, e.g., Tr. 4822 (Jones) (need to consider the reliability of the main feedwater system and feed-and-bleed operation, in addition to the EFW system); Tr. 16,674-76 (Keaten) (if one is looking for a number that represents the probability of major core damage occurring due to lack of heat removal, one would have to multiply the

frequency of demands on the EFW system times the probability that the EFW system would not be available, times the probability that no other method of heat removal would be available); Tr. 16,748 (Curry) (analysis does not represent overall probability of core damage); Tr. 17,079 (Curry) (plant risk should take into account not only the fact that successful EFW system operation will occur even if flow is secured much later than five minutes, but also the fact that TMI-1 is equipped with a feed-and-bleed mode of operation which can successfully cool the core).

458. Nevertheless, the Staff witness expressed his judgment that with the EFW reliability estimate he presented, and based upon his knowledge of the additional system reliabilities to be considered in a sequence that would lead to core damage and his familiarity with reliability analyses of other plants, the probability of core damage at TMI-1 is less than or certainly no greater than in all other operating plants, and that it is not inconsistent with the numerical safety goals now under consideration by the Commission. Tr. 17,089-92 (Curry).

459. Prior to the Staff's reliability evaluation prepared in response to Board Question 6, an evaluation of the reliability of the TMI-1 EFW system as it existed in mid-1979 had been performed in 1979 for Licensee by Babcock & Wilcox.[145]

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145 B&W performed reliability analyses of all operating B&W plant emergency feedwater systems in response to a Staff (continued next page)

Lic. Ex. 15 at 9. The B&W analysis considered the same transients the Staff considered, but also considered reliability for the time periods 5, 15 and 30 minutes. The Staff testified that because of the more detailed design and operational information used by B&W for its analysis, that analysis was necessarily more rigorous and more detailed than the conservative type of analysis performed by the Staff as a check. Wermeil and Curry, ff. Tr. 16,718, at 36; Tr. 17,022-23 (Curry). While the B&W analysis was not intended to establish a num ical reliability value for the TMI-1 EFW system, it did compare the 1979 system with those at Westinghouse and Combustion Engineering plants and found that the TMI-1 system fell in the mid-range.[146] Tr. 5948, 5984-85 (Capodanno); Tr. 6157-59 (Wermeil). The Staff reviewed this study and approved its methodology and results. Wermeil and Curry, ff. Tr. 16,718, at 5.

460. While the Board now is uncertain of the value of the actual numbers produced by these evaluations, we believe the analyses are adequate to accomplish their intended purpose, which is:

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request of the utilities to do so. Tr. 6158 (Wermeil); Wermeil and Curry, ff. Tr. 16,718, at 36; Tr. 16,766-67 (Curry).

146 It should be noted that the original B&W analysis and the Staff review of it assumed that both electric-driven pumps were required for successful EFW system operation. Later analysis, however, indicates that only one electric driven EFW pump is needed for successful heat removal. Wermeil and Curry, ff. Tr. 16,718, at 36.

> the assessment of the reliability of a given auxiliary feedwater system compared to other designs and the identification of major contributors to a given auxiliary feedwater system unreliability so that system upgrading can be most effectively undertaken, if desired.

Wermeil and Curry, ff. Tr. 16,718, at 39; Tr. 6134-35 (Wermeil). In fact, the 1979 B&W analysis identified the major contributors to TMI-1 EFW system unavailability, and led directly to several of the restart modifications to the system design and to plant procedures. See Lic. Ex. 15 at 9, 10; Wermeil et al., ff. Tr. 6035, at 10; Wermeil and Curry, ff. Tr. 16,718, at 4, 5.

461. Prior to the B&W analysis, Licensee itself performed a re-evaluation of the TMI-1 EFW system design and operation in order to determine where upgrades in the timeliness and reliability of the system could be made. This evaluation resulted in eight items that the Staff agreed would result in improvement to the EFW system reliability, and that were subsequently included in short-term action 1(a) of the Commission's Order and Notice of Hearing. Wermeil and Curry, ff. Tr. 16,718, at 3. Subsequent to the B&W study, four additional short-term recommendations were developed based on the Lessons Learned Task Force review and the B&O Task Force review of B&W operating plants. Id. at 5. As a final approach to re-examining the reliability of EFW systems in operating plants, the B&O Task Force performed a comparison of the EFW system designs against the current Standard Review Plan

criteria for a safety-grade sysetm in order to provide further insight into possible areas for improvement that were not identified in previous evaluations. The EFW system review effort was later consolidated into the NRC TMI Action Plan. Id. at 11.

462. Staff witness Wermeil provided the Board with a detailed discussion of the evolution of the Staff's criteria related to the EFW system, and of the manner in which system reliability has been improved as a result of implementation of these criteria at TMI-1. See, generally, Wermeil and Curry, ff. Tr. 16,718, at 1-30; Tr. 16,719 (Wermeil). The EFW system review effort since the TMI-2 accident is substantially more detailed and exhaustive than the Staff's standard deterministic evaluation against the acceptance criteria of the Standard Review Plan. Based upon its review and evaluation of the requirements and of Licensee's compliance with them in terms of the resulting hardware, procedural and technical specification changes to be implemented, the Staff has concluded that the TMI-1 emergency feedwater system meets the requirements identified for implementation at the time of restart, and that with these changes the system will be sufficiently reliable to allow restart. Wermeil and Curry, ff. Tr. 16,718, at 12; Tr. 17,017 (Wermeil).

Conclusion

463. The Board now agrees with Licensee and the Staff that the acceptability of a design should not be based

exclusively on a numerical estimate of its reliability. <u>See</u> Tr. 17,026 (Curry); paragraph 513, <u>infra</u>. There is no special circumstance here which would warrant the Board's reliance upon, or need for, such an estimate in the case of TMI-1. We have learned that there is no basic difference between B&W plants and other PWRs in protecting against a loss of main feedwater transient, Tr. 17,064 (Wermeil), and that operating history shows that the arrival rate for feedwater transients does not depend upon the NSSS design. Tr. 15,769 (D. Ross). That is, B&W plants are no more prone to have feedwater transients than are other PWRs. Tr. 15,770 (D. Ross). Further, there is no reason to suspect that EFW systems at B&W plants are less reliable than at other plants. Tr. 16,687-88 (Keaten); Tr. 17,068-69 (Curry). Yet, many other PWRs do not have the back-up feed-and-bleed cooling capability which exists at TMI-1. Tr. 17,064 (Wermeil).

464. Experience with the TMI-1 main and emergency feedwater systems has been excellent. There have been no failures of the EFW system on demand, nor have there been any total loss of main feedwater events (other than required tests) which would challenge the EFW system. Thus, the TMI-1 design has provided a stable, reliable main feedwater system that is capable of all normal operating and shutdown feedwater services, and it has in reserve a reliable EFW system fully capable of performing the necessary services under abnormal transient or accident conditions. Keaten, ff. Tr. 16,612, at

11. The historical data indicates that the TMI-1 EFW system has been more reliable than the average. Tr. 6219 (Wermeil).

465. The Board is impressed that the Staff EFW system review effort has employed a number of diverse evaluation techniques, including fault-tree probabilistic analysis, to identify design and procedural improvements to the reliability of the TMI-1 emergency feedwater system. While the TMI-1 emergency feedwater system appears to have been a very reliable system previously, the modifications identified by the Staff and Licensee as a result of this intensive review effort will further enhance the reliability of the EFW system. <u>See</u> paragraphs 395-401, <u>supra</u>. <u>See also</u>, paragraphs 498-505, <u>infra</u> (Staff event tree analysis of a loss of main feedwater transient and events initiated by a loss-of-collant accident). No party has challenged the necessity or sufficiency of the modifications to the TMI-1 EFW system proposed in the Commission's Order and Notice of Hearing, and all of the witnesses who testified in response to Board Question 6 expressed the view that the system will be sufficiently reliable to support restart of the plant. Based upon the foregoing detailed findings of fact, and the uncontroverted and exhaustive evidentiary record we have reviewed on this subject, the Board agrees with that assessment.

466. The Board finds that the short-term actions recommended in the Commission's Order and Notice of Hearing to improve the timeliness and reliability of the TMI-1 emergency

feedwater system are necessary and sufficient to provide reasonable assurance that the facility can be operated without endangering the health and safety of the public, and should be required before resumption of operation should be permitted. The Board also finds that the recommended long-term actions with respect to the EFW system are necessary and sufficient to provide reasonable assurance that the facility can be operated for the long term without endangering the health and safety of the public, and should be required of Licensee as soon as practicable.

R. Valve Testing

Board Question/UCS Contention No. 6:	Reactor coolant system relief and safety valves form part of the reactor coolant system pressure boundary. Appropriate qualification testing has not been done to verify the capability of these valves to function during normal, transient and accident conditions. In the absence of such testing and verification, compliance with GDC 1, 14, 15 and 30 cannot be found and public health and safety is endangered.
Board Question Regarding UCS Contention 6:	The board wants more than just a schedule for testing of reactor coolant system safety and relief valves, as is required pursuant to NUREG-0578. Is there reasonable assurance that the tests will be successful, e.g., that there is good evidence that the valves will indeed perform in an accident environment?

467. In its Contention No. 6, UCS had alleged that appropriate qualification testing had not been performed to verify the capability of reactor coolant system relief and safety valves. UCS withdrew its sponsorship of Contention No. 6 on July 31, 1980. Subsequently, the Board not only retained the contention as a Board question, but also posed its own question, quoted above, regarding the former UCS Contention No. 6. Consequently, an evidentiary record on the qualification testing of reactor coolant system relief and safety valves was compiled only because the Board, in the exercise of its discretion, chose to pursue the issue.

468. Former UCS Contention No. 6 asserted that appropriate qualification testing has not been performed to verify the capability of the reactor coolant relief and safety valves. We have already noted that the pressurizer safety valves are part of the reactor coolant pressure boundary and functionally provide overpressure protection for the reactor coolant system. The valves were designed for, and protect the integrity of the reactor coolant system at, the design conditions of the primary system -- 2500 psig and 670°F. The reactor coolant system is adequately protected by either of the two safety valves, since each is capable of relieving the required capacity. Correa et al., ff. Tr. 8746, at 5 (Urquhart).

469. The relief capacity of the safety valves was established consistent with the applicable edition and addenda

of Section 9 of Section III of the ASME Boiler and Pressure Vessel Code. This included certification by the valve manufacturer of the capacity of the valves utilizing prototypical testing to establish discharge factors and analytical verification of the ability of the valves to withstand design and operating pressures. Correa et al., ff. Tr. 8746, at 5 (Urquhart).

470. The safety valves were also designed in accordance with the requirements of Section III of the ASME Code to assure reactor coolant pressure boundary integrity. Testing and examination of the valves during and following manufacturing and testing included the following:

(a) Chemical and mechanical testing of the materials.

(b) Volumetric examination of the materials.

(c) Surface examination of the materials.

(d) Hydrostatic pressure testing of the completed valves at the manufacturer and after installation.

(e) Verification of set pressure.

(f) Seat leakage testing following opening and closing.

Correa et al., ff. Tr. 8746, at 5, 6 (Urquhart). See also, Zudans, ff. 8824, at 4, 5.

