NUREG-0645
Vol. I

Report of the Bulletins and Orders Task Force

Office of
Nuclear Reactor Regulation

U.S. Nuclear Regulatory
Commission

Available from

GPO Sales Program
Division of Technical Information and Document Control
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

and

National Technical Information Service
Springfield, Virginia 22161

NUREG-0645
Vol. I

Report of the Bulletins and Orders Task Force

Manuscript Completed: January 1980
Date Published: January 1980

Division of Project Management
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

TABLE OF CONTENTS

VOLUME 1

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

VOLUME II

TABLE OF CONTENTS (Continued)

SUMMARY AND CONCLUSIONS

The accident at Three Mile Island Unit 2 (TMI-2) on March 28, 1979, involved a main feedwater transient coupled with a stuck-open pressurizer power-operated relief valve and a temporary failure of the auxiliary feedwater system. The resulting severity of the ensuing events and the potential generic aspects of the accident on other operating reactors led the NRC to initiate prompt action to: (a) assure that other reactor licensees, particularly those with plants similar in design to TMI-2, took the necessary action to substantially reduce the likelihood for TMI-2 type events, and (b) investigate the potential generic implications of this accident on other operating reactors.

The Bulletins & Orders Task Force was established within the NRC Office of Nuclear Reactor Regulation (NRR) in early May 1979 and discontinued operations on December 31, 1979. This task force was responsible for reviewing and directing the TMI-2-related staff activities associated with the NRC Office of Inspection and Enforcement (IE) bulletins, Commission Orders and generic evaluations of loss of feedwater transients and small-break loss-of-coolant accidents for all operating plants to assure their continued safe operation.

This report summarizes the results of the Bulletins & Orders Task Force review of IE bulletins, Commission Orders, and the NRR generic evaluation of feedwater transients, small-break loss-of-coolant accidents, and other TMI-2 related events in operating plants to confirm or establish the bases for their continued safe operation. The results of this evaluation are presented in this report.

The Bulletins & Orders Task Force generic review of the operating plants has resulted in the following conclusions:

(1) The continued operation of the operating plants is acceptable provided that certain actions related to the plants' designs and operation, and training of operators identified in this report are implemented consistent with the recommended implementation schedules.

(2) The actions taken by the licensees with operating plants in response to the IE bulletins (including the actions specified in NUREG-0623, "Generic Assessment of Delayed Reactor Coolant Pump Trip During Small Break Loss-of-Coolant Accidents in Pressurized Water Reactors") provide added assurance for the protection of the health and safety of the public.

In addition, the Bulletins and Orders Task Force has independently confirmed the safety significance of those related short-term and long-term actions recommended by other Office of Nuclear Reactor Regulation task forces as discussed in this report.

1.0 INTRODUCTION

1.1 Background

The accident at Three Mile Island Unit 2 (TMI-2) on March 28, 1979, involved a main feedwater transient coupled with a stuck-open pressurizer power-operated relief valve and a temporary failure of the auxiliary feedwater system. The resulting severity of the ensuing events and the potential generic aspects of the accident on other operating reactors led the NRC to initiate prompt action to: (a) assure that other reactor licensees, particularly those with plants similar in design to TMI-2, took the necessary action to substantially reduce the likelihood for TMI-2-type events, and (b) investigate the potential generic implications of this accident on other operating reactors.

The purpose of this report is to summarize the results of our evaluation of licensees' responses to Office of Inspection and Enforcement (IE) bulletins, Commission Orders, and the Office of Nuclear Reactor Regulation (NRR) generic evaluation of feedwater transients, small-break loss-of-coolant accidents (LOCAs), and other TMI-2-related events in operating plants to confirm or establish the bases for their continued safe operation. The results of this evaluation are presented in this report.

1.2 Bulletins & Orders Task Force

The Bulletins & Orders Task Force (B&OTF) was established within the NRR in early May 1979 and discontinued operations on December 31, 1979. The B&OTF was responsible for reviewing and directing the TMI-2-related staff activities on loss of feedwater transients and small-break loss-of-coolant accidents for all operating reactors to assure their continued safe operation. In conducting this activity, the B&OTF concentrated its efforts on: (1) the assessment of systems reliability, (2) the review of the analytical predictions of plant performance for both feedwater transients and small-break LOCAs, (3) the evaluation of generic operating guidelines, (4) the review of emergency plant operating procedures, and (5) the review of operator training.

The B&OTF worked directly with operating plant licensees on plant specific matters. For the review of generic matters, a working relationship was established with owners groups for plants designed by each nuclear steam supply vendor (Babcock & Wilcox Company (B&W), Westinghouse Electric Corporation (W), Combustion Engineering, Incorporated (C-E), and General Electric Company (GE)) and, in some cases with the individual nuclear steam supply system vendors themselves. At the outset, the highest priority was placed on plants of the B&W design; as short-term actions on these plants were completed, priority was shifted to those pressurized water reactor (PWR) plants designed by W and C-E, and then to boiling water reactor (BWR) plants, a significantly different light water reactor plant type. All BWRs incorporated nuclear steam supply systems (NSSS) designed by GE except for one plant, LaCrosse, whose NSSS was designed by Allis-Chalmers.

The B&OTF was composed of approximately thirty technical professionals in widely varying disciplines and areas of expertise. The Director of the B&OTF was Dr. D. F. Ross, Jr., and the Deputy Director was T. M. Novak. The members of the B&OTF are listed in Figure 1-1.

Figure 1-1

BULLETINS & ORDERS TASK FORCE ORGANIZATION

D. F. Ross, Jr., Director

T. M. Novak, Deputy Dirctor

Projects Group*

W. Kane, Group Leader

- I. Villalva, Alternate
- P. O'Reilly
- C. Thomas
- R. Capra
- J. Lee, Licensing Assistant

Systems Group

S. Israel, Group Leader

Section A

G. Mazetis
Section Leader

- F. Ashe
- W. LeFave
- G. Kelly
- M. Rubin
- K. Mahan
- D. Thatcher

Section B

P. Matthews
Section Leader

- W. Hodges
- J. Joyce
- C. Liang
- N. Wagner
- B. Siegel
- T. Greene
- B. Wilson

Analysis Group

Z. Rosztoczy, Group Leader

- P. Norian, Alternate
- R. Audette
- B. Sheron
- W. Jensen
- E. Throm
- J. Guttman
- R. Frahm

*C. J. Heltemes, Jr., served as group leader of the Projects Group until late September 1979.

The charter and scope of activities of the B&OTF are discussed in Sections 1.2.1 and 1.2.2 of this report, respectively. Section 1.2.3 of this report summarizes the B&OTF activities regarding: (1) the evaluation of licensees' responses to the IE bulletins, (2) the issuance and subsequent lifting of Commission Orders issued to licensees with B&W-designed operating reactors, and (3) the evaluation of the system reliability and predicted plant performance for the designs of each of the other reactor vendors with regard to feedwater transients and small-break LOCAs. These activities are discussed in greater detail in Sections 2.0 through 8.0 of this report.

The appendixes to this report are contained in a separate volume, NUREG-0645, Volume 2.

1.2.1 Charter

The charter of the B&OTF was to review the generic implications of the TMI-2 accident for all operating plants to confirm or establish the bases for their continued safe operation. The end products for this task included:

(1) Safety evaluations and authorizations to resume or continue operations.

(2) Licensing positions regarding the implementation of short-term measures on operating light water reactor plants.

(3) Recommendations for further improvements in the areas of design and operation, and administrative procedures.

(4) Notification of the Lessons Learned Task Force* of any required actions identified the during B&OTF review.

1.2.2 Scope of Activities

The scope of the B&OTF activities was limited to the review of loss of feedwater transients and small-break LOCAs. The specific areas of review related to these events are as follows:

(1) Reliability of systems.

(2) Analyses.

(3) Guidelines for the preparation of emergency operating procedures.

(4) Training of operators in emergency operating procedures.

In conducting its activities, the B&OTF had an overlapping responsibility with certain short-term Lessons Learned items as indicated below:

(1)	Item 2.1.3b (Part 1 only)	-	Instrumentation for Inadequate Core Cooling (Develop Procedures and Describe Existing Instrumentation)
(2)	Item 2.1.7a	-	Auto Initiation of Auxiliary Feed

*The Lessons Learned Task Force was also formed in NRR in response to the TMI-2 accident to identify and evaluate those safety concerns originating with the TMI-2 accident that required licensing actions (beyond those that had been specified in IE bulletins and Commission Orders) for currently operating reactors, as well as for pending operating license and construction permit applications. For this reason, the scope of the Lessons Learned Task Force was more general than the scope of the B&OTF.

(3) Item 2.1.7b - Auxiliary Feed Flow Indication

(4) Item 2.1.9 - Transient and Accident Analysis

These items are described in detail in NUREG-0578, "TMI-2 Lessons Learned Task Force Status and Short-Term Recommendations." The B&OTF reviewed the licensees' responses to these short-term Lessons Learned items since these requirements were identified in the generic reviews conducted by the B&OTF.

1.2.3 Summary of Activities

Bulletins

The NRC staff's preliminary review of the TMI-2 accident identified several errors and malfunctions that occurred during the accident and contributed significantly to its severity. As a result, all holders of operating licenses were subsequently instructed to take a number of immediate actions to avoid repetition of these errors. These instructions were specified in a series of bulletins issued by IE. Copies of these bulletins are provided in Appendix A to this report.

The initial bulletins defined actions to be taken by the licensees of operating plants using B&W-designed nuclear steam supply systems. As the NRC staff's evaluation continued, it was determined that additional actions were necessary, and these bulletins were subsequently expanded, clarified, and issued to the licensees of all operating plants for action. For example, holders of operating licenses for B&W-designed reactors were instructed by IE bulletins to take further actions, including immediate changes to decrease the reactor high pressure trip point and increase the pressurizer power-operated relief valve setpoints.

