UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

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

NRC STAFF TESTIMONY OF J. WERMIEL, W. JENSEN, E. LANTZ, AND B. BOGER REGARDING EMERGENCY FEEDWATER SYSTEM RELIABILITY (BOARD QUESTION 6)

(Tr. 2394-96) Emergency Feedwater Reliability

Question 6a. 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?

(Witness Wermiel)*

Response: No. The loss of emergency feedwater following a main feedwater transient is not an accident which must be protected against with safety-grade equipment.

With respect to the Oconee Incident, a complete loss of all feedwater will not result from a failure in the integrated control system/non-nuclear instrumentation (ICS/NNI), since modifications will be made to the EFW system (as described

*Note: Name in parenthesis indicates preparer of response.

in the TMI-1 Restart SER, NUREG-0680, pages C8-35, 36 and 37) that will completely eliminate any intertie between the ICS/NNI and EFW systems. These modifications will upgrade the EFW system to a fully safety-grade system. Prior to installation of the fully safety-grade system, an Oconee type event may result in a momentary loss of EFW. However, this situation would be detected by the operator through the EFW flow indicators and steam generator level indication which are separate from the NNI as described in the TMI-1 Restart SER, NUREG-0680. The operator can then take the necessary manual action in the control room to open the EFW control valves and restore feed flow.

(Witness Jensen)

In the unlikely event that both main and emergency feedwater cannot be restored, the operator at TMI-1 is instructed to actuate High Pressure Injection as described in Inadequate Core Cooling Emergency Procedure EP1202-39. This action will provide adequate cooling to the core by feed and bleed if two high pressure injection pumps are available. The high pressure injection system operates independent of NNI/ICS.

Question 6b. In what respect is the emergency feedwater system vulnerable to non-safety-grade system failures and to operator errors?

(Witness Wermiel)

Response: The emergency feedwater system currently meets all requirements of a safety-grade system and is, therefore, not vulnerable to non-safety-grade system failures with one exception. A single failure in the non-safety-grade integrated control system could result in a loss of flow by closure of the flow control valves. Implementation of the safety-grade modification to the automatic initiation design, which will eliminate the single failure

possibility, will correct this deficiency. In addition, a question has been raised by the licersee in Licensee Event Report 80-012/0IT-0 dated July 11, 1980, concerning the adequacy of the environmental qualification for the emergency feedwater system components in the event of a main steam line break in the Intermediate Building. This area is being reviewed in connection with IE Bulletin 79-01B.

With regard to the vulnerability of the EFW system to operator errors, the EFW system is vulnerable to operator errors as is any safety-grade system. For example, the operator may leave valves closed or turn off pumps. However, improved emergency and operating procedures coupled with operator training in these procedures (as described in the TMI-1 Restart SER, NUREG-0680) should limit the possibility of operator error.

Question 6c. 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?

(Witness Lantz)

Response: We have reviewed the Licensee Event Reports for plants with safety-grade emergency feedwater (EFW) systems. The available data for plants that are in commercial operation and that have safety-grade emergency feedwater systems show that in the vast majority of the cases, the failures that occurred did not defeat the functional capability of the system. Data

on EFW system success on demand is not maintained. However, it should be noted that all plants perform routine periodic EFW system surveillance testing in accordance with specific plant Technical Specifications. Except for the following reported cases of common cause and operator-induced failure, which resulted in an overall system failure, the system was found to be operable when initiated. These are cases where sufficient emergency feedwater was not available, although emergency feedwater was not required at the time to cool the reactor.

Plant	Date	LER#	Description
Ginna	12/14/73	73-01T	The suction was lost on two emergency feedwater pumps during a startup test because of the lack of air vents.
Turkey Point 3	05/08/74	74-01T	During a start test two emergency feedwater pumps failed to start due to tight packing. A third feedwater pump started but tripped because of foreign matter in the governor.
Kewaunee 1	11/05/75	75-01T	During startup operations resin beads clogged the strainer to all emergency feedwater pumps.
Haddam Neck	07/05/76	76-03L	When the plant was in the startup mode, both emergency feedwater pumps were vapor bound due to a leaky check valve.

It should be noted that prior to commercial operation at the Davis Besse plant, there was one event in which the safety grade emergency feedwater system failed on demand.

Question 6d. What operator action is required to operate in a feed-and-bleed mode following a loss of emergency feedwater?