471. Also of significance with regard to the capability of the pressurizer safety valves is the transient which occurred February 26, 1980, at the Crystal River nuclear unit, a plant with a B&W nuclear steam supply system and

components similar to TMI-1. During the transient, one of the two safety valves lifted at approximately 2400 psig and flowed saturated steam, two-phase fluid and liquid water. The water flow rate was up to 700 gpm and the valve reseated at approximately 2300 psig, a blowdown of about 4% below the opening pressure. Correa et al., ff. Tr. 8746, at 6 (Urquhart).

472. Subsequent to the transient, the affected valve was subjected to detailed laboratory inspection and testing to determine if any damage had been sustained. The set pressure of the valve was checked three times and determined to be approximately the 2400 psig experienced during the transient. Leakage was measured at about 1.1 gpm. Disassembly and inspection identified steam cutting of the valve disc and a damaged bellows assembly. The steam cutting was most likely caused by leakage that was present prior to the transient. The damage to the bellows did not appear to be due to the February 26, 1980 transient. Neither the steam cutting of the disc nor the damaged bellows impaired the intended pressure relief function of the valve. In summary, no damage detrimental to the proper operation of the valve was discovered even though it had experienced flow conditions other than saturated steam. Correa et al., ff. Tr. 8746, at 6, 7 (Urquhart); Tr. 8787-88, 8806, 09 (Urquhart); Zudans, ff. Tr. 8824, at 7.

473. The pressurizer PORV was designed for the same system conditions as the safety valves -- 2500 psig and 670°F. The valve design was governed by the same ASME Code requirements as the safety valves as it related to pressure boundary

integrity, and the valve was tested and examined in a manner similar to the safety valves. Because the PORV is power operated in response to an independent pressure signal, verification of set pressure was not applicable. Verification of valve opening and closing was performed, however, both prior to shipment and following installation. Correa, et al., ff. Tr. 8746, at 7 (Urquhart); Zudans, ff. Tr. 8824, at 5. The PORV is seismically qualified, and its solenoid operator is qualified for up to 300°F and 2 x 10^8R. The PORV block valve is environmentally and seismically qualified, as is its control circuitry. The control circuitry for the PORV itself is environmentally qualified. Tr. 8768 (Correa); Tr. 8800-01, 8997 (Urquhart).

474. The PORV which will be installed in TMI-1 prior to restart is the TMI-1 spare PORV. This valve was ordered per the original PORV requirements, was manufactured in 1978, was "N" stamped per Code Case 1581, and in general satisfies the 1977 Edition with the Winter 1979 Addendum of Section III of the ASME B&PV Code for fabrication requirements. Correa et al., ff. Tr. 8746, at 8.

475. The valve is being modified per the manufacturer's latest design features to improve seat tightness. The modification is being performed per the latest ASME B&PV Code, Section III, requirements. As part of the modification effort, the valve will be disassembled and all critical dimensions will be recorded and checked against drawing requirements. In addition, all moving parts will be inspected for surface finish and

signs of wear caused by the original testing of the valve prior to its shipment in 1978. This inspection of the valve internals will ensure that the valve parts meet all requirements. After reassembly of the valve, it will be seat leak tested and opened at its setpoint. This will ensure that the valve will function properly. Correa et al., ff. Tr. 8746, at 8; Tr. 8809-10 (Correa).

476. Prior to being installed in TMI-1 the valve will again be seat leak tested. During hot functional testing the valve also will be actuated to ensure its functional ability and to test all downstream instrumentation. Correa et al., ff. Tr. 8746, at 9.

477. A valve testing program is also in progress in response to recommendation 2.1.2 of NUREG-0578. The performance testing of PWR relief and safety valves is being conducted by the Electric Power Research Institute (EPRI). Licensee has submitted its plant specific data to EPRI for inclusion in the test program, and B&W-supplied operational transient and postulated accident sequence data is being used in defining test parameters for the EPRI test matrix. One of the relief valve types chosen to be tested is the same model as the TMI-1 relief valve, and one of the safety valve types chosen to be tested is the same model as the TMI-1 safety valve. Therefore, the EPRI test results can be directly applied to TMI-1. Correa et al., ff. Tr. 8746, at 9-12. See also, Zudans, ff. Tr. 8824, at 5; Tr. 8922 (Zudans).

478. The NRC Staff has concluded that the PORV and safety valve test program is scheduled to be completed on the schedule required by NUREG-0737, and that the NUREG-0737 technical requirements for relief and safety valves, associated piping and supports, can be met. The Staff has found that Licensee has committed to the requirements of this item (NUREG-0578, 2.1.2) consistent with other operating reactors, noting that Licensee is participating in the EPRI program and is monitoring the program to assure that the test results apply to TMI-1 plant specific valves and associated piping and supports.[147] Staff Ex. 14 at 25, 26.

479. There are several reasons why Licensee believes that, in the absence of a completed EPRI valve test program, there is reasonable assurance that the valves will perform in an accident environment. _See_ Tr. 8789-91 (Urquhart). The only function required of the safety valves in order to provide overpressure protection or for feed and bleed operation is to open and discharge fluid. The disc lifts in response to the system pressure force on the disc face. The pressure at which the disc lifts -- _i.e._, at which the valve opens, or functions -- is dependent on the opposing force applied by the valve spring. Because of the construction of the valves there is no

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147 Block valve qualification is a new recommended requirement added by NUREG-0737, which was not in NUREG-0578. EPRI and the Staff are still discussing a formulation for such a test program. Staff Ex. 14 at 25, 26; Tr. 21,223-24 (Jacobs).

reason to expect that liquid or two-phase flow conditions would have a detrimental effect on the ability of the valves to perform their required function. Correa and Urquhart, ff. Tr. 8746, at 2.

480. This conclusion is specifically supported by the experience at Crystal River on February 26, 1980, and the examinations subsequent to that transient. See paragraphs 471 and 472, supra. The valve opened at 2400 psig; was open for approximately 20 minutes; experienced saturated steam, two-phase fluid and water at 2400 psig, 410°F with a maximum flow rate of 700 gpm; and reseated at 2300 psig (4% blowdown). These conditions are similar to those in one of the valve tests in the EPRI test program, in which the valve is set to open at 2500 psig, pass 450°F water at a maximum flow rate of 1000 gpm, and reseat at approximately 2375 psig (5% blowdown). Correa and Urquhart, ff. Tr. 8746, at 2.

481. Also, safety valves are used extensively in fossil power applications. Many of those valves are similar in basic design to the valves at TMI-1 and have experienced flow conditions other than steam. There is no known power industry incident of a properly set and maintained safety valve failing to open upon demand, even though liquid and two-phase flow through these valves has occurred. Correa and Urquhart, ff. Tr. 8746, at 2, 3; Tr. 8778-79 (Urquhart). See also, Zudans, ff. Tr. 8824, at 7.

482. The Staff is sufficiently confident in the outcome of the EPRI tests that it believes restart of TMI-1

should be permitted before the tests are completed. Tr. 8838 (Zudans). In addition to the fact that analysis of a stuck-open PORV shows that no fuel damage is predicted to occur, the Staff relies on the following: improved PORV position indication; TMI-1 procedures which require closure of the block valve early in a LOCA; the emergency power supplies for the PORV and block valve; and the generally upgraded TMI-1 emergency procedures for small-break LOCAs. Further, the setpoint changes and installation of anticipatory reactor trips will considerably lower the PORV challenge rate. This has been verified by operating experience. Zudans, ff. Tr. 8824, at 6, 7; Tr. 8838-39 (Zudans).

483. Neither the Commission nor the Staff has ordered any pressurized water reactor shut down pending completion of the EPRI valve test program. Tr. 8841 (Zudans). The Board finds that, contrary to former UCS Contention No. 6, the TMI-1 pressurizer relief and safety valves have been appropriately designed and tested. In addition, actions are being taken to provide further assurance that the valves will function properly and reliably. The Board also finds that recommendation 2.1.2 of NUREG-0578 is necessary and sufficient to provide reasonable assurance that TMI-1 can be operated in the long-term without endangering the health and safety of the public, and that pending the completion of the tests called for by that recommendation there is reasonable assurance that the valves will perform in an accident environment.

S. Accident Design Bases

Board Question/UCS Contention No. 13: The design of TMI does not provide protection against so-called "Class 9" accidents. There is no basis for concluding that such accidents are not credible. Indeed, the staff has conceded that the accident at Unit 2 falls within that classification. Of the realm of possible accidents, the Staff's method of determining which fall within the design basis accidents and those for which no protection is required is faulty in that the design basis accidents for TMI do not bound the credible accidents which can occur. Therefore, there is not reasonable assurance that TMI-1 can be operated without endangering the health and safety of the public and resumption of operation should not be permitted.[148]

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148 In addition to UCS Contention No. 13, the Board admitted "Class 9" accident contentions advanced by ECNP [ECNP-4(b) and -4(c)] and by Mr. Sholly [Sholly-17], which identified particular accident sequences with a nexus to the TMI-2 accident. For reasons discussed in paragraph 19 of our general Introductory Findings, supra, we rejected other "Class 9" contentions advanced by UCS, ECNP, ANGRY, and CEA. However, we permitted ECNP, ANGRY, and CEA each to adopt UCS Contention No. 13 in place of their rejected contentions. ECNP and CEA subsequently lost their rights to adopt UCS Contention No. 13, upon default on Board orders. Mr. Sholly withdrew his Contention No. 17 by memorandum dated December 23, 1980, and UCS, lead intervenor on its contention, withdrew its sponsorship of UCS Contention No. 13 by letter dated January 5, 1981. The Board did not adopt Sholly Contention No. 17, or ECNP Contentions 4(b) and 4(c) on which ECNP had defaulted, Tr. 11,025-26, but retained UCS Contention No. 13. ANGRY, the sole remaining intervenor with an interest in UCS Contention No. 13, conducted limited cross-examination of Licensee's witness and departed, and did not attend the evidentiary session at which the Staff presented its testimony on the issue. Compare Tr. 11,088 and Tr. 11,103.

484. UCS Contention No. 13 mounts a frontal attack on the Staff's methodology for determining which accidents, in the realm of possible accidents, fall within the design basis. Pointing to the Staff's determination that the TMI-2 accident was a "Class 9" accident, UCS asserts that there is no longer a basis for concluding that "Class 9" accidents are not credible. UCS further asserts that the design of TMI-1 does not provide protection against "Class 9" accidents, and that resumption of operation should not be permitted.[149]

485. The Board begins by reviewing briefly the foundation of UCS Contention No. 13 -- the Staff's determination that TMI-2 was a "Class 9" accident. We then discuss UCS's allegation that the design of TMI-1 does not provide protection against "Class 9" accidents. We next consider the specific measures employed at TMI-1 to provide protection against those events with a nexus to the TMI-2 accident. Finally, the Board examines the methodology used by the Staff to determine whether a particular accident is characterized as "credible" and included in the design basis envelope, or as "not credible" and excluded as "Class 9" accident.

486. The term "Class 9" event is derived from a proposed rule published by the Atomic Energy Commission in

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149 While UCS Contention No. 13 specifically refers to TMI-1, the Board has been presented with no evidence to suggest that TMI-1 is unique, among pressurized water reactors, either in the manner in which the Staff determined the design basis of the plant, or in the plant's capability to provide protection against "Class 9" accidents.