The B&OTF directed the evaluation of each licensee's response to the IE bulletins. This evaluation process involved an interoffice review group, which included representatives from IE and from the NRR Division of Operating Reactors. When it was concluded that a licensee understood the concerns expressed in the bulletins and provided acceptable responses to the bulletins, the bulletin review was completed and the evaluation issued as an NRC staff report.

The prompt actions taken by licensees in responding to the IE bulletins were considered to be an important contribution to assuring continued safe plant operation. In addition, the bulletins and related evaluations provided substantive input to other staff activities, such as those associated with the generic study efforts and the Lessons Learned Task Force. Thus, many of the subjects addressed by the bulletins were studied in greater depth through other staff activities and studies. Further, the bulletins and the associated responses were used as bases for inspecting plants and auditing reactor operator training.

Orders on B&W Plants

Soon after the TMI-2 accident, the NRC staff began a reevaluation of the design features of B&W-designed reactors to determine whether additional safety corrections or improvements were necessary. This evaluation involved numerous meetings with B&W and the affected licensees.

The conclusion of these preliminary studies was documented in an April 25, 1979, status report to the Commission. It was found that the B&W-designed reactors appeared to be unusually sensitive to certain transient conditions originating in the secondary system. The features of the B&W-designed plants that contributed to this sensitivity were: (1) the relatively small liquid volumes in the secondary side of the steam generators, (2) lack of direct initiation of reactor trip upon the occurrence of off-normal conditions in the feedwater system, (3) reliance on an integrated control system to automatically regulate feedwater flow, (4) actuation before reactor trip of a power-operated relief valve on the primary system pressurizer (which, if the valve sticks open, can aggravate the event), and (5) a low steam generator elevation (relative to the reactor vessel) which provides a smaller driving head for natural circulation (except for the Davis-Besse plant).

Because of these features, the B&W-designed plants rely more than other PWR designs on the reliability and performance characteristics of the auxiliary feedwater system, the integrated control system, and the emergency core cooling system (ECCS) performance for certain anticipated transients such as loss of offsite power and loss of normal feedwater. This, in turn, can require greater operator knowledge and skill to safely manage the plant controls during such anticipated transients. As a result of the work supporting the April 25, 1979, report, the NRC staff concluded that certain other short-term design and procedural changes at operating B&W-designed facilities were necessary in order to assure adequate protection to public health and safety.

After a series of discussions between the NRC staff and licensees of operating B&W-designed plants, the licensees agreed to shut down these plants until the actions identified to the Commission in the April 25, 1979, report could be completed. This agreement was confirmed by a Commission Order to each of the licensees. Authorizations to resume operation were issued during the period late May through early July 1979, as individual licensees satisfactorily completed the short-term actions and the NRC staff completed an onsite verification of the plant readiness to resume operation. In addition to the modifications to be implemented promptly, each licensee also proposed to carry out certain additional long-term modifications to further enhance the capability and reliability of the plant systems to cope with transient events.

Some of the long-term modifications involve the design, procurement, and qualification of safety-grade hardware. Therefore, all of the actions of the long-term portion of the Commission Orders have not yet been completed. NRC staff involvement will continue to assure that licensees complete each long-term action of the Commission Orders "as promptly as practicable," and that the Commission Orders are closed out by a prompt NRC staff acceptance review.

Generic and Plant-Specific Studies

For B&W-designed operating reactors, an initial NRC staff study was completed and published in NUREG-0560, "Staff Report on the Generic Assessment of Feedwater Transients in Pressurized Water Reactors Designed by the Babcock & Wilcox Company." This study considered the particular design features and operational history of B&W-designed operating plants in light of the TMI-2 accident and related current licensing requirements. As a result of this study, a number of findings and recommendations resulted which are now being pursued.

Generally, the activities involving the B&W-designed reactors are reflected in the actions specified in the Commission Orders. Consequently, as noted earlier, a number of actions have been specified regarding transient and small-break analyses, upgrading of auxiliary feedwater reliability and performance, procedures for operator action, and operator training. The results of the NRC staff review of the B&W small-break analysis is published in NUREG-0565, "Generic Evaluation of Small-Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox-Designed Operating Plants."

Similar studies have been completed for the W, C-E, and GE-designed operating plants. These studies, which also focus specifically on the predicted plant performance under different accident scenarios involving feedwater transients and small-break loss-of-coolant accidents, are published in NUREG-0611, "Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants"; NUREG-0635, "Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in Combustion Engineering-Designed Operating Plants"; and NUREG-0626, "Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in GE-Designed Operating Plants and Near-Term Operating License Applications."

1.3 Conclusions

Based on our review of the operating plants, we have reached the following conclusions:

(1) The continued operation of the operating plants is acceptable provided that certain actions related to the plants' designs and operation,

and training of operators identified in this report are implemented consistent with the recommended implementation schedules.

(2) The actions taken by the licensees with operating plants in response to the IE bulletins (including the actions specified in NUREG-0623, "Generic Assessment of Delayed Reactor Coolant Pump Trip During Small Break Loss-of-Coolant Accidents in Pressurized Water Reactors") provide added assurance for the protection of the health and safety of the public.

In addition, the B&OTF has independently confirmed the safety significance of those related actions recommended by other NRR task forces as discussed in this report.

2.0 BABCOCK & WILCOX COMPANY PLANTS

2.1 Introduction

This section describes the status of the B&W operating plants with respect to the actions taken within the purview of the B&OTF. These actions fall under three specific categories; IE bulletins, Commission Orders, and generic review. The requirements, actions completed, and actions pending are discussed for each of these categories.

2.2 List of Plants

The following operating plants with B&W-designed nuclear steam supply systems were reviewed by the B&OTF:

Plant Name	Utility	Power Level (Mwt)	Operating License Issuance Date
Oconee 1	Duke Power Company	2568	02/06/73
Oconee 2	Duke Power Company	2568	10/06/73
Arkansas Nuclear One 1 (ANO-1)	Arkansas Power & Light Company	2568	05/21/74
Oconee 3	Duke Power Company	2568	07/19/74
Rancho Seco	Sacramento Municipal Utility District	2772	08/16/74
Crystal River 3	Florida Power Corporation	2452	01/31/77
Davis-Besse 1	Toledo Edison Company	2772	04/22/77

2.3 Bulletins

2.3.1 Applicable Bulletins

The following IE bulletins, copies of which are provided in Appendix A to this report, are applicable to operating plants with B&W-designed reactors:

IE Bulletin 79-05 - This bulletin was issued on April 1, 1979. Seven specific areas were identified by the bulletin which have generic applicability at operating B&W facilities. Responses to this bulletin were requested within ten days of receipt. This bulletin was superseded by IE Bulletin 79-05A.

IE Bulletin 79-05A - This bulletin was issued on April 5, 1979, and superseded the actions required by licensees previously issued in IE Bulletin 79-05. This bulletin identified six potential human, design, and mechanical failures which resulted in core damage and radiation release at TMI-2. Those actions required by licensees which were substantially the same as IE Bulletin 79-05 required responses by April 11, 1979. The remaining responses were required by April 16, 1979.

IE Bulletin 79-05B - This bulletin was issued on April 21, 1979. Among the significant action items of this bulletin were a reduction in the high pressure reactor trip setpoint and an increase in the opening setpoint of the power-operated relief valve. Responses to this bulletin were required from licensees in stages ranging from 24 hours to 30 days.

IE Bulletin 79-05C - This bulletin was issued on July 26, 1979, and was combined with IE Bulletin 79-06C to make it applicable to all operating PWRs. For the B&W plants, it revised Item 4.c of IE Bulletin 79-05A and required that, following a reactor trip and initiation of high pressure injection caused by low reactor coolant system pressure, all operating reactor coolant pumps would be tripped immediately.

2.3.2 Actions Completed

IE Bulletins 79-05A and 79-05B

All responses have been received from licensees and reviewed by the NRC staff. Draft evaluations of the licensees' responses have been prepared and are presently undergoing final review. The first of the evaluations, the Rancho Seco evaluation, was issued on November 23, 1979.

IE Bulletin 79-05C

Short-term

Item 1: This item required: (1) that all operating reactor coolant pumps be tripped upon reactor trip and initiation of high pressure injection caused by low reactor coolant system pressure, and (2) that two licensed operators be in the control room at all times to accomplish this and other actions necessary in the event of a small-break LOCA. Each licensee has implemented this item.

Short-term

Item 2: This item required that licensees perform additional small-break LOCA analyses for a range of breaks and a range of time lapses between reactor trip and reactor coolant pump trip and to determine the peak cladding temperatures which result. B&W performed generic analyses which were reviewed and endorsed by all B&W licensees. We found these analyses acceptable and documented the results of our review in NUREG-0623, "Generic Assessment of Delayed Reactor Coolant Pump Trip during Small Break Loss-of-Coolant Accidents in Pressurized Water Reactors."

Short-term

Item 3: This item required that licensees develop new guidelines for operator action, for both LOCA and non-LOCA transients, that take into account the impact of reactor coolant pump trip requirements. These guidelines were to be based upon the analyses performed under Item 2 of the bulletin. Three sets of guidelines have been prepared by B&W to cover this requirement. One set of guidelines covers Oconee 1, 2, and 3; Rancho Seco; Crystal River 3; and Three Mile Island 1. The second set of guidelines covers Arkansas Nuclear One 1 and the third set of guidelines covers Davis-Besse 1. These guidelines modify the previously approved guidelines which were submitted to the NRC staff by B&W on May 16, 1979. The revised guidelines are in the final stages of NRC staff review at the present time.