(Witness Boger)

Response: According to EP 1202-26A, "Loss of Steam Generator Feedwater to Both OTSGs," if both main and emergency feedwater are lost, the operator is directed to initiate High Pressure Injection (HPI). HPI is actuated by depressing two pushbuttons on the main control board. The next operator action is to verify starting of the HPI pumps and opening of the HPI discharge valves. The operator is then instructed to verify that the PORV block valve is open and that the PORV is cycling to maintain RCS pressure at approximately 2450 psig. At this point the operator has provided a makeup supply to the RCS (feed) and a relief path to remove core heat (bleed). If the PORV or PORV block valve fail to respond (open), the RCS safety valves will relieve at 2500 psig to provide the bleed path.

Subsequent operator actions include throttling the HPI discharge valves to maintain at least a 50°F margin of subcooling and attempting to restore a feedwater supply to the steam generators from the main or emergency feedwater system or the condensate system.

Question 6e. 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 PORVs and safety valves have not been tested under two-phase mixtures.

(Witness Jensen)

Response: The answer to this question is contained in the NRC response to UCS Contention 1, questions 15, 16, and 17.

Question 6f. 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?

(Witness Jensen)

Response: Analyses by Babcock and Wilcox indicate two high pressure injection pumps are required to adequately cool the core by feed-and-bleed for the first three hours if decay heat is calculated using 1.2 times the ANS-5 decay heat model as required for LOCA analysis in Appendix K to 10 CFR 50. After three hours only one pump would be required. If a best estimate decay heat model is utilized (1.0 times the ANS-5 decay heat model) the analyses indicate that only one HPI pump would be required. See letter from J. Taylor, B&W, to R. Mattson, NRC, May 12, 1979, which transmits Volume 1 Section 6.0 - Supplements 1 and 2 to the "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in a 177 Fuel Assembly Plant."

We have not requested nor has Metropolitan Edison provided us with either procedures or analyses for cooldown of the reactor coolant system by feed-and-bleed, nor have we performed such evaluations. We, therefore, do not know whether feed-and-bleed can be utilized to achieve cold shutdown. However, sufficient water is available in the borated water storage tank (BWST) for at least 19 hours of feed-and-bleed operation with two HPI pumps. 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 operator throughout this time would be to reestablish 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.

See the NRC response to UCS Contention 1 question 17 for a discussion of experience using feed-and-bleed.

Question 6g. 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?

(Witness Jensen)

Response: One EFW pump can supply adequate feedwater for decay heat removal for all postulated accidents and transients. Therefore, the single failure criterion is satisfied. A single motor-driven emergency feedwater pump would have the capability to deliver 460 gpm to the steam generators. As discussed on page C1-4 of NUREG-0680, the NRC evaluated the effect of supplying 500 gpm of emergency feedwater following a loss of main feedwater. The conclusion of this evaluation was that although the PORV might be opened by high reactor coolant pressure, the amount of coolant loss would be bounded by the small-break LOCA analyses. The LOCA analysis performed for a stuck-open unisolated PORV indicated that no core uncovery or damage would occur if only 300 gpm

of emergency feedwater were available. This analysis is described in the B&W report "Evaluation of Transient Behavior and Small Reactor Coolant System Breaks in the 177 Fuel Assembly Plant, May 7, 1979, Section 6.2.3" and Supplement 3 to this document dated May 24, 1979. The coolant lost from a PORV cycling on high pressure would be less than that from a stuck open PORV. Since the stuck open PORV LOCA analysis would bound a loss of feedwater event with 460 gpm of emergency feedwater, we conclude that the flow of one motor-driven emergency feedwater pump would be adequate to cool the reactor core.

Question 6h. Can the turbine-driven pumps and valves be operated on Direct Current, or are they dependent upon the Alternating Current safety buses?

(Witness Wermiel)

Response: The TMI-1 turbine-driven EFW train consists of one turbine-driven pump and its associated flow path (including valves). This train can operate to supply feedwater on direct current power sources only as described in the TMI-1 Restart SER, NUREG-0680, page C1-9 and 10.

Question 6i. 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?

(Witness Wermiel)

Response: Based on knowledge of the improvement in reliability gained by eliminating first order failure sources, it is the staff's judgement that the reliability of the emergency feedwater system will be improved once the fully safety-grade system is installed. The single failure problem associated

with the integrated control system/non-nuclear instrumentation described in the response to 6a and b above will be eliminated. In addition, various other hardware, procedural and administrative improvements as identified in the TMI-1 Restart SER, NUREG-0680 under Order Item 1a. should enhance emergency feedwater system reliability. However, a quantitative reassessment of the reliability of the fully safety-grade EFW system has not been performed. The feed-and-bleed back-up is not required by the staff and, therefore, need not meet all requirements of a safety system. However, it is recognized as additional defense in depth for providing core cooling in the very unlikely event that emergency feedwater is lost, and is, therefore, required to be available.