1971. The proposed rule, which has now been withdrawn by the NRC, set forth a spectrum of accidents divided into nine classes ranging from the most trivial to the most severe, for purposes of evaluating environmental risk pursuant to the National Environmental Policy Act. Class 8 events were characterized as:

> . . . those considered in safety analysis reports and AEC safety evaluations . . . used, together with highly conservative assumptions, as the design-basis events to establish the performance requirements of engineered safety features.

Since the highly conservative assumptions and calculations used in safety evaluations would, if used in environmental evaluations, result in a substantial overestimate of environmental risk, the proposed rule provided that Class 8 events were to be evaluated realistically. "Class 9" events were described as:

> . . . involv[ing] sequences of postulated successive failures more severe than those postulated for the design basis for protective systems and engineered safety features. Their consequences could be severe. However, the probability of their occurrence is so small that their environmental risk is extremely low.

The rule provided that it was not necessary to discuss such events in applicants' Environmental Reports or in Staff environmental impact statements. Rosenthal and Check, ff. Tr. 11,158, at 6, 7.

487. The Class 1 through 9 classification scheme was formally used only in the evaluation of environmental risk.

For purposes of safety evaluation, events are determined to be either "credible" or "not credible." Credible events are required to be considered in determining the adequacy of the design of a facility; incredible events are not. Tr. 11,196 (Rosenthal). See generally, Tr. 11,127-28 (Levy). "Incredible events" under the safety classification scheme (i.e., those beyond the design basis[150]) were thus, by definition, "Class 9" events in the terminology of the environmental classification system. The term "Class 9" has come to be used generally to describe events beyond the design basis. The Board so uses the term infra.

488. In responding to the inquiry of the licensing board in another proceeding, the Staff reviewed the sequence of events in the TMI-2 accident, and determined that the TMI-2 accident "involved a sequence of successive failures (i.e., small break loss-of-coolant accident and failure of emergency core cooling system) more severe than those postulated for the design basis of the plant." Considering that information in light of the definition of "Class 9" events in the proposed rule discussed supra, in paragraph 486, the Staff concluded that the TMI-2 accident was a "Class 9" accident. Rosenthal and Check, ff. Tr. 11,158, at 8. However, the off-site radiological consequences of the TMI-2 accident were inconsistent with the severe radiological consequences previously

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150 The term "design basis" is defined in paragraph 437, footnote 138, in section II.Q, supra, and is further discussed in paragraphs 490 and 491, infra.

generally assumed to be attendant to "Class 9" accidents. The radioactive material actually released to the environment during the TMI-2 accident represented a minimal risk to public health and safety, with consequences far less severe than 10 C.F.R. Part 100 guidelines. Tr. 11,195 (Rosenthal); Rosenthal and Check, ff. Tr. 11,158, at 8, 10. The Staff's classification of the TMI-2 accident as a "Class 9" accident was thus based solely on the number of equipment failures, and not on the severity of the consequences of the accident. Tr. 11,237 (Rosenthal); Tr. 11,128 (Levy). The classification was a difficult judgmental determination, and was the subject of dissent within the Staff. In fact, there are still a spectrum of opinions within the Staff. Tr. 11,238-39 (Check). While no party has asked the Board to reconsider the Staff's official position on the matter, and we proceed below to examine the merits of UCS's allegations, we remain conscious of the rather equivocal nature of the Staff's classification of the TMI-2 accident, the sole asserted basis for former UCS Contention No. 13.

489. The Board first assesses UCS's allegation that the design of TMI-1 does not provide protection against "Class 9" accidents.[151] We initially note that UCS's simplistic

151 UCS does not seem to claim that TMI-1 is <u>required</u> to be designed against "Class 9" events. Rather, the <u>apparent</u> thrust of the contention -- reading all the allegations together -- is that since the Staff's methodology for enveloping design basis events is "faulty," TMI-1 <u>should</u> be designed against "Class 9" events.

reliance on the classification of the TMI-2 accident as a "Class 9" accident as a basis for its contention does not support this particular allegation. Since the off-site radiological consequences of the TMI-2 accident were well below established 10 C.F.R. Part 100 guidelines (see, paragraph 488, supra), one can no more argue that the accident proved that the design of nuclear plants provides no protection against "Class 9" accidents than one can argue that the accident proved that the design of nuclear plants provides protection against all "Class 9" accidents. As the evidence on the merits of UCS's allegation indicates, the truth lies somewhere between the two extremes. See generally, Tr. 11,272-73 (Check).

490. In the licensing process, a nuclear plant and its various safety systems are performance-tested to meet the criteria for design basis events, the set of prescribed anticipated operational occurrences and accidents used to assess the responses of specific systems to upset conditions.[152] Specific event sequences have been developed over the years to determine conservative requirements for various safety systems.[153] Plant

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152 Anticipated operational occurrences are events or conditions expected to occur one or more times during the life of a nuclear plant; accidents are events expected to occur less frequently, if at all. Rosenthal and Check, ff. Tr. 11,158, at 4.

153 For example, uncontrolled rod withdrawal, moderator dilution, and the ejected rod events represent a spectrum of events which challenge the reactivity control system. Similarly, a spectrum of reactor coolant pipe breaks are postulated to specify the design requirements for the emergency core cooling system (ECCS). We briefly discuss the development of the en-
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response to the design basis events is assessed against the requirements of 10 C.F.R. Part 50 (e.g., Appendix A, General Design Criteria) and other regulatory standards (such as Regulatory Guides). The potential radiological consequences of the design basis events are calculated to ensure conformance with the guidelines of 10 C.F.R. Part 100. Rosenthal and Check, ff. Tr. 11,158, at 4, 5; Tr. 11,056-57 (Levy).

491. Each of the design basis events, and particularly the design basis accidents, imposes severe performance demands (or loading conditions) on the various safety systems which must function in response to such events if the plant design is to meet regulatory requirements. Rosenthal and Check, ff. Tr. 11,158, at 4-5, 8-9; Levy, ff. Tr. 11,049, at 2-4. Moreover, each of the events is analyzed using conservative assumptions regarding equipment availability and performance capability, as well as conservative values of process variables. The plant is thus tested not only against a set of challenges to its safe operation, but also under additional conservative assumptions regarding plant conditions before and during those challenges.[154] This results in a design

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velope of design basis events in paragraph 506, infra. The design basis events for TMI-1 are delineated in Chapter 14 of the FSAR, and include loss-of-coolant flow, steam-line break, ejected rod, and loss-of-coolant accidents. Rosenthal and Check, ff. Tr. 11,158, at 4, 5; Levy, ff. Tr. 11,049, at 2-4.

154 A few of the many conservative assumptions employed are: (1) the assumption of the worst or most limiting single failure in any safety-related system or function utilized in the (continued next page)

capability with multiple and redundant systems for coping with very severe performance demands (or loading conditions), and provides protection against unforeseen events, including multiple equipment failures and operator error. Rosenthal and Check, ff. Tr. 11,158, at 9; Levy, ff. Tr. 11,049 at 4, 5.

492. Further, design basis events do not establish the <u>limits</u> of a plant's <u>performance</u>. For example, the postulated steam-line break and the postulated loss-of-coolant accident are used to establish <u>minimum</u> requirements for the containment with respect to differential pressure. The actual design pressure of the containment always exceeds the pressure required by the design basis analyses, to provide design margin. Due to the conservative requirements of the ASME code, the failure pressure of the containment is well beyond the design pressure. As a result, the Staff is finding that containments subjected to uniform static pressure loadings probably have ultimate capabilities of <u>at least</u> two-and-one-half times design pressure. Rosenthal and Check, ff. Tr. 11,158, at 9; Levy, ff. Tr. 11,049, at 10.

493. The inherent flexibility incorporated into many plant systems and the multiplicity of installed systems afford

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required analyses; Tr. 11,070 (Levy); (2) the assumption that whenever Departure from Nucleate Boiling occurs, the fuel cladding fails; Tr. 11,067-69 (Levy); and (3) licensing calculation methods which overpredict by approximately 500° to 1000° F the temperatures measured at the Loss-of-Fluid Test (LOFT) facility during simulated loss-of-coolant accidents; Tr. 11,069 (Levy). <u>See generally</u>, Levy, ff. Tr. 11,049, at 4, 5.

additional margin for overall safe response to unforeseen events. Thus, the plant design can tolerate unforeseen event sequences through appropriate use of installed emergency safety features, as well as through other equipment not credited in the design basis analyses. (For example, alternative systems configurations -- i.e., valve line-ups, electrical interconnections, etc. -- may be used, or equipment may be manually initiated if automatic logic circuits do not initiate action). Rosenthal and Check, ff. Tr. 11,158, at 10.

494. Finally, the source terms used in dose calculations for TMI-1 are based on the assumptions that 100% of the core noble gases and 25% of the core iodines are released to the containment atmosphere. However, analyses of containment air samples indicate that, during the TMI-2 accident, 60 to 70% of the core noble gases but only 0.6% of the core iodines were present in the containment atmosphere (due to a number of chemical and physical attenuation phenomena which are not considered in current dose calculation assumptions. See generally, Levenson, ff. Tr. 19,525. We discuss these phenomena further in the portion of this Initial Decision on emergency planning). There is thus a spectrum of severe core damage scenarios for which adequate radiological protection has been provided, as long as containment integrity is maintained. Rosenthal and Check, ff. Tr. 11,158, at 10; Levy, ff. Tr. 11,049, at 7, 9-10; Tr. 11,098-100 (Levy).

495. In addition, Licensee's expert witness testified extensively on the general theory of "defense-in-depth"

in the design and licensing of nuclear plants, which relies upon multiple physical barriers designed to prevent the release of radioactive fission products to the environment. The first barrier of defense-in-depth is the ceramic form of fuel and the fuel cladding employed, which limit the release of radioactivity from the fuel, even in extreme cases of fuel melt. Any radioactivity which is released from the fuel is confined by the reactor coolant system piping and vessel, the second barrier, as long as they remain intact. Finally, even if the reactor coolant system boundary is breached, the containment building is designed to confine any radioactivity escaping from the reactor coolant system, and so serves as a third barrier.[155] Thus, defense-in-depth ensures further inherent

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155 As Licensee's witness noted, the factors of defense-in-depth are sometimes listed as (1) the multiple physical barriers to the release of radiation, (2) siting, and (3) emergency planning. These last two can be particularly important in mitigating the consequences of beyond design basis events. Levy, ff. Tr. 11,049, n. at 9; Tr. 11,052 (Levy). Licensee's witness did not attempt to quantify the protection afforded by either siting or emergency planning at TMI-1, but observed that the regulatory policy of siting nuclear plants in areas of relatively low population density _per se_ reduces in some measure the consequences that might otherwise attend a release of radioactivity (if, for example, plants were sited in the middle of large metropolitan areas). This is true for all licensed plants, including TMI-1, even though -- solely considering population density -- the TMI site does not provide as great an advantage as that of the average operating reactor. Tr. 11,052-53, 11,056-58, 11,064 (Levy). Though Licensee's witness was not conversant on the specifics of TMI-1 emergency planning, he was sufficiently familiar with emergency planning regulations to know that Licensee's compliance with those requirements would _per se_ provide some measure of additional protection to the public in the event of an accidental release of radioactivity from TMI-1. Tr. 11,051, 11-057-58, 11,060-61 (Levy). We discuss emergency planning in great detail in a later portion of the Initial Decision.

reserve design capability, since the integrity of at least one physical barrier is preserved and is available for protection following a design basis accident. Levy, ff. Tr. 11,049, at 8-10, 12-13.