Short-term

Item 4: This item required that emergency procedures, based on the guidelines developed under Item 3 above, be developed and that all licensed reactor operators and senior reactor operators be retrained as required. By September 15, 1979, all B&W operating plants had completed this requirement.

Short-term

Item 5: This item relates to inadequate core cooling (as discussed in NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," Section 2.1.9) regarding analyses and development of guidelines and procedures. B&W performed analyses and once again revised the B&W small-break LOCA operator guidelines to incorporate the subject of inadequate core cooling. These guidelines have been reviewed by the licensees and submitted to the NRC staff for review. The procedures associated with this item were required to be implemented by January 31, 1980.

Long-term

Item 1: This item pertains to the design of a system which will assure automatic tripping of operating reactor coolant pumps under all circumstances in which this action may be required. Preliminary designs were submitted by the licensees which incorporate the use of coincident signals from reactor coolant pump power/current and a low pressure engineered safety features actuation system signal. Preliminary design approval was given to the licensees by NRC staff letters dated December 17 and 18, 1979 to proceed with the final design.

2.3.3 Status of Actions Pending

IE Bulletins 79-05A and 79-05B

As discussed above, draft evaluations have been prepared for all B&W operating plants. The NRC staff evaluation for Rancho Seco was issued to the licensee on November 23, 1979. It is expected that the NRC staff evaluations for the remaining plants will be issued in early 1980.

IE Bulletin 79-05C

Short-term

Item 3: NRC staff approval of the revised B&W small-break LOCA guidelines, incorporating the reactor coolant pump trip requirements of this bulletin, are expected to be completed by February 29, 1980.

Short-term

Item 5: NRC staff approval of the revised B&W small-break LOCA guidelines, incorporating the subject of inadequate core cooling is not required prior to implementation by the licensees. The NRC staff plans to review these guidelines and supporting analyses; however, at the present time, no review schedule has been established.

Long-term

Item 1: As discussed above, NRC staff letters dated December 17 and 18, 1979, were issued to the B&W licensees to acknowledge our preliminary design approval of their proposed design for the automatic reactor coolant pump trip system. By February 15, 1980, each licensee is to submit a schedule for supplying final design information and a schedule for installation of the system.

2.4 Orders

2.4.1 Bases for Orders

On April 2, 1979, while post-accident recovery operations were taking place at TMI-2, a task group was appointed under the direction of R. Tedesco to perform a generic assessment of feedwater transients in B&W-designed plants in light of operating experiences including the TMI-2 accident. The purpose of this study was to determine the bases for continued safe operation of these plants in both the short term and long term. The complete findings of this task group were later published in NUREG-0560. Based on the preliminary findings of this task group, a document entitled "NRR Status Report on Feedwater Transients in B&W Plants," was issued on April 25, 1979. This document (a copy of which is provided as Appendix B to this report) highlighted the sensitivity of the B&W plants to feedwater transients and the role that this sensitivity might play as a precursor or contributor to a TMI-2 type accident.

NUREG-0560 identified several design differences that distinguished the B&W plants in their responses to feedwater transients. The features of the B&W design which contribute to this sensitivity are: (1) the design of the steam generators to operate with a relatively small liquid volume in the secondary side which made changes in feedwater flow manifest itself quickly as changes in heat transfer from the primary system; (2) the lack of direct initiation of a reactor trip upon upsets in the secondary system such as loss of feedwater and turbine trip; (3) reliance on the complex integrated control system to automatically regulate feedwater flow; (4) actuation of the power-operated relief valve on certain anticipated transients before a reactor trip; and (5) a low steam generator elevation (relative to the reactor vessel for the lowered loop plants) which provided a smaller driving head for natural circulation (except for the Davis-Besse plant).

Because of these features, the B&W-designed reactors placed more reliance on the reliability and performance characteristics of the auxiliary feedwater system, integrated control system and high pressure injection system to mitigate the consequences of transients such as loss of feedwater and small break loss-of-coolant accidents than other PWR designs.

NUREG-0560 concluded that at that time the NRC staff did not have reasonable assurance that the B&W-designed plants could continue to operate without undue risk to the health and safety of the public and that the plants should be shut down until certain items identified in the report were completed to the satisfaction of the NRC staff.

The Commission was briefed on the contents of NUREG-0560 on April 25, 1979. On April 26 and 27, 1979, the NRC staff held meetings with representatives of the B&W licensees. As a result of these meetings, the licensees agreed to shut down their facilities (or to keep them shutdown if they were already shutdown) until certain short-term actions were completed.

The bulk of short-term design and procedural changes committed to by the licensees were those identified as Items (a) through (e) on Pages 1-7 of the "NRR Status Report on Feedwater Transients in B&W Plants."

On the basis of these commitments, the NRC staff again met with the Commission on April 27, 1979. The purpose of that meeting was to clarify the licensees' commitments to the Commission. It was during this meeting that the Commission directed the NRC staff to prepare Confirmatory Orders to formalize the agreements reached with the licensees.

These Orders were prepared by the NRR Division of Operating Reactors in conjunction with the Office of the Executive Legal Director and the Office of the General Counsel and were issued to the licensees of each of the B&W operating plants between May 7 and 17, 1979. Copies of the orders are provided in Appendix C to this report.

2.4.2 Applicable Orders

Copies of the Commission Orders and the letters lifting the Orders are provided in Appendixes C and D to this report, respectively.

Facility	Commitment Letters	Order Issued	Order Lifted
Oconee 1, 2 and 3	04/25/79 04/26/79 05/04/79	05/07/79	05/18/79
Rancho Seco	4/27/79	05/07/79	06/27/79
Crystal River 3	05/01/79	05/16/79	07/06/79
Davis-Besse 1	04/27/79 05/04/79	05/16/79	07/06/79
Arkansas Nuclear One 1	05/11/79	05/17/79	05/31/79

2.4.3 Short-Term Actions Required by Orders

The short-term actions required by the Commission Orders of May 1979 are listed in Table 2-1 of this report.

2.4.4 Long-Term Actions Required by Orders

The long-term actions required by the Commission Orders of May 1979 are listed in Table 2-2 of this report.

2.4.5 Actions Completed

Short-Term Actions

All specific short-term requirements specified in the Commission Orders of May 1979 were completed by the licensees, evaluated by the NRC staff and found acceptable prior to authorizing restart of the facilities.

Following completion of the actions by the licensees, a restart team was sent to each of the facilities. The restart teams consisted of individuals from IE and NRR. The purpose of the site visits were to verify completion of the short-term requirements of the Commission

TABLE 2-1

SHORT-TERM ACTIONS REQUIRED BY COMMISSION ORDERS OF MAY 1979 - B&W PLANTS

Action	Oconee, 1-3	ANO-1	Rancho Seco	Crystal River-3	Davis-Besse-1
A. UPGRADE TIMELINESS AND RELIABILITY OF AFW SYSTEM					
1. Provide for automatic start of all three pumps upon a signal from any unit. Cross connect the discharge headers	X				
2. Provide for starting of motor-driven AFW pumps from a vital bus		X	X	X	
3. Station an operator at local valves during testing in communications with control room		X	X	X	
4. Develop procedures for control of AFW independent of ICS		X	X	X	X
5. Verify AFW pumps operable		X		X	
6. Provide for obtaining alternate sources of water for AFW		X	X	X	
7. Provide for automatic start of motor-driven pumps		X		X	
8. Provide for timely operator notification of AFW automatic initiation		X	X	X	
9. Provide for timely operator verification of AFW flow to steam generators upon automatic initiation		X	X		
10. Verify that Technical Specifications requirements for AFW are in accordance with the accident analysis		X	X		
11. Provide AFW flow rate indication in control room		X	X		
12. Verify failure position of AFW flow control valves			X	X	X
13. Remove interlock which prevents turbine-drive pump from injecting when motor-driven pump is operating				X	

TABLE 2-1 (Continued)

	Action	Oconee, 1-3	ANO-1	Rancho Seco	Crystal River-3	Davis-Besse-1
	14. Install Dynamic breaking on turbine-driven pump speed change motor					X
B.	IMPLEMENT OPERATING PROCEDURES AND TRAINING FOR INITIATING AND CONTROLLING AFW INDEPENDENT OF ICS	X	X	X	X	X
C.	INSTALL A HARD WIRED, CONTROL-GRADE REACTOR TRIP UPON LOSS OF FEEDWATER AND TURBINE TRIP	X	X	X	X	X
D.	COMPLETE SMALL BREAK LOCA ANALYSIS, PROCEDURES AND OPERATOR TRAINING	X	X	X	X	X
E.	COMPLETE TMI-2 SIMULATOR TRAINING FOR ALL OPERATORS	X	X	X	X	X
F.	REEVALUATE ANALYSIS FOR DUAL LEVEL SETPOINT CONTROL					X
G.	REEVALUATE SEPTEMBER 24, 1977 TRANSIENT IN LIGHT OF TMI-2					X

TABLE 2-2

LONG-TERM ACTIONS REQUIRED BY ORDERS OF MAY 1979 - B&W PLANTS

	Action	Oconee, 1-3	ANO-1	Rancho Seco	Crystal River-3	Davis-Besse-1
A.	CONTINUED UPGRADE OF AFW SYSTEM					
	1. Install two motor-driven pumps per unit	X				
	2. Connect motor-driven AFW pump to vital bus		X			
	3. Install AFW Control System (being developed by B&W)		X			
	4. Modify suction piping to improve separation		X			
	5. Provide control room annunciation for all auto start conditions of AFW		X			
	6. Add redundant pressure switch to the AFW pump suction and a redundant low pressure annunciation in the control room		X			
	7. Identify and implement any design changes which relate to the short-term items previously completed which would improve safety			X		
	8. Provide AFW flow verification in the control room				X	
	9. Continue to review performance of AFW system to assure reliability and performance					X
B.	SUBMIT A FAILURE MODE AND EFFECTS ANALYSIS OF THE ICS	X	X	X	X	X
C.	UPGRADE THE REACTOR TRIP ON LOSS OF FEEDWATER AND TURBINE TRIP TO SAFETY GRADE	X	X	X	X	X
D.	CONTINUE OPERATOR TRAINING AND DRILLING TO ASSURE A HIGH STATE OF PREPAREDNESS	X	X	X	X	X
E.	CONTINUED ATTENTION TO TRANSIENT ANALYSIS AND PROCEDURES FOR MANAGEMENT OF SMALL BREAKS					X

Orders and conduct audits of the licensed reactor operators and senior reactor operators.