Question 6j. Will the short-term actions proposed improve the reliability of the emergency feedwater system to the point where restart can be permitted?

(Witness Wermiel)

Response: Yes. The proposed short-term modifications as described in the TMI-1 Restart SER will improve emergency feedwater system reliability to the point where restart can be permitted. This is discussed in detail in the TMI-1 Restart SER, NUREG-0680, page C8-37.

Question 6k. 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.

(Witness Wermiel)

Response: A loss of emergency feedwater (concurrent with an accident or transient) is not a design basis event for the following reasons:

(a) In ALAB-603, the Board refers to Standard Review Plan Section 2.2.3 as the criterion for acceptability of the plant design to mitigate the assumed event. This criterion was intended by the staff to be applied only to external plant hazards such as nearby transportation of toxic gases or explosives, and not events within the plant. It is, therefore, not applicable to a postulated loss of emergency feedwater.

(b) The staff performs a deterministic review of the emergency feedwater system to verify compliance with applicable General Design Criteria.

(c) Additionally, we have applied reliability techniques as a tool for improving emergency feedwater (EFW) system reliability. The results of this evaluation were used to identify and correct the primary sources of system unreliability. Requirements were determined based on these reliability studies. This effort has been included in our review of TMI-1 for restart, and is discussed in the TMI-1 Restart SER, NUREG-0680.

(d) THE NRC has not established a numerical safety goal at this time. Work in this area is proceeding as described in SECY-80-379, "Proposed Plan for Developing a Safety Goal," dated August 12, 1980. Until guidance is established, reliability studies will continue to be used as indicated in (c) above.

(e) The single failure criterion continues to be employed for design basis events as indicated in SECY-77-439, "Single Failure Criterion," dated August 17, 1977. Additionally, probabilistic methods will be used as a tool for further insight such as described in (c) above.

(f) Based on the EFW system design and the modifications to be implemented as described in the TMI-1 Restart SER, NUREG-0680, we believe that further additional hardware changes will not significantly improve EFW reliability. The common cause failure mode as a result of operator error still remains as the dominant source of system unreliability. This failure mode is being further minimized with improvements in the human factors aspects of the plant, i.e., improved operating and emergency procedures, improvements in instrumentation, and continuous operator training.

Jared S. Wermiel

Professional Qualifications

Auxiliary Systems Branch
Division of Systems Integration
Office of Nuclear Reactor Regulation

I am a Reactor Engineer in the Auxiliary Systems Branch in the Division of Systems Safety, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission. In this position I perform technical reviews, analyses, and evaluations of reactor plant features pursuant to the construction and operation of reactors.

I received a Bachelor of Science Degree in Chemical Engineering from Drexel University in 1972. Since 1972 I have taken courses on PWR and BWR System Operation, Reactor Safety, and Fire Protection.

My experience includes seven years with the Bechtel Power Corporation as a Systems Design Engineer engaged in the design of various nuclear power plant auxiliary and balance of plant systems. These have included cooling water systems, water treatment systems and fire protection systems.

I joined the Auxiliary Systems Branch of the Commission in March, 1978. Since joining the Commission I have performed safety evaluations on nuclear power plant auxiliary systems including auxiliary feedwater systems for the Virgil C. Summer Nuclear Station, Palo Verde Nuclear Generating Station, Waterford Steam Electric Station, Diablo Canyon Nuclear Power Plant, Byron/Braidwood Stations and Trojan Nuclear Plant. I have also reviewed various topical reports and provided comments on proposed ANSI Standards dealing with various auxiliary systems.

I have responsibility for the review of the following nuclear power plant auxiliary systems and concerns: new and spent fuel storage, spent fuel pool cooling, fuel handling, service water, component cooling water, condensate storage, ultimate heat sink, instrument air, chemical and volume control, main steam isolation valve leakage control, heating ventilating and air conditioning, portions of the main steam system, main feedwater, auxiliary feedwater, high and moderate energy pipe breaks, flood protection and internally generated missiles.

I am a registered Professional Engineer in the State of Maryland.

I am an Associate Member of the American Institute of Chemical Engineers.

WALTON L. JENSEN, JR.

PROFESSIONAL QUALIFICATIONS

I am a Senior Nuclear Engineer in the Reactor Systems Branch of the Nuclear Regulatory Commission. In this position I am responsible for the technical analysis and evaluation of the public health and safety aspects of reactor systems.

From June 1979 to December 1979, I was assigned to the Bulletins and Orders Task Force of the Nuclear Regulatory Commission. I participated in the preparation of NUREG-0565, "Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox Designed 177-FA Operating Plants."