496. The Board therefore rejects UCS's broad assertion that "[t]he design of TMI does not provide protection against . . . 'Class 9' accidents." We find, to the contrary, that, while no nuclear plant can provide protection against <u>all</u> possible "Class 9" accidents, the design of TMI-1 -- particularly the combination of the use of conservative assumptions in design basis analyses; the difference with respect to equipment between design basis analysis requirements and actual design specifications, and the differences between design specifications and ultimate capability; the inherent flexibility and multiplicity of plant systems; the conservatisms in dose calculations; and the underlying philosophy of defense-in-depth -- provides protection for a wide range of "Class 9" events. Indeed, as we have discussed <u>supra</u>, at paragraph 488, the TMI-2 accident, a "Class 9" event, was mitigated with radiological consequences far less severe than 10 C.F.R. Part 100 guidelines.

497. Moreover, since the TMI-2 accident, many actions have been taken to reduce the likelihood of "Class 9" events. Immediately following the TMI-2 accident, the Staff issued a number of bulletins and orders, which were followed by the formulation of a TMI Action Plan which included extensive recommendations related to operator training and procedures,

instrumentation, equipment reliability and hardware modifications -- generally "lessons learned" from the TMI-2 accident. Rosenthal and Check, ff. Tr. 11,158, at 11. We generally review the sufficiency of these actions in our findings on Staff Review and Recommendations, in section II.T, infra.

498. Further, the Staff constructed event trees for the analysis of event sequences with a nexus to the TMI-2 accident, and correlated those events with the measures taken in response to various requirements related to the restart of TMI-1 which will reduce the probability of the occurrence of the identified sequences. The Staff also identified the actions taken to reduce the probability of occurrence of each functional failure, relating each functional requirement to systems performance and, in turn, to equipment performance. In addition, the Staff prepared a list of actions taken which would mitigate the failures of functions shown on the event trees. The Staff's analyses are documented in Staff Exhibit 3, TMI-1 Potential Core Damage Accident Sequences and Preventive and Mitigative Measures.[156] Rosenthal and Check, ff. Tr. 11,158, at 11.

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156 In the Board's First Special Prehearing Conference Order, LBP-79-34, 10 N.R.C. 828, 835 (1979), we directed the Staff:

> to inform this Board and the Commission whether or not (and the reasons therefor) any specific accident sequence, which has a reasonable nexus to the TMI-2 accident and which heretofore may have been regarded as a Class 9 accident, should be considered in the analyses of the acceptability of returning TMI Unit 1 to operation.

(footnote continued next page)

499. Two event trees were considered. Event Tree T has as its initiating event a loss of main feedwater (LMFW) transient; Event Tree S is assumed to be initiated by a loss-of-coolant accident (LOCA). Given the two initiating events, the method presented in Appendix I to WASH-1400 was followed. Thus, the Staff identified the functions which must be performed by systems and equipment following the initiating event in order to preclude core damage. Each tree proceeds from left to right by the addition of branches corresponding to two alternatives: successful performance of function (upper branch) and failure (lower branch).[157] After Event Trees T and S were drawn, the Staff traced a path across each tree by choosing successive branches. Each path corresponds to an accident sequence. As indicated on Event Tree T, some sequences initiated by LMFW transients result in LOCAs. In those

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(continued)
Our March 31, 1980 Memorandum and Order to NRC Staff Regarding Class 9 Accidents further directed the Staff to specify and describe each of the critical accident sequences that it would analyze in order to assure that the proposed short- and long-term actions necessary and sufficient to provide adequate protection to public health and safety have been taken. Staff Exhibit 3 was compiled in response to the Board's orders.

157 The Staff identified the potential functional failures related to Event Tree T as: loss of main feedwater, failure of emergency secondary heat removal, requirement for primary system pressure relief, failure of primary system pressure relief, and failure of primary system integrity. The potential functional failures related to Event Tree S were identified as: failure of emergency coolant injection, failure of post-accident radioactivity removal, failure of post-accident heat removal from containment, and failure of emergency coolant recirculation.

cases it is necessary to move from Event Tree T to Event Tree S to complete the accident sequence. Staff Ex. 3 at 2-3, 6, 11, 13-14.

500. The Staff addressed each node in the event trees, to ensure that action has been taken at each step to reduce the likelihood of progressing beyond that node. Tr. 11,250-51 (Check, Rosenthal); Tr. 11,252-53 (Rosenthal). The specific hardware modifications which the Staff identified which will improve the reliability of the emergency feedwater (EFW) system for Event Tree T include: automatic initiation of EFW, modification of EFW control valves, automatic block loading of motor driven EFW pumps on diesels, indication of EFW to each steam generator, indication of EFW supply water, automatic termination of EFW flow to a depressurized steam generator with automatic supply to the intact steam generator, and planned separation of EFW from ICS.[158] The modifications which will reduce challenges to the pressure relief valve for Event Tree T include raising the relief valve setpoint and reducing the high-pressure reactor trip setpoint, and the installation of anticipatory trips for loss-of-feedwater and turbine trip. The modifications which will improve the reliability of the pressure relief valves and increase the probability of maintaining primary system integrity for Event

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[158] We have examined the modifications to the TMI-1 EFW system in detail in section II.Q, supra.

Tree T include testing of relief and safety valves, the installation of direct indication of valve position, and the provision of emergency power to relief and block valves and associated instrumentation control. Rosenthal and Check, ff. Tr. 11,158, at 13-15; Staff Ex. 3 at 42, 51-52, 54, 56.

501. The specific hardware modifications related to Event Tree S which the Staff identified which will provide the operator with enhanced information for proper initiation and termination of emergency core coolant injection systems, and which will also enhance availability of emergency coolant recirculation, include the installation of a subcooling meter and instrumentation to detect inadequate core cooling (to be implemented on a schedule consistent with requirements of other operating reactors). Rosenthal and Check, ff. Tr. 11,158, at 15; Staff Ex. 3 at 61-62, 75-76.

502. In addition to these specific hardware modifications, the Staff identified many other modifications and improvements (e.g., procedural changes) which it related to specific nodes of the event trees. Rosenthal and Check, ff. Tr. 11,158, at 12; Staff Ex. 3, Tables 7-15. The Staff further identified a number of measures taken which are generally applicable to all nodes on Event Trees T and S, including requirements for licensee review of operating experience, operational quality assurance, verification of management and technical capability, verification of capability for safety review and operational guidance, training of operators, review

of facility procedures, review of plant maintenance capability, requirements for shift turnover procedures, requirements related to shift manning, requirements for an onsite safety engineering group, systematic assessment of licensee safety, and requirements for a shift technical advisor. Rosenthal and Check, ff. Tr. 11,158, at 13; Staff Ex. 3 at 77, Table 16. The effect of these changes is, first, to enhance the maintenance and operation of the systems involved in each step of the identified event sequences, thus diminishing the probability of malfunction of the various components of these systems; and second, to upgrade significantly the ability of the operators and the operating organization to recognize and to take the proper remedial action to cope with malfunction should it occur. There is a combined effect from improvement in both these aspects on each and every step in the event sequences. Thus, the cumulative effect of both of these aspects is a substantial increase in the overall likelihood of successful safe termination of the initiating events. Rosenthal and Check, ff. Tr. 11,158, at 13.

503. As discussed in paragraph 509, <u>infra</u>, the Staff often uses quantitative probability assessments to focus its attention on weak systems (<u>i.e.</u>, those with dominant contributions to total risk). However, in analyzing events with a nexus to the TMI-2 accident, as presented in Event Trees T and S, the Staff did not limit its attention to a perceived dominant sequence but rather took a "broad brush" approach and

proposed remedial actions which affect many systems involved in the entire chain, from initiating event to potential core melt. The Staff's comprehensive approach to these nexus events obviated the need for a probabilistic assessment screening to focus attention on particular sequences. Rosenthal and Check, ff. Tr. 11,158, at 16; Tr. 11,263-65 (Rosenthal). (While the Staff did not quantitatively evaluate the nexus events and corresponding remedial actions, Licensee's witness generated a quantitative estimate -- based on his professional engineering judgment and experience -- of the increase in the margin of safety, achieved by specific modifications, in typical accident sequences with a nexus to the TMI-2 accident. Though the Board believes the Staff's comprehensive approach to be a sufficient basis for the findings here, we note that Licensee's witness concluded, for example, (1) that certain identified actions provide for an overall improvement factor of approximately two to three orders of magnitude for a TMI-2 type accident initiated by a LMFW; (2) that similar substantial improvements have been made for other overpressure transients involving relief valves; and (3) overall, substantial action has been taken to ensure that event sequences with a nexus to the TMI-2 accident will be terminated long before the core reaches a degraded condition. Levy, ff. Tr. 11,049, at 14-18.)

504. The Staff believes that Licensee's implementation of the identified measures reduces the analyzed event sequences from "dominant contributors to total risk" to a level

of risk "consistent with other contributory risks of the facility as a whole." Thus, the Staff concludes, the probability of the event sequences with a nexus to the TMI-2 accident occurring and leading to a core melt, with concurrent or consequential containment failure such that 10 C.F.R. Part 100 guidelines are exceeded, is sufficiently low that the event sequences may be considered not "credible," and there exists reasonable assurance that TMI-1 can be operated without endangering the health and safety of the public. Rosenthal and Check, ff. Tr. 11,158, at 16. Moreover, the Staff's qualitative evaluation of the post-TMI-2 "fixes" indicates that they result in an overall improvement in plant safety, beyond events with a nexus to the TMI-2 accident. Tr. 11,218 (Rosenthal). Cf., Tr. 11,094 (Levy). Finally, although the Staff's Event Trees T and S only follow sequences to the stage where either severe core damage or core melt is indicated, the Staff has also identified extensive measures, both short-term and long-term, which are being implemented by the Staff and by Licensee and which will result in an enhanced ability to deal with and mitigate the consequences of degraded core conditions, as well as situations involving inadequate core cooling. Staff Ex. 3 at 83, Table 17; Tr. 11,270-74 (Rosenthal).

505. The Board therefore finds that the Staff has identified and analyzed -- through detailed event tree procedures -- those critical event sequences with a reasonable nexus to the TMI-2 accident. Based on the Staff's analyses of these

event trees, and the corresponding preventive and mitigative measures prescribed by the Staff and implemented by Licensee to address potential failures at each node of the event trees, the Board concludes that the event sequences with a nexus to the TMI-2 accident represent risks consistent with other contributory risks of the facility as a whole.