During the course of our review, we identified four items which would require additional work by the licensees. These items were identified to the licensees in an NRC staff letter to all B&W operating plant licensees except the Three Mile Island 1 and 2 licensee, dated August 21, 1979. These items were as follows:

(1) Perform a detailed analysis of the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater.

This issue was identified in the NRC staff evaluations for Rancho Seco, Davis-Besse 1, and Crystal River 3; however, it is also applicable to Oconee 1, 2 and 3 and Arkansas Nuclear One 1. A request for this analysis was forwarded to B&W in an NRC staff letter dated July 12, 1979. The analysis is scheduled to be completed by B&W on February 7, 1980, and submitted to the NRC staff following a review by the licensees of B&W operating plants.

(2) Power-operated relief valve and safety valve lift frequency and mechanical reliability.

In an NRC staff letter to all B&W operating plants dated September 28, 1979, we informed the licensees that: (1) NUREG-0578, Section 2.1.2 addressed the NRC staff's concerns with regard to mechanical reliability of power-operated relief valves and safety valves. By completing the requirements identified in NUREG-0578, no further action would be taken on this matter under the Commission Orders; and (2) with regard to lift frequency of power-operated relief valves and safety valves, the NRC staff requested a report which would address the probability of challenging the power-operated relief valves and safety valves as a result of the revised setpoints of the high pressure reactor trip and power-operated relief valve actuation setpoint. This report was prepared by B&W and forwarded to the licensees for review. The report was also sent to the NRC staff for review. Our review of this document has led us to conclude that the inversion of the setpoints for the high pressure reactor trip and power-operated relief valve opening has significantly reduced the number of challenges and only slightly increased the reactor trip frequency. This topic is discussed further in NUREG-0565, "Generic Evaluation of Small-Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox-Designed Operating Plants." As specified in NUREG-0565, additional information will be required to be submitted by the licensees of B&W operating plants to close out this item.

(3) Additional small-break LOCA analyses.

In addition to the small-break LOCA analyses submitted to the NRC staff, we required the licensees to perform additional work in this area. The specific information requested is delineated in Attachment A to Enclosure 1 of our letter to all B&W operating plant licensees except the Three Mile Island 1 and 2 licensee, dated August 21, 1979. The majority of these analyses have been completed and submitted to the NRC staff for review. Those items which have not been completed to date are listed in Section 2.4.6 of this report. The information which has been received by the NRC staff has been reviewed. The results of that review are documented in NUREG-0565.

(4) Analysis of loss of feedwater and other anticipated transients.

This item is discussed in Section 8.4.1 of NUREG-0560 and endorsed in the NRC staff's safety evaluation for each plant. Some of this work has been completed; however, the scope and schedule of this requirement has been superseded by Section 2.1.9 of NUREG-0578. The NRC staff has met with the B&W Owners Group on several occasions to discuss the program outlined by the Owners Group to comply with this portion of NUREG-0578. When completed, this program will provide the licensees with much improved "symptom"-oriented generic guidelines for a series of anticipated and abnormal operational transients. The individual licensees will then develop detailed emergency procedures based on the B&W generic guidelines. As noted above, this requirement has been superseded by the requirements of NUREG-0578.

Following completion of the NRC staff's review of the licensees' actions to comply with the short-term portion of the Commission Orders, plant specific safety evaluations were prepared and sent to the licensees.

The letters to the licensees which authorized resumption of power operations and forwarded the NRC staff's safety evaluations were as follows:

Facility	Date of Letter
Oconee 1, 2, and 3:	05/18/79
ANO-1:	05/31/79
Rancho Seco:	06/27/79
Crystal River 3:	07/06/79
Davis-Besse 1:	07/06/79

A copy of each of these letters is provided in Appendix D to this report.

Long-Term Actions

The majority of long-term actions lended themselves to generic resolution. As a result, the B&W Owners Group formed a special subcommittee called the TMI-2 Effects Subcommittee to deal with matters related to the Commission Orders.

The following is a status of each of the long-term requirements:

(1) Continued upgrade of auxiliary feedwater system.

The long-term portion of the Commission Orders require that licensees continue to identify areas or components within the auxiliary feedwater system which will further enhance system performance and reliability. Some licensees have committed to making improvements in the system which are specifically listed in the Commission Orders. Other licensees have provided a general statement to the effect that they will identify and implement any design changes which will improve safety.

The NRC staff, in conjunction with the B&W Owners Group, has agreed on one unified approach to identify areas or components within the auxiliary feedwater system which are in need of further improvements. The licensees have contracted with B&W to perform a reliability analysis of the auxiliary feedwater system on each of their facilities. This analysis uses simplified event tree and fault tree techniques to evaluate the reliability of the auxiliary feedwater systems for three specific cases: (1) loss of feedwater, (2) loss of feedwater combined with a loss of offsite power, and (3) a loss of feedwater combined with loss of all alternating current power. The reliability analysis will identify any dominant failure contributors. Draft reliability studies have been completed by B&W and have been reviewed by the licensees. These reports have been revised by B&W and sent to the licensees for final review prior to submitting them to the NRC staff. The final reports for each facility have been received and are being reviewed.

In addition to forwarding the final auxiliary feedwater system reliability study to the NRC staff, each licensee included in its submittal a generic report which compares the reliability of all B&W operating plants' auxiliary feedwater systems. The licensees were to forward with these reports documentation of all planned upgrades for the auxiliary feedwater systems. Areas that were to be considered in identifying design changes included: (1) any specific commitments made as a result of the Commission Orders; (2) upgrades to comply with Sections 2.1.7a and b of NUREG-0578; (3) any upgrades as a result of the identification of dominant failure contributors in the auxiliary feedwater reliability study; and (4) applicable NRC staff recommendations identified in the B&OTF generic reports on W and C-E operating plants (NUREG-0611 and NUREG-0635, respectively).

(2) Submit a failure mode and effects analysis of the integrated control system.

On August 17, 1979, B&W submitted a report entitled, "Integrated Control System Reliability Analysis," BAW-1564. Upon the completion of their review, each licensee submitted a letter to the NRC staff endorsing the report as applicable to its plant. The NRC staff contracted with the Instrumentation and Controls Division of the Oak Ridge National Laboratory (ORNL) to review the report. The purpose of the review by ORNL is to determine the adequacy of the B&W evaluation as documented in BAW-1564.

The preliminary review of the report by ORNL indicated that the recommendations made by B&W in the report were reasonable. On November 7, 1979, in an NRC staff letter to all B&W operating plant licensees except the Three Mile Island 1 licensee, we requested that licensees address these recommendations and discuss any followup actions which were underway or proposed. Responses to this letter are expected in January 1980.

On December 4, 1979, we received a draft copy of an ORNL report entitled, "Review of Babcock & Wilcox Report - Integrated Control System Reliability Analysis - BAW-1564, August 1979." In general, the report concludes that only a small number of integrated control system malfunctions resulted in reactor trips (six out of 162) and that this data, supported by conversations with plant operators, demonstrates that the integrated control system is failure tolerant to a significant degree. This feature is also evidenced by noting the large number of postulated failures in the failure modes and effects analysis that could result in reactor trips compared with the low trip rate experience in practice. The ORNL report tends to agree with the B&W conclusion that the integrated control system prevents or mitigates many more upsets than it creates and is generally superior to manual or fragmented control schemes. The ORNL report also points out areas of weakness which need further evaluation and offers areas which should be considered for further study.

It is expected that the final report from ORNL will be received in January 1980, at which time the NRC staff will review its recommendations, develop positions, and inform the licensees what additional work, if any, will need to be done to improve the integrated control system or its interfacing systems.

(3) Upgrade the reactor trip on loss of feedwater and turbine trip to safety grade.

The NRC staff has reviewed preliminary design information submitted by the licensees to upgrade the anticipatory reactor trip for loss of feedwater and turbine trip. In addition, the NRC staff has reviewed the implementation schedule proposed by the licensees. Safety evaluations were prepared by the NRC staff and forwarded to each of the licensees by letters on December 20, 1979. These letters authorize each licensee to proceed with detailed design and procurement of components such that the systems could be operational within approximately six months. These letters also identified the information which would have to be submitted to the NRC staff for review prior to final approval of the design.

(4) Continue operator training and drilling to assure a high state of preparedness.

In a letter to all B&W operating plant licensees, except the Three Mile Island 1 and 2 licensee, we requested that all licensees document the actions they have taken to comply with the long-term training requirements of the Commission Orders.

Responses to this request have been received from the licensees and reviewed by the NRC staff. Based on our review, we have concluded that the training programs have been sufficiently improved such that the operators will be trained and drilled to assure a high state of preparedness for both LOCA and non-LOCA transients.

The NRC staff critiera used to determine compliance with the long-term training requirements of the Commission Orders for the requalification training program and the hot license training program at each B&W facility are presented below:

(a) The following lecture subjects are to be included or expanded, as applicable, in each of the programs:

(1) Thermodynamics,

(2) Hydraulics,

(3) Fluid flow,

(4) Heat transfer,

(5) Small-break LOCA phenomena,

(6) Inadequate core cooling, and

(7) Transient training, including loss of feedwater.