From 1972 to 1976, I was assigned to the Containment Systems Branch of the NRC/AEC, and from 1976 to 1979, I was assigned to the Analysis Branch of the NRC. In these positions I was responsible for the development and evaluation of computer programs and techniques to calculate the reactor system and containment system response to postulated loss-of-coolant accidents.

From 1967 to 1972, I was employed by the Babcock and Wilcox Company at Lynchburg, Virginia. There I was lead engineer for the development of loss-of-coolant computer programs and the qualification of these programs by comparison with experimental data.

From 1963 to 1967, I was employed by the Atomic Energy Commission in the Division of Reactor Licensing. I assisted in the safety reviews of large power reactors, and I led the reviews of several small research reactors.

I received an M.S. degree in Nuclear Engineering at the Catholic University of America in 1968 and a B.S. degree in Nuclear Engineering at Mississippi State University in 1963.

I am a graduate of the Oak Ridge School for Reactor Technology, 1963-1964.

I am a member of the American Nuclear Society.

I am the author of three scientific papers dealing with the response of B&W reactors to Loss-of-Coolant Accidents and have authored one scientific paper dealing with containment analysis.

PROFESSIONAL QUALIFICATIONS LIST

BRUCE A. BOGER

Education

June 1971	Received BSNE - University of Virginia
June 1972	Received MENE - University of Virginia

Work Experience

June 1972 to June 1977	Virginia Electric and Power Company Surry Nuclear Power Station
	Assistant Engineer - Performed startup testing on Unit No. 2.
	Engineer - Assisted the Supervisor-Engineering Services; trained for and received a Senior Reactor Operator License.
	Supervisor - Engineering Services - Directed the activities of the onsite engineering staff.
June 1977 to September 1977	Virginia Electric and Power Company Richmond, Virginia
	Supervisor - Nuclear Engineering Services - Directed the activities of the offsite engineering staff in support of Surry Power Station.
October 1977 to Present	U. S. Nuclear Regulatory Commission Bethesda, Maryland
	Reactor Engineer in the Operator Licensing Branch - Administer licensing examinations to nuclear power plant and research reactor personnel.

Professional Affiliations

Registered Professional Engineer - State of Virginia

Member - American Nuclear Society

Participation in TMI Activities

Bruce A. Boger

November 1978, April 1980: Administered operator license examinations on Unit One.

November 1978, March 1979, March 1980: Administered operator license examinations on Unit Two.

March - April 1979: Member of the TMI-2 emergency response team, assisted in the preparation of emergency and contingency procedures.

July 1979 - Present: Member of the TMI Technical Support Staff, conducted audit examinations on post-accident installed equipment on TMI-2. Also participated in the review of training and procedures in conjunction with the TMI-1 restart programs. This included preparation of SER inputs and testimony.

EDWARD LANTZ

DIVISION OF SAFETY TECHNOLOGY

U.S. NUCLEAR REGULATORY COMMISSION

PROFESSIONAL QUALIFICATIONS

As an Engineering Systems Analyst in the Operating Experience Evaluation Branch I am responsible for reviewing and evaluating experience related to the safety of nuclear power plant operation.

I have a Bachelor of Science degree in Engineering Physics from the Case Institute of Technology and a Masters of Science degree in Physics from Union College and a total of 30 years of professional experience, with over 20 years in the nuclear field. My experience includes work on reactor transients and safeguards analysis, nuclear reactor analysis and design, research and development on nuclear reactor and reactor control concepts and investigations of their operational and safety aspects.

I have held my present position with the Commission since May 1, 1980. My previous position, which I held for about four and one half years, was Engineering Systems Analyst in the Plant Systems Branch where I was responsible for technical reviews and evaluations of component and system designs and operating characteristics of licensed nuclear power reactors. Prior to that I was a Project Manager in the Gas Cooled Reactors Branch, Division of Reactor Licensing, U.S. Nuclear Regulatory Commission, where I was responsible for the technical review, analysis, and evaluation of the nuclear safety aspects of applications for construction and operation of nuclear power plants. For about ten years prior to that I was Head of the Nuclear Reactor Section in NASA. My section was responsible for the development and verification of nuclear reactor analysis computer programs, conceptual design engineering, and development engineering contracting. Prior to my employment with NASA, I was a nuclear engineer at the Knolls Atomic Power Laboratory for about six years, where I worked on the safeguards and nuclear design of the S3G reactors and the initial development of the nuclear design of the S5G reactors. Previous experience includes system engineering and electrical engineering with the General Electric Company and electronic development engineering with the Victoreen Instrument Company.