506. Finally, the Board examines UCS's allegation that the Staff's methodology for determining which events fall within the design envelope and which do not is "faulty in that the design basis accidents for TMI do not bound the credible accidents which can occur." Reactor regulation is necessarily an evolving process, with new information and techniques incorporated into the process as they are developed and verified. Historically, in the early days of reactor safety assessments, the accidents against which plants were designed were determined through group assessment of potential failures. Efforts focused on bounding those events that might reasonably be expected to occur -- the "credible" events -- by postulating the failure of each major system in turn, and requiring that the plant mitigate the consequences, to ensure that predicted off-site doses were within 10 C.F.R. Part 100 guidelines. The process of identifying the bounding conditions frequently took the form of extended discussion, and debate -- both oral and written -- among the Staff, the Advisory Committee on Reactor Safeguards (ACRS), and nuclear industry experts. Increasingly sophisticated safety assessments, refined by continuing

analyses and experience, were performed through the 1960s and early 1970s. By the mid-1970s, the Staff had developed the basic set of design basis events (anticipated operational occurrences and accidents) now used by the Staff in all case reviews to assess the overall adequacy of the design of a plant. Rosenthal and Check, ff. Tr. 11,158, at 17, 18.

507. The Staff's fundamental methodology for determining whether a particular sequence is "credible" or "incredible" for design review purposes is thus based primarily on engineering judgment, informed by engineering assessment of the performance characteristics of the various reactor systems and components, and by engineering evaluation of the system and component failures that may occur. The judgment as to whether a given sequence is "credible" or "incredible" does not rest upon the application of a specified numerical criterion or a particular definition. Rather, the Staff's approach, generally characterized as a "mechanistic" or "deterministic" approach, relies upon the composite of the engineering experience and technical expertise of the Staff, supplemented by that of the ACRS, with substantial contribution from the experience and expertise of the designers, builders and operators of nuclear power reactors. Rosenthal and Check, ff. Tr. 11,158, at 17; Tr. 11,180-81 (Check); Tr. 11,203 (Rosenthal, Check); Tr. 11,247-48 (Check); Levy, ff. Tr. 11,049, at 2.

508. However, though the Staff's accident classification methodology can be generally characterized as a

"mechanistic" or "deterministic" approach, elements of "probabilistic" analysis have historically been included. Tr. 11,201, 11,253 (Check). The most comprehensive use of risk assessment to date appears in the Reactor Safety Study, WASH-1400.[159] Rosenthal and Check, ff. Tr. 11,158, at 25. In the development of the envelope of design basis events, the probability associated with calculated sequences was given limited (generally quantitative) consideration, and some elements of probability were reflected, primarily in the selection of event sequences to be analyzed. Rosenthal and Check, ff. Tr. 11,1-8, at 20. Other examples of Staff use of quantitative probabilistic technique include: (a) the development of a numerical probability criterion (described in Section 2.2.3 of the Standard Review Plan) for the assessment of consequences of accidents arising from external hazards to a plant; (b) the preparation of an analysis of pressure vessel reliability; (c) the assessment of probabilities associated

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159 After the issuance of the "Lewis Report," the Commission issued a policy statement accepting the Lewis Committee's major findings and disclaiming endorsement of the Executive Summary of WASH-1400. The policy statement provided, inter alia, that the WASH-1400 absolute values of risk should not be used uncritically in the regulatory process, but that the Staff should use components of WASH-1400 where the data base is adequate and analytical techniques permit. See, Florida Power & Light Co. (St. Lucie Nuclear Power Plant, Unit No. 2), ALAB-603, 12 N.R.C. 30, 47 n.60 (1980). Cf., Tr. 11,124-25, 11,173 (Smith); Tr. 11,153 (Smith, Baxter); Rosenthal and Check, ff. Tr. 11,158, at 24, 25. The use of WASH-1400 by the witnesses of Staff and Licensee was consistent with this guidance. Tr. 11,146, 11,172-74 (Levy); Tr. 11,160-61 (Check, Rosenthal).

with BWR rod drop events; and (d) the consideration of the Anticipated Transients Without Scram (ATWS) issue. Rosenthal and Check, ff. Tr. 11,158, at 21-25; Tr. 11,249 (Check). See generally, Tr. 11,205-12 (Rosenthal, Check).

509. Moreover, there has been a gradual, continuing increase in the employment of probabilistic risk analysis techniques to augment the Staff's traditional deterministic methodology. Tr. 11,175 (Rosenthal); Tr. 11,177 (Check). For example, the Staff now often uses quantitative probabilistic assessment to identify areas of relative strength and weakness (i.e., relative contributions to total risk), and to focus resources and attention on areas which should be subject to detailed engineering review. Rosenthal and Check, ff. Tr. 11,158, at 17, 20; Tr. 11,182 (Rosenthal). In addition, Staff engineers are making increasing use of event and fault tree methodology in specific analyses, particularly in examining isolated elements of larger problems. In this context, probabilistic risk assessment serves as a systematic construct for studying plants and understanding plant behavior. Tr. 11,175, 11,183 (Rosenthal). Quantitative probabilistic assessment techniques were also used in the recent Staff evaluations of certain alleged "high risk" sites -- Indian Point, Dresden and Limerick. Tr. 11,183 (Check).

510. Further, the Staff plans to broaden its use of quantitative risk assessment techniques in the future. As indicated in the TMI Action Plan (NUREG-0660, Section II.C.2),

a wide-ranging National Reliability Evaluation Program (NREP) is planned. The initial stage (the first four plant evaluations), known as the Integrated Reliability Evaluation Program (IREP), is underway, and steps are being taken to broaden the study.[160] NREP is designed to include all plants (including TMI-1), using common data bases, recognized techniques, peer review process, and comparable methodology (i.e., event tree and fault tree methodology). Rosenthal and Check, ff. Tr. 11,158, at 27; Tr. 11,259 (Rosenthal). Finally, although the Commission does not presently have a numerical safety goal, it recently approved a document entitled Plan For Developing Safety Goal (NUREG-0735). The safety goal itself is a socio-political, economic decision which will not involve employment of quantitative probabilistic assessment techniques. However, the development of such a goal may have a great effect on the regulatory process as a whole, including the use of probabilistic techniques. Rosenthal and Check, ff. Tr. 11,158, at 27; Tr. 11,178 (Check); Tr. 11,193 (Rosenthal).

511. With the increasing use of risk assessment, and with a perception of event sequences as a continuum of probabilities with an associated continuum of consequences, the

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160 IREP is designed to provide information to allow the Commission to make sound judgments on the value and scope of planned subsequent phases of comprehensive reliability assessment programs. Tr. 11,190, 11,259 (Rosenthal); Tr. 11,261 (Check). We discuss IREP further in section II.T, infra.

Staff has recently begun to extend its consideration of failure sequences, to include events which have not previously been considered to be design basis events. Thus, the Staff now explicitly considers a much wider range of event sequences than were considered prior to the TMI-2 accident, some involving multiple failures and some involving systems not traditionally considered safety systems. Rosenthal and Check, ff. Tr. 11,158, at 18, 19; Tr. 11,196-98, 11,246-47 (Rosenthal); Tr. 11,248 (Check).

512. The Staff addresses those event sequences designated as design basis events primarily by requiring installation of engineered safety features, though administrative controls have also been employed in some cases. Event sequences not designated as design basis events are "fixed" through a variety of requirements, including increased surveillance and testing of existing equipment, procedural modifications, and improved operator training, as well as some equipment modifications. In addressing sequences not designated as design basis events, the Staff focuses on modifications which eliminate the initiating event or improve the capability of some other systems to compensate for or cope with the initial malfunction, as well as on improvements in mitigating system characteristics. These "fixes" are designed to diminish the likelihood of the particular sequence to a low level relative to other potential reactor safety system malfunctions, or to reduce the potential consequences of such an event to a level

less severe than those associated with analyzed design basis events. Rosenthal and Check, ff. Tr. 11,158, at 18, 19.

513. Though the Staff believes that probabilistic risk assessment techniques can be employed profitably to augment the more traditional deterministic techniques used by the Staff, the Staff believes it would be ill-advised at this time to rely solely upon probabilistic techniques as a basis for regulatory decisions. Tr. 11,177, 11,181-82, 11,201-02, 11,253 (Check). Licensee's expert witness agrees. Tr. 11,090-91 (Levy). The Staff's reservations about the exclusive use of probabilistic techniques reflect three basis concerns. First, due to a lack of sufficient failure-rate data and difficulties in developing complete system models, comprehensive assessments which have been adequately tested are rare (though the available data base is improving). Cf., Tr. 11,164 (Check); Tr. 11,168, 11,176-77, 11,182-83, 11,190, 11,235 (Rosenthal); Tr. 11,240-43 (Check). Second, the Staff presently has no numerical probability goal against which to assess compliance; moreover, any such numerical goal would be difficult to apply given the range of uncertainty in calculating probabilities. Cf., Rosenthal and Check, ff. Tr. 11,158, at 26, 27; Tr. 11,178 (Check); Tr. 11,193-95, 11,198-99 (Rosenthal). And, finally, the Staff believes its current approach -- which utilizes composite engineering experience and judgment, as well as probabilistic techniques -- provides a sound, comprehensive basis for its decisions. Cf., Tr.

11,180-81, 11,247-49 (Check). See generally, Rosenthal and Check, ff. Tr. 11,158, at 20.

514. The Board heard no evidence whatsoever -- other than a discussion of the Staff's equivocal classification of the TMI-2 accident as a "Class 9" accident, which we reviewed supra, at paragraph 488 -- which would suggest that the methodology used by the Staff to determine the envelope of design basis accidents is "faulty" as alleged by UCS. Rather, the totality of the record on the issue leads us to conclude that the Staff has historically engaged the nuclear community -- including industry experts and the ACRS -- in a continuing dialogue designed to systematically identify, through methodical analysis and the evaluation of operating experience, those accident sequences that might reasonably be expected to occur. Moreover, since the TMI-2 accident, the Staff has expanded its reviews to include consideration of certain sequences traditionally considered to be beyond design basis events. Further, while the Staff's methodology is primarily based on composite engineering experience and technical expertise, the Staff has in the past incorporated probabilistic risk assessment techniques into its methodology, and plans to make increasing use of probabilistic assessment in the future. The Board believes that there is great utility in probabilistic risk assessment, and encourages the Staff to continue to incorporate such techniques into the regulatory process wherever appropriate. However, in light of the present lack of

an adequate data base, appropriate models, and a numerical safety goal, the wholesale substitution of probabilistic risk assessment for professional engineering judgment (including use of probabilistic techniques) would be at best premature at this time. We nevertheless note that the Commission's overall safety goal, now being developed, may have far-reaching implications on the use of probabilistic techniques in the regulatory process.

515. In summary, then, the Board rejects the broad implication of UCS Contention No. 13 -- that the Staff's equivocal classification of the TMI-2 accident as a "Class 9" accident in and of itself undermines the Staff's methodology for enveloping design basis accidents. We similarly reject UCS's allegation that the design of TMI-1 does not provide protection against "Class 9" accidents and find, to the contrary, that the TMI-1 design provides protection for a wide range of "Class 9" events. We further find that, based on the Staff's analyses of the event sequences with a nexus to the TMI-2 accident, as well as the corresponding preventive and mitigative measures being implemented to address each potential failure in each sequence, the risks represented by the nexus sequences are not dominant contributors to the total risk but, rather, represent risks consistent with other contributory risks of the facility as a whole. Finally, we reject as without basis UCS's allegation that the Staff's methodology for enveloping design basis accidents is "faulty." We believe

there is great value in probabilistic risk assessment techniques, however, and encourage the Staff to continue to incorporate them into the regulatory process as appropriate.