In addition, the TMI-2 sequence of events is to be included in at least the first year's requalification program and all hot license programs.

(b) All programs are to include simulator training in which the operators are to be provided with hands-on experience in handling small breaks and other transients that could lead to loss of heat removal, inadequate core cooling, and natural circulation. It should be noted that, at the present time, the simulator at Lynchburg, Virginia (which represents the Rancho Seco facility), is the only B&W-type simulator; however, a simulator representing Washington Public Power Supply System Units 1 and 4 will be completed shortly. In addition, Duke Power Company has contracted for a simulator representing Oconee.

The Commission has approved the recommendations of SECY-330E, "Qualifications of Reactor Operators." Some of these recommendations that will also assure an increased level of operator training and awareness at B&W facilities as well as other plants are:

(a) Increase the passing grade of written licensing and requalification examinations to 80 percent overall and 70 percent minimum on every category;

(b) Require and expand the use of simulators;

(c) All requalification examinations are to be administered by the NRC; and

(d) Add additional categories to operator and senior operator written examinations, including subjects 1 through 4 of the lecture topics identified above.

(5) Continued attention to transient analysis and procedures for management of small breaks.

This specific requirement is identified only in the Commission Order for Davis-Besse 1 issued May 16, 1979. This requirement has the same intent as Item 2.1.9 of NUREG-0578. Toledo Edison Company has been working in conjunction with the B&W Owners Group to complete the three specific areas covered by this section (i.e., small-break LOCA analysis and procedures, inadequate core cooling analysis and procedures, and anticipated and abnormal transient and accident analysis and procedures).

2.4.6 Status of Actions Pending

Short-Term Actions

All short-term actions required by the Commission Orders of May 1979 have been completed; however, certain items, as discussed in Section 2.4.5 of this report, developed during the course of the NRC staff review of the licensees' compliance with the short-term requirements of the Commission Orders, were identified by the NRC staff as areas requiring additional work. The following is the status of those additional areas which have not yet been completed:

(1) Perform a detailed analysis of the thermal-mechanical conditions in the reactor vessel during recovery from small breaks with an extended loss of all feedwater.

The request for this generic study was made in an NRC staff letter to B&W dated July 12, 1979. The analysis is expected to be completed by B&W on February 7, 1980, and will be sent to the NRC following a review by the B&W licensees.

(2) Additional small-break LOCA analysis.

The information requested in this area was made in NRC staff letters to all B&W operating plant licensees except the Three Mile Island 1 and 2 licensee dated August 21, 1979 and November 21, 1979. The outstanding information is as follows:

(a) Provide a benchmark analysis of sequential auxiliary feedwater flow to the steam generators following a loss of main feedwater. This analysis was provided in a B&W letter dated June 15, 1979. However, in this analysis the TRAP-2 code with a six-node steam generator model was utilized. All small-break LOCA analyses presented to the NRC staff by B&W have been performed using the CRAFT-2 code with a three-node steam generator model. We require a benchmark analysis for sequential auxiliary feedwater flow also be performed using CRAFT-2 code with a three-node representation of the steam generator.

This analysis is expected to be completed by B&W in early 1980, and will be sent to the NRC staff following a review by the B&W licensees.

(b) Provide the reactor system response to a stuck-open power-operated relief valve for the case of a small-break LOCA which causes the reactor system to pressurize to the power-operated relief valve setpoint. A qualitative assessment of this transient was completed by B&W and forwarded to the NRC staff by the licensees in November 1979. A detailed quantitative assessment from the licensees is scheduled to be submitted to the NRC on February 1, 1980.

(c) Pretest calculations of the Loss of Fluid Test (LOFT) small-break test L3-1. Pretest calculations were not completed by B&W by the time test L3-1 was run. However, B&W submitted its final predictions on December 13, 1979. This analysis is presently being reviewed. The LOFT L3-1 test program is discussed further in Section 7.0 of this report.

(d) By use of analysis and/or experiment, address the mechanical effects of slug flow on steam generator tubes. The schedule for submittal of this analysis by the licensees is March 1, 1980.

(e) Evaluation of the impact of reactor coolant pump seal damage and leakage due to loss of seal cooling on loss of offsite power. The schedule for submittal of this evaluation by the B&W licensees is March 1, 1980.

Long-Term Actions

A significant amount of work has been completed on all of the long-term actions required by the Commission Orders of May 1979. However, as discussed in Section 2.4.5 of this report, certain items still require additional followup work. The following is the status of those items which have not yet been completed:

(1) Continued upgrade of the auxiliary feedwater system.

Prior to December 31, 1979, all licensees had submitted a final auxiliary feedwater reliability study. The report addressed areas of the auxiliary feedwater system which are in need of design improvement to increase the timeliness and reliability of the system. The planned upgrades will come from a combination of all of the following:

(a) Specific commitments made as a result of the Commission Orders of May 1979;

(b) System upgrades to comply with Sections 2.1.7a and b of NUREG-0578;

(c) System upgrades as a result of the identification of dominant failure contributors in the auxiliary feedwater system reliability study; and

(d) System upgrades necessary to comply with applicable NRC staff recommendations identified in the B&OTF generic reports on W and C-E operating plants (NUREG-0611 and NUREG-0635, respectively).

NRC staff review of the auxiliary feedwater system reliability reports, the proposed upgrades, and the implementation schedule is presently underway. Item 2.1.7b of NUREG-0578 does not require NRC staff review prior to implementation.

(2) Failure modes and effects analysis of the integrated control system.

By late January 1980, the NRC staff should have the final report from ORNL evaluating B&W's "Integrated Control System Reliability Analysis," BAW-1564. In addition, the licensees' responses to our request for additional information to all B&W operating plants, dated November 7, 1979, should be received. NRC staff review of the ORNL report and the licensees' responses will be required to determine what further modifications and or upgrades, if any, will be required for the integrated control system.

(3) Upgrade of the anticipatory reactor trip for loss of feedwater and turbine trip to safety-grade.

In letters dated December 20, 1979, the licensees were given preliminary design approval to proceed with final design and procurement of equipment necessary to upgrade the present control-grade anticipatory reactor trip to safety-grade. These letters also informed licensees what additional information would be required to be submitted for final NRC staff review prior to installation and operation of the system.

2.5 Generic Review

2.5.1 Scope

The B&W generic review has been accomplished in two steps. The first post-TMI-2 accident evaluation of the B&W-designed operating plants was performed in April and May of 1979. That effort dealt with providing an early assessment of the generic aspects of the feedwater transient and related ensuing events at TMI-2 to determine bases for continued safe operation of other operating B&W-designed plants. The study culminated in the publication of NUREG-0560 in May 1979. The conclusions and recommendations made in NUREG-0560 served as partial bases for issuance of the Commission Orders of May 1979. In addition, further work by the B&OTF and the TMI-2 Lessons Learned Task Force expanded upon and implemented many of the recommendations contained in NUREG-0560.

The second generic report, NUREG-0565, describes the NRC staffs' review of generic analyses performed by B&W on combinations of potential small-break LOCAs and loss of feedwater events.

2.5.2 Interfaces with Short-Term Lessons Learned Items

The following items are included in NUREG-0578 as part of the short-term recommendations, but are also within the purview of, and are being managed by, the B&OTF:

Generic Reports	NUREG-0578 Items
NUREG-0560 NUREG-0565	Section 2.1.3.b: "Instrumentation for Detection of Inadequate Core Cooling in PWRs and BWRs (analysis and procedures involved with this item only). The requirements of this item also interface with those in Recommendation 2.1.9.
NUREG-0560	Section 2.1.7a: "Automatic Initiation of the Auxiliary Feedwater System for PWRs."
NUREG-0560	Section 2.1.7b: "AFW Flow Indication to Steam Generators for PWRs."
NUREG-0560 NUREG-0565	Section 2.1.9: "Analysis of Design and Off-Normal Transients and Accidents."

2.5.3 Actions Completed

NUREG-0560: As discussed in Section 2.5.1 of this report, most of the recommendations listed in NUREG-0560 have been implemented or are scheduled for implementation through actions taken under the short-term and long-term requirements of the Commission Orders of May 1979 or have been broadened in scope in NUREG-0578 and NUREG-0585, "TMI-2 Lessons Learned Task Force Final Report."

NUREG-0565: The work completed in NUREG-0565 can be separated into four areas: (1) staff evaluation of B&W analysis of small-break LOCAs brought about by a failure of a relief or safety valve to close during an anticipated transient; (2) NRC staff evaluation of B&W's responses to the concerns of C. Michelson of the Tennessee Valley Authority regarding small-break LOCAs; (3) NRC staff evaluation of the analyses and methods used to predict the plant response to small-break LOCAs; and (4) NRC staff audit calculations of B&W calculations.

2.5.4 Status of Actions Pending

Based on our review of the B&W small-break LOCA analyses and other related topics as documented in NUREG-0565, the NRC staff has recommended that the licensees make certain system modifications and conduct specific additional analyses. The principal conclusions and recommendations of the report are contained in Section 2.0 of NUREG-0565. In addition, Table 2-1 of the NUREG-0565 provides the recommended implementation schedule for the B&W licensees to follow is complying with the recommendations of the report. These recommendations and implementation schedule are included as Table 2-3 of this report.