T. Staff Review and Recommendations

Board Question No. 1:

Prior to the opening of the evidentiary hearing, the staff should inform the board as to when the staff will take a position on the applicability to this proceeding of NUREG-0694, "TMI-Related Requirements for New Operating Licenses". The following items in NUREG-0694 and/or NUREG-0660, "Action Plans for Implementing Recommendations of the President's Commission and Other Studies of TMI-2 Accident", are of particular interest to the board:

a. I.D.1 -- Control Room Design (following a human factors analysis).

b. II.E.1.1 -- Auxiliary Feedwater System (reliability evaluation using event-tree logic).

c. II.B.8 -- Rulemaking proceeding on degraded core accidents.

d. II.B.7 -- Analysis of response of containment structures to hydrogen explosions. Do the proposals for hydrogen control for Sequoyah, which include "spark plugs", have any applicability to TMI-1?

Board Question No. 2:

The board stated its concern with having an adequate record on the sufficiency of the proposed short-term and long-term actions to protect the health and safety of the public.

Without further explanation the question may appear to invite conclusionary testimony on the ultimate factual issues to be decided by the board. (Commission's August 9, 1979 order, 10 NRC 141, 148.) This is not what the board has in mind as a response to the question. Our concerns were expressed in part in the June 23, 1980 memorandum on the staff's report on TMI-1 accident sequences. To explain further: We assume that the staff and licensee may present evidence that each Category A and each Category B recommendation in Table B-1 of NUREG-0578 (Order items ST 8 and LT 3), and that each preventative and mitigative measure identified with respect to a given accident sequence in the staff's TMI-1 Core Damage Accident Sequence Report will be, at least, sufficient to resolve the related safety problem or accident sequence. However, nowhere have we seen in the Restart Report, SER, the accident sequence report, or elsewhere, an explanation as to how the staff or licensee has determined that all of the necessary TMI-2 related recommendations have been identified and that all the appropriate accident sequences have been addressed. The board wants testimony or other evidence which explains, if such be the case, how the licensee and the staff have concluded that the NUREG-0578 short and long-term recommendations, other subsequent safety recommendations, and the identified accident sequences (with their respective preventative or mitigative measures) are <u>in their totality</u> sufficient to provide reasonable assurance that TMI-1 can be operated without endangering the health and safety of the public. The question is not intended to enlarge the scope of the hearing. The response may be limited to consideration of accidents following a loss-of-feed-water transient.

Board Question No. 3: The results of the Interim Reliability Evaluation Plan (IREP), as applied to Crystal River, was scheduled for completion in July 1980. (The board wants to receive a copy of this report.)

a. When will the IREP be applied to TMI-1?

b. Does the IREP address the adequacy of the proposed actions for B&W plants?

Board Question No. 5: When does the staff plan to report on its review of NUREG-0660 as applied to TMI-1? (The board and the parties should be kept informed as quickly as the staff has identified any additional action plans that should be required for implementation, either before any proposed restart or for the long-term.)

Board Question No. 7: Following the investigation of the Crystal River incident, the staff issued NUREG-0667, "Transient Response of Babcock & Wilcox-Designed Reactors". Which of the recommendations in Table 2.1 of that report does the staff believe should be implemented for TMI-1 prior to start-up, which should be included in the long-term actions, and which, if any, are not needed for TMI-1 and why not?

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516. The Board views these questions, which we propounded during the prehearing conference of August 12-13, 1980, as inquiring generally into the sufficiency of the actions recommended by the Staff in the wake of the TMI-2 accident and, specifically, as to the sufficiency of the

actions recommended (both short and long-term) for TMI-1. In this portion of the Initial Decision, the Board will first address the mechanisms employed by the Staff to determine what actions are necessary in order to adequately protect the public health and safety, the criteria used to determine whether these actions are required prior to the restart of TMI-1 and, for those actions which the Staff concluded are not required prior to restart, the criteria by which the Staff found that the actions taken by Licensee constituted "reasonable progress". Finally, the Board will examine the necessity and sufficiency of the individual actions referenced in our questions.[161]

517. In the aftermath of the TMI-2 accident, a number of independent groups, as well as separate groups within the NRC Staff, conducted investigations into the causes of the accident and recommended actions to be taken by all licensees in order to prevent or mitigate the consequences of a similar event. The recommendations of each of these groups were collected, assessed and consolidated into one discrete set of

161 The Board has previously made findings of necessity, sufficiency and reasonable progress on a number of items which were litigated by the parties. Board questions as to certain specific recommended actions are addressed elsewhere in this decision, along with closely-related issues raised (at least once) by the intervenors, or in the case of Board Question 6 (Emergency Feedwater Reliability), where no party has challenged a system of interest to the Board. We do not repeat these individual findings here. Additionally, the Board has addressed the matter of the sufficiency of the accident sequences examined (as raised in Board Question No. 2) in Section II.S, supra.

recommendations and published by the Staff as NUREG-0660, "NRC Action Plan Developed as a Result of the TMI-2 Accident" (the "Action Plan") in May, 1980.[162] Prior to its publication, drafts of the Action Plan were reviewed by the Commissioners, the ACRS, the NRC Executive Director for Operations and the directors of NRC's program offices. The Staff believes that the recommendations contained in the Action Plan,[163] in view of their promulgation and assessment by persons having expert knowledge over a broad range of technical disciplines, provide reasonable assurance that the causes of the TMI-2 accident and their associated corrective measures have been completely and adequately identified. Ross, ff. Tr. 15,555, at 3-5; Tr. 15,622-23 (D. Ross). The Board agrees that the inclusion in the Action Plan of the recommendations of the various investigatory groups, in combination with the ongoing Staff review efforts (<u>see</u> paragraph 522, <u>infra</u>), provides adequate assurance that all important TMI-related recommendations have been identified.

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162 Investigations of the TMI-2 accident which the Staff considered in formulating the Action Plan include those performed by: the President's Commission on the Accident at Three Mile Island; Congress; the General Accounting Office; the NRC Special Inquiry Group; the Advisory Committee on Reactor Safeguards (ACRS); the Staff's Lessons Learned Task Force; the Bulletin and Orders Task Force; and, the Office of Inspection and Enforcement's Special Review Group. Ross, ff. Tr. 15,555, at 3, 4.

163 Fifty-four of the Action Plan recommendations are encompassed within the scope of the long- and short-term actions contained in the Commission's Order and Notice of Hearing of August 9, 1979. Ross, ff. Tr. 15,555, at 8 and Table 2.

518. In concert with the publication of the Action Plan, the Staff undertook an assessment of these recommendations to identify those that were known to have significant safety improvement potential and required that these items be implemented by applicants prior to the issuance of a new operating license.[164] Those Action Plan items which the Staff identified as required for NTOLs were subsequently officially published as NUREG-0694, "TMI-Related Requirements for New Operating Licenses." Earlier in this proceeding, the Staff had taken the position that TMI-1 was to be treated as an NTOL and would be required to comply with the requirements of NUREG-0694 on a schedule consistent with that of other NTOLs (i.e., the "fuel-load" or "full-power" requirements of NUREG-0694 were to be met prior to restart or prior to exceeding 5% power). Ross, ff. Tr. 15,555, at 8, 9; Tr. 10,525-26 (Ramirez); Tr. 15,647, 15,656-57 (D. Ross).

519. Of the forty-eight requirements contained in NUREG-0694 which the Staff believed were applicable to TMI-1, thirty were encompassed by the requirements contained in the Commission's Order and Notice of Hearing of August 9, 1979. It was the Staff's position that the remaining eighteen items should be completed prior to the restart of TMI-1.[165] Ross, ff. Tr. 15,555, at 9, 10.

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164 These items have commonly been referred to throughout the course of this hearing as the "NTOL (near-term operating license) requirements."

165 Of these eighteen items, all but three have been required to be completed by all operating reactors, albeit on a different
(continued next page)

520. The Commission, in its Order of March 23, 1981 (CLI-81-3), stated that it believed that TMI-1 "should be grouped with reactors which have received operating licenses, rather than with the units with pending operating license applications" except where the Board finds to the contrary when the record so dictates. 13 N.R.C. ____, CLI-81-3, slip op. at 7. Pursuant to this directive from the Commission, the Staff reassessed its previous position that Licensee should be required to comply with all NTOL requirements outside the scope of the Commission's Order and Notice of Hearing. The Staff reviewed these eighteen NTOL requirements and concluded, based upon the safety significance of the requirements, that five of the NTOL items should remain as prerequisites for the restart of TMI-1. The Staff is now of the position that the remaining items should be implemented on the same schedule as other operating reactors, as set forth in NUREG-0737. Tr. 21,325-29 (Jacobs, Silver); Staff Ex. 11. The Staff has reviewed Licensee's actions taken in response to these five additional pre-restart requirements derived from NUREG-0694 and has concluded that the requirements of these items have been or will be met prior to restart. Id.; see also, paragraph 538, infra.

521. The Staff believes that the combination of the short-term Order items and the five NUREG-0694 recommendations

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schedule (see, for example, Tr. 15,650-51 (D. Ross, Capra)). Ross, ff. Tr. 15,555 at 9.

provide the most significant improvements in safety, and thus is proposing that these actions be implemented at TMI-1 prior to restart. Ross, ff. Tr. 15,555, at 12; Tr. 21,327-30 (Silver). Further, the Staff has reviewed Licensee's actions taken pursuant to the short-term Order items and the applicable NUREG-0694 items and has concluded that Licensee has completed the actions called for by these items (or will complete these actions prior to restart). Since those actions which are vital to safety will be completed prior to restart, the Staff has concluded that there is reasonable assurance that TMI-1 can be restarted without posing a threat to the public health and safety. Ross, ff. Tr. 15,555, at 6; Tr. 21,045, 21,118-19 (Silver); Staff Ex. 14 at 3; Staff Ex. 11.

522. The Staff anticipates that, as part of its on-going review of the Action Plan and its continual efforts to improve plant safety, additional requirements will be issued to all licensees, including TMI-1. Such was the case with NUREG-0737, "Clarification of TMI Action Plan Requirements," issued by the Staff in October, 1980. NUREG-0737 provides a preliminary clarification of all Action Plan recommendations which have been approved for implementation by the Commission; modifications to the scope and/or schedule of several previously issued Action Plan and long-term Order items are included in NUREG-0737.[166] The Staff has stated that these modifications

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166 The Board notes that many of these items are outgrowths of NUREG-0578 recommendations and IE Bulletin requirements and have been the subject of previous clarifications issued by the Staff.

will now be binding on applicants and licensees, and supersede the requirements of its predecessor documents. NUREG-0737 has also imposed an additional eleven TMI-related requirements for TMI-1. Ross, ff. Tr. 15,555, at 10, 11; Staff Ex. 14 at 3.

523. The Staff, pursuant to the Commission's guidance to treat TMI-1 as an operating reactor, has stated that TMI-1 will be required to implement the additional NUREG-0737 requirements on the same schedule as other operating reactors. Several of the implementation dates for NUREG-0737 items fall due before the earliest estimated restart of TMI-1 (October, 1981); the Staff, therefore, has taken the position that these items should be completed prior to restart.[167] Ross, ff. Tr. 15,555, at 11; Tr. 21,049 (Silver); see generally, Staff Ex. 12. The Board notes that these NUREG-0737 items are not being required by the Staff prior to restart because of their safety significance, but simply because the proposed implementation dates fall due prior to the estimated time of restart.[168] Tr. 21,048-49, 21,323-24 (Silver).

524. The last group of restart requirements which have been proposed by the Staff are those recommended by the

167 See paragraphs 526 through 528, infra, for the Board's findings on several of these items.

168 NUREG-0737 implementation dates after June 30, 1981, have not yet been reviewed by the Staff, and are subject to being amended generically. If the dates for those NUREG-0737 items which the Staff has identified as prerequisites for restart are extended past the estimated restart date, the Staff would no longer consider these items to be restart requirements. Tr. 21,049, 21,136-38 (Silver); Staff Ex. 14 at 12, n.(1).

Division of Human Factors Safety of the Office of Nuclear Reactor Regulation, as documented in NUREG-0752 and Supplement No. 1 thereto (Staff Exhibits 2 and 15, respectively). Based upon its review of the TMI-1 control room and the control room/human factors documentation submitted by Licensee (see Section II.N, supra, for a complete examination of this issue), the Staff has recommended that certain modifications be implemented by Licensee prior to restart, prior to escalation beyond 5% power or as long-term modifications. The Staff believes the modifications which it has recommended to be implemented prior to restart and prior to escalation above 5% power will sufficiently reduce the potential for operator error leading to serious consequences due to human factors deficiencies to permit restart of TMI-1.[169] Staff Ex. 15 at 12, 13.

525. The Board, upon review of the Staff's proposed restart requirements, agrees generally with the Staff's view that the combination of the short-term Order items with certain of the additional pre-restart items derived from NUREG-0694 and NUREG-0752 comprise that subset of post-TMI requirements which are of sufficient importance to require that they be satisfactorily completed prior to restart of TMI-1. We do, however,

169 At the time that Staff Exhibit 15 was published, Licensee had not yet committed to implement all of the recommended modifications; however, at the May 14, 1981 hearing session, Licensee's counsel committed, on behalf of Licensee, to implement the modifications in accordance with the schedule set out at page 13 of Staff Exhibit 15. Tr. 21,431-32.

question the basis for the Staff's imposition of certain other recommendations, as discussed more fully below. To the extent that any of the Staff's proposed pre-restart requirements are not addressed below and, subject to the limitations expressed in n.168, supra (i.e., that the implementation dates of several of the NUREG-0737 pre-restart items are subject to being generically extended), the Board finds that these actions are necessary and sufficient to provide reasonable assurance that TMI-1 can be restarted without endangering the public health and safety, and therefore finds that Licensee must implement these items prior to restart.

526. Staff Exhibit 12 documents the Staff's evaluation of Licensee's compliance with those items in NUREG-0737 whose current implementation date falls due prior to October 1, 1981 (see n.168, supra). The Board initially notes that the Staff has not made a great deal of progress in evaluating the responses of other operating reactors, i.e., while the Staff has reviewed Licensee's compliance with all NUREG-0737 items due for implementation prior to October, 1981, only a few of those items have been evaluated for any other operating reactor. Further, Staff witness Jacobs knew of no instance in which the Staff had imposed any of the NUREG-0737 items as license conditions for any other operating reactor. Tr. 21,433-34 (Jacobs).

527. NUREG-0737 items II.K.2.14/II.K.3.7 required all B&W licensees to provide analyses documenting that the PORV

will open in less than 5% of all anticipated overpressure transients. Licensee submitted the requested analysis (performed generically by B&W for all B&W licensees); however, as documented in Staff Exhibit 12, the Staff has requested that Licensee provide, prior to restart, additional information in order to respond to Staff concerns regarding the data base utilized in the analysis. Staff witnesses Jacobs and Silver testified that no other B&W operating reactors have been reviewed for compliance with this item, nor has the Staff communicated its concerns about the analysis to any other B&W licensees, nor has any enforcement action been taken against any other B&W licensee with respect to this item. Tr. 21,436-37 (Jacobs, Silver); Staff Ex. 12 at II.K.2.14-1 through -3. On this basis alone, the Board could find that the Staff has discriminated against TMI-1 (in light of the Commission's Order, CLI-81-3) by requiring that Licensee submit this additional information before it would be allowed to restart, while at the same time allowing similar B&W plants to remain in operation. The Board further notes, however, that the Staff witnesses testified that the submission of this additional information will not affect the safe operation of the plant and that, if the same criteria were being used to evaluate TMI-1 as were being applied to other plants, the submission of the initial analysis would constitute reasonable progress toward completion of this item. Tr. 21,438 (Silver), 21,441 (Jacobs). For these reasons, then, the Board finds that completion of this item is not required in order to provide reasonable assurance that TMI-1 can be restarted without endangering the public health and that safety, and that therefore Licensee need not be required to complete this item prior to restart.

528. A situation similar to that described in paragraph 527, supra, exists with respect to NUREG-0737 item II.K.3.2, which required the submission by all licensees of an analysis of the probability of a small break loss-of-coolant accident caused by a stuck-open PORV and of safety-valve failure rates. The identical set of facts is involved with this item as with item II.K.2.14 (i.e., the B&W report submitted by Licensee caused the Staff to request additional information; the Staff has not reviewed other B&W licensees for compliance, nor initiated any type of enforcement action; and, the Staff witnesses do not believe that the submittal of the requested additional information prior to restart will have an impact upon the public health and safety). Tr. 21,438-41 (Jacobs); Staff Ex. 12 at II.K.3.2-1 through -4. Therefore, the Board finds that the completion of this item prior to restart is not necessary in order to reasonably assure that the public health and safety is not endangered.[170]

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170 NUREG-0737 item II.K.3.1 calls for the submittal of design documentation for an automatic PORV block valve closure system by July 1, 1981, if such a system is found to be necessary based upon the analysis conducted pursuant to item II.K.3.2. Staff Ex. 12 at II.K.3.1-1. In that the Board has found that item II.K.3.2 need not be completed prior to restart, it follows then that item II.K.3.1 also need not be completed prior to restart.

529. The Commission's August 9, 1979 Order and Notice of Hearing included both short-term and long-term recommended actions. In addition to these long-term Order recommendations, the Staff has proposed a number of other long-term TMI-2 related recommendations.[171] The long-term actions are not as narrowly defined, specific or urgent in nature as the pre-restart requirements. Certain of these items will require detailed and complex engineering analyses prior to identifying any additional modifications to plant systems or components which may be necessary; others require procurement of components or systems which are still in the developmental stage; still others require research studies or rulemaking on the part of the NRC. It is the Staff's belief that such long-term items need not be completed prior to restart in light of the enhanced margin of safety provided by the short-term requirements. Further, the Staff believes that deliberate, planned improvements would be preferable to imposing short-term actions that have not been well thought out. Ross, ff. Tr. 15,555, at 6; see, e.g., paragraph 541, infra, and Tr. 15,587 (D. Ross). The completion of these longer-term actions will result in a gradually increasing improvement in safety as they are completed and the initial, short-term modifications are

171 These recommendations are documented in the Action Plan and NUREG-0737. A number of the long-term actions contained in the Commission's Order and Notice of Hearing of August 9, 1979, are incorporated within the scope of these additional Action Plan items (see Ross, ff. Tr. 15,555, Table 2).

replaced or supplemented by long-term, more durable improvements. Ross, ff. Tr. 15,555, at 6.

530. With respect to those long-term actions which are included in the Commission's August 9, 1979 Order and Notice of Hearing, the Board notes that the completion dates for some of these actions have changed since the Commission's Order. As originally recommended, the completion dates for certain Category B items in Table B-1 of NUREG-0578 would have become due prior to the estimated restart of TMI-1. The completion dates now recommended by the NRC Staff for these items are those established by NUREG-0737 for other operating reactors, many of which fall due well after the expected date of restart. Completion of these items would be recommended prior to restart only if restart occurs after the completion dates specified in NUREG-0737. Ross, ff. Tr. 15,555, at 11; see also n.168, supra. In light of the Commission's Order of March 23, 1981 (CLI-81-3), which expressed the Commission's intention that NUREG-0737 implementation schedules be the same for TMI-1 as other operating reactors, the Board accepts the Staff's present recommendations for implementation of these items.

531. The Staff has assessed, in its safety evaluation reports and supplements, whether Licensee has demonstrated "reasonable progress" towards completion of the recommended long-term actions. The concept of what constitutes reasonable progress and the criteria utilized by the Staff in

determining whether Licensee has made reasonable progress has been the subject of extended examination by the Board and parties in this hearing. See, e.g., Tr. 10,880-83; 15,594, 15,970-79; 15,985-87; 16,019-31; 21,042-50; 21,207-09; 21,434-35. The Staff, in Supplement 3 to NUREG-0680, has stated that "[r]easonable progress toward completion of the long-term actions required by the Order for TMI-1 will be considered to be a degree of progress consistent with that of the other operating reactors, except as noted in individual evaluations, so that there is reasonable assurance that the action will be completed on the NUREG-0737 schedule as it may be amended." Staff Ex. 14 at 3; Tr. 21,042-44 (Silver).

532. As the Commission held in its Order of March 23, 1981 (CLI-81-3), TMI-1 is to be treated the same as other operating reactors with respect to the NUREG-0737 implementation schedules (unless the record dictates to the contrary). Further, the Commission stated that it intended to retain its flexibility with regard to these implementation dates, where developments occur which could affect the ability of Licensee to comply with the requirements recommended by this Board or imposed by the Commission. CLI-81-3, slip op. at 7, 8. Based upon this guidance by the Commission, with which the Board concurs, the Board views the criteria expressed by the Staff (see paragraph 531, supra) as an appropriate means by which to judge whether Licensee has made reasonable progress toward the completion of the long-term Order items.[172]

172 The Staff has not followed this general definition of reasonable progress in one case. See section II.B (Detection of Inadequate Core Cooling), supra.

533. In conjunction with the examination conducted by the parties with respect to the Staff's findings of reasonable progress and reasonable assurance that TMI-1 is sufficiently safe to allow restart, questions were raised regarding the Staff's reliance on Licensee's "commitments," the enforceability of those commitments and the need for license conditions to assure that these commitments are carried out. See, e.g., Tr. 21,145-53, 21,282-92 and 21,350-56.

534. Initially, the Board would note that, with respect to any concerns the parties may have regarding Licensee's commitments to implement any of the short-term items contained in the Commission's Order and Notice of Hearing, the Order itself requires that the Director of Nuclear Reactor Regulation must certify to the Commission that all short-term actions have been satisfactorily completed. CLI-79-8, 10 N.R.C. 141, 148-149 (1979).

535. It is with respect to the long-term actions committed to by Licensee that the question of enforceability is of more concern. The Staff has stated that it will assure that all licensee commitments made in response to the NUREG-0737 requirements are appropriately enforceable and will, as needed, issue Confirmatory or Show Cause Orders to enforce these items. Ross, ff. Tr. 15,555, at 10. Further, the Staff's project manager for TMI-1 assures us that, to the extent that the Staff's conclusions concerning the operation of TMI-1 are based upon Licensee's commitments, where any changes to those

commitments would cause a concommitant change to the Staff's conclusion, the Staff would "most definitely" take appropriate enforcement action to assure that the Staff's conclusions could be substantiated. Tr. 21,168-89 (Silver).