TABLE 2-3

IMPLEMENTATION OF B&OTF RECOMMENDATIONS FOR B&W-DESIGNED OPERATING PLANTS

Recommendation Number	Required Action	Schedule Date
EXPECTED FREQUENCY OF SMALL-BREAK LOCAs		
2.1.2a	Automatic block valve closure system - Installed and Operational	07/01/80
	Testing of automatic block valve closure system	Note 1
2.1.2b	Evaluation of PORV opening probability during overpressure transients	05/01/80
2.1.2c	Reporting of failures and challenges to the PORV	Note 2
2.1.2d	Evaluation of safety valve reliability	06/01/80
2.1.2e	Reporting of failures and challenges to safety valves	Note 2
SMALL-BREAK LOCA ANALYTICAL MODEL		
2.2.2a	Analysis methods for SBLOCA (including evaluation of noding)	07/01/80
2.2.2b	Plant-specific calculations to show compliance with 10 CFR 50.46	01/01/81
2.2.2c	Evaluation of effects of core flood tank injection on SBLOCAs	07/01/80
SMALL-BREAK LOCA BEHAVIOR		
2.3.2a	Automatic trip for RCPs during SBLOCA - Installed and Operational (also see Section 7.3.1 of NUREG-0623)	01/01/81
2.3.2b	Review and upgrade reliability and redundancy of nonsafety-grade equipment upon which SBLOCA mitigation relies	TMI-2 ACTION PLAN
2.3.2c	Minimum simulator training requirements for SBLOCAs	01/01/81
STAFF AUDIT CALCULATIONS		
2.4.2a	Additional staff audit calculations of B&W's SBLOCA analyses	NRC ACTION

TABLE 2-3 (Continued)

Recommendation Number	Required Action	Schedule Date
TWO-PHASE NATURAL CIRCULATION AND ACCOMMODATION OF LOSS OF ALL FEEDWATER		
2.5.2a	Consideration of diverse decay heat removal path for Davis-Besse Unit 1	TMI-2 Action Plan
TENNESSEE VALLEY AUTHORITY (TVA) SMALL-BREAK CONCERNS		
2.6.2a	Experimental verification of two-phase natural circulation	01/01/81
2.6.2b	Instrumentation to verify natural circulation - Installed and Operational	04/01/81
2.6.2c	Analysis of plant response to a SB which is isolated, causing RCS repressurization and subsequent stuck-open PORV	06/01/80
2.6.2d	Analysis of plant response to a SB in the pressurizer spray line with a stuck-open spray line isolation valve	05/01/80
2.6.2e	Evaluation of effects of water slugs in piping caused by HPI and CFT flows	05/01/80
2.6.2f	Evaluation of RCP seal damage and leakage during a SBLOCA	03/01/80
2.6.2g	Submit predictions for LOFT Test L3-6 (RCPs running) (Schedule for performing test not finalized)	Pretest
2.6.2h	Submit requested information on the effects of noncondensible gases: (1) justification for omission radiolytic decomposition as a source on noncondensible gases and (2) verification of predicted condensation heat transfer degradation.	05/01/80
*2.6.2i	Evaluation of mechanical effects of slug-flow on steam generator tubes	03/01/80

NOTES:

1. Confirmatory test of automatic block valve closure system shall be performed during the first refueling outage after installation of the system.

2. PORV and safety valve failures and challenges:
   Failures shall be reported to the NRC promptly by licensees.
   Challenges shall be reported in licensees' annual reports.

3. TMI-2 Action Plan refers to NUREG-0660 entitled, "Action Plans for Implementing Recommendations of the President's Commission and Other Studies of the TMI-2 Accident," draft dated 12/10/79.

* Information requested in recommendations 2.6.2f and 2.6.2i were previously requested from the B&W licensees by NRC staff letter dated 11/21/79.

3.0 WESTINGHOUSE ELECTRIC CORPORATION PLANTS

3.1 Introduction

This section describes the status of the W operating plants with respect to the actions taken within the purview of the B&OTF. These actions fall under three specific categories; IE bulletins, Commission Orders, and generic review. The requirements, actions completed, and actions pending are discussed for each of these categories.

3.2 List of Plants

The operating plants with W-designed nuclear steam supply systems which were reviewed by the B&OTF are identified in Table 3-1.

3.3 Bulletins

3.3.1 Applicable Bulletins

The following IE bulletins, copies of which are provided in Appendix A to this report, are applicable to operating plants with W-designed nuclear steam supply systems:

IE Bulletin 79-05 - This bulletin was issued on April 1, 1979. Although this bulletin was issued for action by B&W-designed plants, it was also sent to operating plants designed by W for information purposes, and later incorporated by reference in IE Bulletin 79-06, a bulletin applicable to W-designed plants. This bulletin was superseded by IE Bulletin 79-05A.

IE Bulletin 79-05A - This bulletin was issued on April 5, 1979, at which time it superseded IE Bulletin 79-05, and was sent to operating plants

TABLE 3-1

OPERATING PLANTS WITH WESTINGHOUSE-DESIGNED NUCLEAR STEAM SUPPLY SYSTEMS

Plant	Utility	Power Level (thermal megawatts)	Date OL Issued
Haddam Neck	Connecticut Yankee Atomic Power Co.	1825	06/30/67
Yankee Rowe	Yankee Atomic Electric Co.	600	07/09/60
San Onofre 1	Southern California Edison	1347	03/27/67
Prairie Island 1 & 2	Northern States Power Co.	1650/Unit	08/09/73& 10/29/74
Farley 1	Alabama Power Co.	2652	06/25/77
Salem 1	Public Service Electric & Gas	3338	08/13/76
North Anna 1	Virginia Electric & Power Co.	2775	11/26/76
D.C. Cook 1 & 2	Indiana & Michigan Electric Co.	3250, 3391	10/25/74& 12/23/77
Indian Point 3	Power Authority of the State of New York	2760	12/12/75
Ginna	Rochester Gas & Electric Co.	1520	09/19/69
Kewaunee	Wisconsin Public Service Corp.	1650	12/21/73
Zion 1 & 2	Commonwealth Edison Co.	3250/Unit	04/06/73& 11/14/73
Point Beach 1 & 2	Wisconsin Electric Power Co.	1518/Unit	10/05/70& 11/10/71
Turkey Point 3 & 4	Florida Power & Light Co.	2200/Unit	07/19/72& 04/10/73
Indian Point 2	Consolidated Edison Co.	3025	10/19/71
Trojan	Portland General Electric	3411	11/21/75
H.B. Robinson 2	Carolina Power & Light Co.	2200	07/31/70
Surry 1 & 2	Virginia Electric & Power Co.	2441/Unit	05/25/72& 01/29/73
Beaver Valley 1	Duquesne Light Co.	2652	01/30/76

designed by W for information purposes only. This bulletin was later made applicable to operating nuclear power plants with W-designed reactors by reference in IE Bulletins 79-06 and 79-06A, the bulletins applicable to W-designed plants.

IE Bulletin 79-06 - This bulletin was issued on April 11, 1979, and, except for B&W-designed facilities, was applicable to all operating PWR plants. This bulletin was superseded by IE Bulletin 79-06A for operating PWR facilities with W-designed reactors.

IE Bulletin 79-06A - This bulletin was issued on April 14, 1979, at which time it superseded IE Bulletin 79-06 for operating PWR facilities with W-designed reactors. The requirements contained in this bulletin represent the greatest number of actions that must be taken by licensees with W-designed reactors.

IE Bulletin 79-06A, Revision 1 - This bulletin was issued on April 18, 1979, to the licensees of all W-designed PWR facilities. This bulletin revised two of the positions contained in IE Bulletin 79-06A.

IE Bulletin 79-06C - This bulletin was issued on July 26, 1979. This bulletin, which is applicable to all operating PWR plants, revises one of the positions in IE Bulletin 79-06A and introduces supplemental requirements. The most salient feature of this bulletin is that it reverses the requirement in the previous bulletins regarding the operation of the reactor coolant pumps upon a small-break LOCA, i.e., this bulletin requires that the reactor coolant pumps be tripped upon a small-break LOCA whereas the previous bulletins required that some of the reactor coolant pumps be kept running.

3.3.2 Actions Completed

The paragraphs that follow describe the actions that have been completed with respect to IE Bulletins 79-06A; 79-06A, Rev. 1; and 79-06C, the bulletins applicable to W-designed plants.

IE Bulletins 79-06A and 79-06A, Revision 1 - Initial Responses - All licensees provided initial responses to all the items in this bulletin within the prescribed schedule.

Evaluation Reports - Draft evaluation reports were prepared for each W-designed plant. Where appropriate, open items and/or deficiencies in the licensees' responses were identified. The licensees were then requested to provide additional information to resolve these matters so that the evaluation reports could be issued. The review of the licensees' responses, as supplemented, has been completed, with all of the open items and deficiencies resolved. The first evaluation report, that for the Salem 1 facility, was issued on December 31, 1979.

IE Bulletin 79-06C - Initial Responses - The initial responses to the various items in this bulletin have either been received directly from each licensee, or from the W Owners Group as a generic response (which has subsequently been incorporated by reference in each specific docket) as follows:

Short-Term Actions:

Item 1: This item requires: (1) that all operating reactor coolant pumps be tripped upon reactor trip and initiation of high pressure injection caused by low reactor coolant system pressure, and (2) that two licensed operators be in the control room at all times to accomplish the above action. Each licensee has implemented this action.

Items 2 and 3: These items require: (1) that the licensees perform analyses for a range of small-break LOCAs and a range of time lapses between reactor trip and pump trip, and (2) that guidelines for operator action for both LOCA and non-LOCA transients be developed. The W Owners Group submitted W report WCAP-9584, "Analysis of Delayed Reactor Coolant Pump Trip During Small Loss-of-Coolant Accident for Westinghouse Nuclear Steam Supply Systems" as a generic response to these items. Since the generic guidelines for operators originally submitted in the small-break analysis report, WCAP-9600, "Report on Small Accidents for Westinghouse NSSS System," were considered consistent with the pump trip guidance, additional guidelines were not proposed. WCAP-9584 has been referenced in each licensee's response to Items 2 and 3. By letters dated October 16, October 31, November 2, and December 21, 1979, the W Owners Group modified the guidelines contained in WCAP-9600. By NRC staff letters dated November 5 and December 6, 1979, we approved the guidelines for emergency operating procedures regarding small-break LOCAs for all operating W-designed plants. Our evaluation of the W analyses pertaining to reactor coolant pump trip is contained in NUREG-0623.

Item 4: This item requires that emergency procedures, based on the guidelines developed under Item 3 above, be developed and that all licensed reactor operators and senior reactor operators be retrained as required. We have audited the procedures developed for Salem 1 (on December 10, 1979) and Surry 1 (on December 11, 1979) as well as the training program being implemented. Results of the audits are summarized in Section 6.0 of this report. Future audits will be conducted by IE.

Item 5: This item, which is related to inadequate core cooling (as discussed in NUREG-0578) concerns analyses and development of guidelines for operational procedures and procedures. By letter dated October 30, 1979, the W Owners Group submitted a document, "Westinghouse Inadequate Core Cooling Analysis Performed to Meet the Requirements Set Forth in NUREG-0578," which addressed this item. This document has been incorporated into the docket of each operating W-designed plant by reference. The procedures associated with this item are required to be implemented by January 31, 1980.

Long-Term Action

Item 1: This item pertains to the design of circuitry which will assure automatic tripping of the operating reactor coolant pumps under all circumstances in which such action may be necessary. To date, only one W-designed operating plant (San Onofre 1) has provided a commitment to install an automatic pump trip feature.

3.3.3 Status of Actions Pending

IE Bulletins 79-06A and 79-06A, Revision 1

As discussed previously, draft evaluations have been prepared for all W-designed plants. One evaluation (Salem 1) has been issued to the licensee (on December 31, 1979). It is expected that the 18 remaining evaluations will be issued in early 1980.

IE Bulletin 79-06C

Item 5: NRC staff approval of the revised W small-break guidelines, incorporating the subject of inadequate core cooling, is not required prior to implementation. Although the NRC staff will review these guidelines and the supporting analyses, no schedule has been identified at this time.

Long-Term

Item 1: By February 15, 1980, each licensee is to submit a schedule for supplying a proposed design for automatic tripping of the reactor coolant pumps. The automatic trip system is required to be installed and made operational by December 31, 1980, unless a later date can be justified, as indicated in Section 7.3.1 of NUREG-0623.

3.4 Orders

No orders, were issued to operating plants with W-designed reactors as a result of the TMI-2 accident.

3.5 Generic Review

3.5.1 Scope

The scope of the generic review of W-designed operating plants included an assessment of the following items:

(1) Comparison of W plant design features; i.e., a general comparison with B&W-designed plants,

(2) Main feedwater systems,

(3) Auxiliary feedwater systems (generic assessment),

(4) Control systems,

(5) Safety systems,

(6) Pressurizer power-operated relief valves and safety valves,

(7) Natural circulation,

(8) Analysis,

(9) Operator training and procedures,

(10) Auxiliary feedwater systems (plant-specific, with generic considerations), and

(11) IE bulletins.

3.5.2 Interfaces with Short-Term Lessons Learned Items

The following items are included in NUREG-0578 as part of the short-term recommendations, but are also within the purview of, and are being managed by, the B&OTF:

2.1.3.b - "Instrumentation for Detection of Inadequate Core Cooling PWRs and BWRs" (analyses and procedures involved with this item, only). The requirements of this item interface with those in Recommendation 2.1.9.

2.1.7.a - "Automatic Initiation of the Auxiliary Feedwater System for PWRs."

2.1.7.b - "AFW Flow Indication to Steam Generators for PWRs."

2.1.9 - "Analysis of Design and Off-Normal Transients and Accidents."

3.5.3 Actions Completed

We performed an evaluation of the auxiliary feedwater systems at each W and C-E-designed operating PWR facility. This evaluation consisted of a deterministic review and a limited reliability assessment. Based on the results of this evaluation, a number of recommendations (both generic and plant-specific) for changes in auxiliary feedwater system design and related procedures were developed. These recommendations were implemented in the form of NRC staff requirements via letters sent to each licensee during the September and November 1979. Copies of these letters are provided in Appendix E of this report. Details of our auxiliary feedwater evaluation are contained in NUREG-0611 as discussed below. To date, not all licensees have responded to these requirements. Of those responses which we have received, we have completed an evaluation of the licensees' responses in some cases. Open items and matters requiring resolution have been identified and transmitted to the licensees. This effort is still in progress.

In response to a number of concerns regarding small-break LOCA analysis and corresponding guidelines for emergency operating procedures expressed in our June 4, 1979, letter to W, the W Owners Group submitted WCAP-9600, "Report on Small Break Accidents for Westinghouse NSSS System," for our review on July 9, 1979. Our review of WCAP-9600 has been completed. The results of our review are documented in NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-designed Operating Plants."

WCAP-9600 also contained proposed W generic guidelines for operational procedures regarding small-break LOCAs. In NRC staff letters dated November 5 and December 6, 1979, we approved the generic guidelines for implementation by licensees with W-designed plants. Copies of our letters approving the guidelines are provided in Appendix G to this report.

On December 10, 1979, representatives of the B&OTF conducted an audit of the emergency procedures and operator retraining associated with small-break LOCAs at Salem 1. In conjunction with the plant audit, a number of NRC representatives also visited Virginia Electric Power Company's Surry Plant Simulator at the Surry 1/2 site to observe its responses to a small-break LOCA. The Salem 1 plant audit and Surry Plant Simulator visit are discussed further in Section 6.0 of this report.

NUREG-0623

As discussed in Section 3.3.2 of this report, our generic evaluation of the W Owners Group response to IE Bulletin 79-06C was reported in NUREG-0623. This evaluation interfaced with our generic review of small-break LOCA analysis and our review of the generic guidelines for operators.

LOFT and Semiscale Tests

In NRC staff letters dated July 9 and September 2, 1979, we requested that the W Owners Group perform predictions of specified Semiscale and LOFT tests, respectively. In their letters dated October 5 and December 15, 1979, the W Owners Group provided the requested predictions for the Semiscale Small Break Experiment (S-07-10B) and LOFT Small Break Experiment L3-1, respectively. These predictions are currently under NRC staff review.

NUREG-0578 Requirements

The B&OTF reviewed responses by licensees with W-designed operating plants to the following requirements of NUREG-0578:

Item 2.1.3.b - "Instrumentation for Detection of Inadequate Core Cooling in PWRs and BWRs" (analyses and procedures only).

Item 2.1.7.a - "Automatic Initiation of the Auxiliary Feedwater System for PWRs."

Item 2.1.7.b - "AFW Flow Indication to Steam Generators for PWRs."

Items 2.1.9 - "Analysis of Design and Off-Normal Transients and Accidents."

Item 2.1.3.b - This item has already been discussed under Item 5 of IE Bulletin 79-06C in Section 3.3.2.

Item 2.1.7.a & 2.1.7.b - Our review of the responses from licensees is continuing. A number of licensees who indicated that the existing auxiliary feedwater system design needs to be modified to meet control grade requirements have not yet provided design information describing the required modifications. In addition, some plants whose licensees claim that the auxiliary feedwater system design meets control grade requirements have not yet provided information describing how these criteria are met. This design information must be reviewed and approved for Item 2.1.7a prior to implementation by licensees.

Item 2.1.9 - Analysis of Design and Off-Normal Transients and Accidents.

(1) Small-break LOCAs - As discussed previously, the W Owners Group submitted WCAP-9600 containing small-break LOCA analyses and guidelines for operational procedures. Our review of WCAP-9600 is reported in NUREG-0611. Our November 5 and December 6, 1979, letters to the W Owners Group approved the guidelines for implementation by each licensee with a W-designed operating plant. Procedures based on these approved guidelines were required to be implemented by December 31, 1979.

(2) Inadequate core cooling - By letter dated October 30, 1979, the W Owners Group submitted the required analyses. The W Owners Group has developed guidelines for emergency procedures which will be submitted to the NRC staff in early January 1980. The procedures based on these guidelines are required to be implemented by January 31, 1980. No prior NRC staff approval of the guidelines is required.

(3) Transients and Accidents - The W Owners Group has informed us that they will provide the required analyses and guidelines for procedures prior to March 31, 1980.

3.5.4 Status of Actions Pending

Small-Break LOCA Guidelines - NRC staff followup is required regarding the audit of licensee implementation of the W generic guidelines at the remaining W-designed operating plants. If significant deviations from the approved guidelines are identified, further NRC staff action may be required.

NUREG-0623 - The proposed designs for automatic tripping of the reactor coolant pumps which will be submitted in response to the specified requirements will need to be reviewed and approved by the NRC staff. Subsequent NRC staff followup will be required to confirm that the approved design has been installed at each plant according to the schedule in Section 7.5.1 of that report.

NUREG-0611 - This report, when published, will be transmitted to each licensee with a W-designed plant for implementation of the recommendations contained therein, according to the schedule specified in Chapter 3 of the report. These recommendations and implementation schedule are included as Table 3-2 of this report. Followup action by the NRC staff will be required to assure that the recommendations are implemented in an acceptable manner by each licensee.

NUREG-0578 Requirements - (Items 2.1.3.b (Analysis & Procedures only), 2.1.7.a, 2.1.7.b, and 2.1.9) - NRC staff followup is required to assure that each licensee with a W-designed operating plant satisfactorily meets the requirements specified for each of the above-listed items according to the implementation schedule contained in the report as modified by H. Denton's October 30, 1979, letter to all operating reactor licensees, a copy of which is provided as Appendix F to this report.