536. The Commission's August 9, 1979 Order and Notice of Hearing expressly granted this Board the authority, similar to that provided in 10 C.F.R. §50.57(b), to impose such limitations or conditions on the restart of TMI-1, with respect to any uncompleted items, as it believes necessary to protect the public health and safety. CLI-79-8, 10 N.R.C. 141, 148-149 (1979). The Board does not believe it necessary, however, to impose as license conditions the requirement that specific long-term modifications must be completed by a date certain. Our reasons for this position are two-fold: (1) the Staff has provided adequate assurance that the commitments to implement the required long-term modifications will be enforced (see paragraph 535, supra); (2) the Commission's Order of March 23, 1981, left to the Commission itself the flexibility to consider, on a case-by-case basis, any developments which impact Licensee's ability to comply with the implementation dates imposed by the NUREG-0737 requirements. CLI-81-3, slip op. at 7, 8. Therefore, the Board rejects the proposition that dates certain for the completion of these actions should be imposed as license conditions.

537. The Board turns now to consideration of the specific individual actions which are referenced in Board Questions 1, 3 and 7.

538. Board Question No. 1 requested the Staff to provide its position as to the applicability of NUREG-0694 and NUREG-0660 to TMI-1. This subject is discussed generally in paragraphs 517 through 523, supra. The Board also indicated its particular interest in the status and applicability of the following four Action Plan items. Item I.D.1 in NUREG-0694 calls for NTOLs to perform a preliminary control room design review in order to identify significant human factors and instrumentation problems and establish a schedule for correcting these deficiencies.[173] Licensee has conducted a control room/human factors review (see section II.N, supra), which the Board believes meets the requirements of this item. See also, Staff Exs. 11 and 15. NUREG-0737, item I.D.1, will require all licensees to perform a detailed control room design review; this detailed review is not required prior to restart. Staff Ex. 15 at 5 and App. I. Item II.E.1.1, Auxiliary Feedwater System, contains both long- and short-term components; the Staff has determined that Licensee has completed the short-term items and has made reasonable progress toward the completion of the long-term actions. Tr. 15,562-65 (Capra); Staff Ex. 1 at C1-1 through C1-11; Staff Ex. 14 at 13-14 and Tables B-1 and B-2. A detailed discussion of the modifications being made to Licensee's emergency feedwater system can be found at paragraphs 395 through 401 of our consideration of Board Question

173 NUREG-0694 item I.D.1 is one of the five NTOL requirements which the Staff has proposed as a pre-restart requirements. See paragraphs 518 through 520, supra.

No. 6 (Section II.Q). Item II.B.8 calls for the Staff to initiate a rulemaking proceeding on degraded core considerations. Subsequent to the promulgation of this question, the Commission published its Advanced Notice of Proposed Rulemaking on the Consideration of Degraded or Melted Cores in Safety Regulation (45 Fed. Reg. 65474, October 2, 1980). Item II.B.7 calls for an analysis of hydrogen control measures. We inquired further as to whether the Sequoyah "spark plug" proposal is applicable to TMI-1. Although no testimony was presented on the applicability of the Sequoyah proposal to TMI-1,[174] we are informed that the spark plug method of hydrogen control is not applicable to TMI-1. See Tr. 15,759-60 (Cutchin). Further, we note that the Commission on October 2, 1980 published its Proposed Rule on Interim Requirements Related to Hydrogen Control and Certain Degraded Core Conditions, which requires licensees to perform analyses of the ability of their containment structures to withstand uncontrolled hydrogen-oxygen recombination without the loss of safety function or to show that such recombination would not take place in the containment. 45 Fed. Reg. 65466 at 65472. The Board finds that the actions taken by the Staff and Licensee with respect to these items adequately address our concerns.

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174 The pre-filed direct testimony of Robert W. Reid on behalf of the NRC Staff in response to Board Question Nos. 1 and 5, which did specifically respond to this item, was not offered into evidence.

539. Board Question No. 3 inquired as to the status of the Staff's Interim Reliability Evaluation Plan ("IREP") (in particular, the IREP study performed on Crystal River-3), its applicability to TMI-1, and whether IREP will assess the adequacy of the modifications proposed for implementation at B&W plants. A draft of the IREP report on Crystal River-3 ("CR-3") was submitted to the Staff in May, 1980. However, reviews of that report discovered certain deficiencies (such as IREP's inability to predict loss of NNI/ICS power supply events similar to those which have occurred at Rancho Seco and CR-3) in the methodology employed by IREP. The contractor who performed the study, Science Applications, Inc., has been requested to revise the report and the Staff expects ultimately to publish the results of the CR-3 study. Rowsome, ff. Tr. 16,907, at 2, 3; Tr. 16,908-10, 16,913, 16,920 (Rowsome).

540. The identification of the deficiencies in the CR-3 study did prove helpful to the Staff in evaluating the adequacy of the IREP approach and has resulted in a modification of IREP procedures to assure that such weaknesses do not recur in future studies. Tr. 16,913 (Rowsome). Four additional IREP studies,[175] begun in September, 1980, have incorporated these revised procedures. Tr. 16,908 (Rowsome).

541. Staff witness Rowsome testified that the event tree/fault tree technique currently being utilized in the IREP

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175 One of the four plants being studied is a B&W reactor, Arkansas Nuclear One, Unit 1. Tr. 16,908 (Rowsome).

studies will be a proper method by which to investigate interactions between safety and non-safety systems, a subject in which both the Board and the ACRS have expressed great interest. Tr. 16,914 (Rowsome); see also Staff Exhibit 14 at 54 and Appendix C. However, the Staff is also investigating other methodologies for performing systems interactions studies and will develop a policy on the best method for conducting such studies prior to requiring licensees to conduct systems interactions studies. Staff Ex. 15 at 54; Tr. 15,615-18 (D. Ross), 16,915 (Rowsome). One of the principal goals of the IREP program is the development, "debugging" and trial use of standard procedures for performing studies of systems interactions and multiple failure scenarios. Tr. 16,915 (Rowsome). As Dr. Ross pointed out, it would be premature to order licensees to perform these studies prior to the Staff endorsing a single best method. Tr. 15,618 (D. Ross).

542. At the present time, the Staff has not formally issued a requirement that each licensee perform an IREP-type study of their plants. Any such decision will be made after the Office of Research has developed a standard set of procedures and a determination of what constitutes an adequate method of performing such studies; it is expected that the Office of Research will complete this task by the end of 1982. The Office of Nuclear Reactor Regulation will then determine the plants which will be requested to perform the next phase of IREP-type studies and the schedules for such studies. Tr. 16,923 (Rowsome).

543. The IREP studies currently being conducted do not assess the adequacy of the proposed actions for B&W plants, nor has there been any probabilistic risk assessment to determine if the lessons learned requirements are necessary and sufficient. Rowsome, ff. Tr. 16,907, at 3; Tr. 16,928 (Rowsome). However, the Board notes that the IREP studies performed to date have not discovered any potential failure mode that has not been addressed in the modifications being undertaken at TMI-1. Tr. 16,924 (Rowsome).

544. Based upon our review of the record, the Board believes the Staff should continue its efforts to develop an appropriate method which would allow adequate investigations to be conducted of systems interactions; further, the Board concurs with the Staff that requiring such a study at TMI-1 prior to the development of a standard methodology would be premature and, most likely, insufficient.

545. Board Question No. 7 requested the Staff to identify those recommendations contained in Table 2.1 of the Staff's investigation of the Crystal River-3 transient, published as NUREG-0667, Transient Response of Babcock & Wilcox - Designed Reactors, which should be implemented at TMI-1.

546. The study documented in NUREG-0667 is a deterministic review of the transient response of B&W reactors, which resulted in the development of a number of recommendations which have been presented for consideration to the Director of Nuclear Reactor Regulation. Tr. 15,784 (Capra).

Under the management structure in the Office of Nuclear Reactor Regulation, the Director was then required to endorse or reject the recommendations contained in the report. The Director requested the Division of Safety Technology ("DST") to evaluate the safety significance of the recommendations and recommend whether they should be implemented. DST reviewed, among other factors, the risk reduction potential of each recommendation and devised the following prioritization system for implementing these recommendations:

- Priority one recommendations, which should be implemented as soon as possible;
- Priority two recommendations, which should be implemented consistent with existing priorities and resources;
- Priority three recommendations, which DST did not believe would make a significant contribution to reactor safety and therefore should not be implemented.

Tr. 15,786-87 (D. Ross); Staff Ex. 9.

547. DST's recommendations were forwarded to the Director of NRR on August 3, 1980; in September, 1980, the Director endorsed the prioritized implementation scheme developed by DST and authorized the Division of Licensing to proceed with the implementation of these recommendations. At the time that Dr. Ross testified in this proceeding, the implementation plan had not yet been completed or forwarded to licensees. Tr. 15,786-87 (D. Ross).

548. In order to respond to Board Question No. 7, Staff witness Capra prepared a chart, Staff Exhibit 9, which indicates the relationship of the NUREG-0667 recommendations to the requirements of NUREG-0737 or other actions being undertaken by the Staff or Licensee. Tr. 15,788 (D. Ross); Staff Ex. 9. As Staff Exhibit 9 shows, all Priority One items which are applicable to TMI-1 have either previously been completed, or are being implemented in conjunction with Licensee's response to the NUREG-0737 requirements, or have been committed to be implemented in response to other Staff requirements. With respect to the twelve Priority Two items, the four which are applicable to TMI-1 have been or are being implemented by Licensee; the eight remaining items are the subject of Staff actions, for which evaluations are being performed by the Staff as part of its Action Plan studies. Staff Ex. 9.

549. In view of the evidence presented on this matter, the Board concludes that the Staff has taken appropriate action to assure that those safety significant NUREG-0667 recommendations which can contribute to the protection of the public health and safety are being implemented at TMI-1 in a timely fashion.

550. The Board, in promulgating these questions, realized that their scope went to the totality of the hardware and procedural modifications being implemented at TMI-1 in response to the recommended industry-wide improvements which have grown out of the investigations of the TMI-2 accident.

While the record compiled on these modifications was lengthy, we believe that these questions have provided an opportunity for an overview of the sufficiency of the Staff's response to the lessons learned from the TMI-2 accident. Based upon our review of the record, both in response to these particular Board Questions and of the totality of evidence presented on plant design and procedures issues, the Board finds that the requirements recommended by the Staff have identified all the significant lessons learned from the TMI-2 accident and that the implementation of these requirements at TMI-1 will provide reasonable assurance that the plant can be restarted and operated without endangering the health and safety of the public. Additionally, the Board concludes that those items which the Staff has proposed as prequisites for restart (with the exceptions and limitations noted in paragraphs 525, 527 and 528, supra) are, indeed, the most safety-significant of the requirements recommended by the Staff and therefore must be implemented by Licensee prior to restart in order to provide reasonable assurance that TMI-1 can be restarted without endangering the health and safety of the public. Further, the Board finds that the long-term actions recommended by the Staff, with the exception of the Staff's recommendation calling for the installation of a reactor vessel water level instrumentation system (see section II.B, supra), will provide an additional margin of safety and, thus, the Board finds that the long-term actions are necessary and sufficient to provide reasonable assurance that TMI-1 can be operated on a long-term basis without endangering the public health and safety.

Respectfully submitted,

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