Requirements for Auxiliary Feedwater Systems Developed by the B&OTF

Followup action by the NRC staff is required to assure that each licensee provides a timely response to the NRC staff letters identifying requirements for auxiliary feedwater systems arising from the review by the B&OTF. Further NRC staff action will be required to complete the review of the licensees' responses to these letters.

LOFT and Semiscale Test - Predictions of the Semiscale and LOFT Small-Break LOCA Experiments

NRC staff action is required to review the test predictions submitted by the W Owners Group in response to the requests by the B&OTF.

TABLE 3-2

IMPLEMENTATION OF B&OTF RECOMMENDATIONS
FOR W-DESIGNED OPERATING PLANTS

Recommendation	Schedule
Technical Specification (TS) Time Limit on AFW System Train Outage (GS-1)	1/1/80
TS Administrative Control on Manual AFW System Valves Lock and Verify Position (GS-2)	1/1/80
AFW System Flow Throttling - Water Hammer (GS-3)	1/1/80
Emergency Procedures for Initiating Backup Water Supplies (GS-4)	1/1/80
Emergency Procedures for Initiating AFW Flow Following Loss of All AC Power (GS-5)	1/1/80
AFW System Flow Path Verification (GS-6)	1/1/80
Non-safety Grade Non-redundant AFW System Automatic Initiation Signals (GS-7)	1/1/80
Automatic Initiation of AFW Systems (GS-8)	1/1/80
Primary AFW Source Low Level Alarm	1/1/80
Indication of AFW Flow to the Steam Generators	1/1/80*
AFW System Availability During Periodic Surveillance Testing	1/1/80
Automatic Initiation of AFW Systems (GL-1)	1/1/81
Single Valves in the AFW System Flow Path (GL-2)	1/1/81
Elimination of AFW System Dependency on AC Power Following a Complete Loss of AC Power (GL-3)	1/1/81
Prevention of Multiple Pump Damage Due to Loss of Suction Resulting from Natural Phenomena (GL-4)	1/1/81
Non-safety Grade, Non-redundant, AFW System Automatic Initiation Signals (GL-5)	1/1/81

*Implementation modified by letter, H. Denton to all operating plant licensees, dated October 30, 1979. A copy of that letter is provided in Appendix F to this report.

TABLE 3-2 (Continued)

Recommendation	Schedule
Small Break LOCA Analyses	
Analysis Methods Appendix K	7/1/80
Plant-Specific Appendix K Calculations	1/1/81
Role of Non-safety-Equipment in Mitigating Small Break LOCAs	
RCP Pump Trip	1/1/81
Interaction of Safety and Non-safety Systems	TMI-2 Action Plan
Michelson's Concerns	
Two-phase Natural Circulation Experiments	1/1/81
Instrumentation to Verify Natural Circulation	4/1/80
PORV Failures in W-Designed Plants	
Confirmation of Anticipatory Trip	
Evaluation Elimination of PORV Function	TMI-2 Action Plan
PID Controller Modification	4/1/80
Proposed Anticipatory Trip Modification	Plant-specific
CCI-supplied PORV	Plant-specific
Installation of Auto Isolation of PORVs	7/1/80
Testing of Auto Isolation of PORVs	During first refueling outage following installation.
Westinghouse Report on PORV Failure Reduction	10/1/80
Reporting Failures and Challenges of PORVs and Safety Valves Safety Valves	Failures: Promptly Challenges: In Annual Report

TABLE 3-2 (Continued)

Recommendation	Schedule
Audit Calculations	
Modifications to RELAP4 Heatup Calculation	NRC Action
Effects of Accumulator Injection on RELAP4 Calculations	NRC Action
Modification of RELAP4 to Represent Steam Generator Realistically	NRC Action
Expanded Use of Simulators in Operator Training	
Simulator Training Program	7/1/80
Simulation of Small Break LOCA	1/1/81
Review of Procedures (NRC)	TMI-2 Action Plan
Review of Procedures (NSSS Vendors)	TMI-2 Action Plan
Symptom-Based Emergency Procedures	NRC Action
Monitoring Control Board	4/1/80

4.0 COMBUSTION ENGINEERING, INCORPORATED PLANTS

4.1 Introduction

This section describes the status of the C-E operating plants with respect to the actions taken within the purview of the B&OTF. These actions fall under three specific categories; IE bulletins, Commission Orders, and generic review. The requirements, actions completed, and actions pending are discussed for each of these categories.

4.2 List of Plants

The following operating nuclear power plants with C-E-designed nuclear steam supply systems were reviewed by the B&OTF:

Plant Name	Utility	Power (MWT)	Operating License Issuance Date
Palisades	Consumers Power Company	2530	10/16/72
Maine Yankee	Maine Yankee Atomic Power Company	2630	06/29/73
Fort Calhoun 1	Omaha Public Power District	1420	08/09/73
Calvert Cliffs 1	Baltimore Gas and Electric Company	2700	07/31/74
Millstone 2	Northeast Nuclear Energy Company	2700	08/01/75
St. Lucie 1	Florida Power and Light Company	2560	03/01/76
Calvert Cliffs 2	Baltimore Gas and Electric Company	2700	11/30/76
Arkansas Nuclear One 2 (ANO-2)	Arkansas Power and Light Company	2815	09/01/78

4.3 Bulletins

4.3.1 Applicable Bulletins

The following IE bulletins, copies of which are provided in Appendix A to this report, are applicable to operating plants with C-E-designed reactors:

IE Bulletin 79-05 - This bulletin was issued on April 1, 1979. Although this bulletin was for action by B&W plants, it was also sent to licensees of operating reactors designed by C-E for information purposes, and later incorporated by reference in IE Bulletin 79-06, a bulletin applicable to C-E-designed plants. This bulletin was superseded by IE Bulletin 79-05A.

IE Bulletin 79-05A - This bulletin was issued on April 5, 1979, at which time it superseded IE Bulletin 79-05 and was sent to operating plants designed by C-E for information purposes only. This bulletin was later made applicable to operating nuclear power plants with C-E-designed reactors by reference in IE Bulletins 79-06 and 79-06B, the bulletins applicable to C-E-designed plants.

IE Bulletin 79-06 - This bulletin was issued on April 11, 1979, and, except for B&W facilities, was applicable to all operating PWR plants. This bulletin was superseded by IE Bulletin 79-06B for operating PWR facilities with C-E-designed reactors.

IE Bulletin 79-06B - This bulletin was issued on April 14, 1979, at which time it superseded IE Bulletin 79-06 for operating PWR plants with C-E-designed reactors. The requirements contained in this bulletin represent the greatest number of actions that must be taken by licensees with C-E-designed reactors.

IE Bulletin 79-06C - This bulletin was issued on July 26, 1979. This bulletin, which is applicable to all operating PWR plants, revises one of the positions in IE Bulletin 79-06B and introduces supplemental requirements. The most salient feature of this bulletin is that it reverses the requirement in the previous bulletins regarding the operation of the reactor coolant pumps upon a small-break LOCA (i.e., this bulletin requires that the reactor coolant pumps be tripped upon a small-break LOCA whereas the previous bulletins required that some of the reactor coolant pumps be kept running).

4.3.2 Actions Completed

The paragraphs that follow describe the actions that have been completed with respect to IE Bulletins 79-06B and 79-06C, the bulletins applicable to C-E-designed plants.

IE Bulletin 79-06B:

Initial Responses - All licensees provided initial responses to all of the items in this bulletin within the prescribed schedule.

Evaluation Reports - Draft evaluation reports have been prepared for each C-E-designed plant. Where appropriate, open items were identified and the licensees have provided additional information to resolve these matters so that all of the evaluation reports can now be issued. To date, we have issued evaluation reports for the Millstone 2 and Calvert Cliffs 1-2 plants.

IE Bulletin 79-06C:

Initial Responses - The initial responses to the various items in this bulletin have either been provided directly from each licensee or from the C-E Owners Group as a generic response which has subsequently been incorporated by reference in each specific docket as follows:

Short-Term Actions:

Item 1: This item requires: (1) that all operating reactor coolant pumps be tripped upon reactor trip and initiation of high pressure injection caused by low reactor coolant system pressure, and (2) that two licensed operators be in the control room at all times to accomplish the above action. Each licensee has implemented this action. Initially, Palisades did not implement pump trip, but has now done so.

Items 2 and 3: These items require: (1) that the licensees perform analyses range of small-break LOCAs and a range of time lapses between reactor trip and pump trip, and (2) that guidelines for operator action, for both LOCA and non-LOCA transients, be developed for operator actions. The C-E Owners Group submitted CEN-115P, "Response to NRC IE Bulletin 79-06C Items 2 and 3 for Combustion Engineering Nuclear Steam Supply Systems," as a generic response to these items. This report has been referenced by each licensee in response to Items 2 and 3 of the bulletin. By letter dated November 8, 1979, the C-E Owners Group revised the guidelines contained in CEN-115P. In an NRC staff letter dated November 14, 1979, we approved the guidelines for all operating C-E-designed plants (except for Maine Yankee) and transmitted our evaluation report of the guidelines. By letter dated December 13, 1979, we received additional information from the C-E Owners Group pertaining to the guidelines for Maine Yankee. Our review of the revisions to the guidelines for Maine Yankee were approved by letter dated December 26, 1979. Our evaluation of the C-E analyses pertaining to the tripping of reactor coolant pumps is contained in NUREG-0623, "Generic Assessment of Delayed Reactor Coolant Pump Trip During Small Break Loss-of-Coolant Accidents in Pressurized Water Reactors." Copies of the November 14 and December 26, 1979 letters are provided in Appendix G to this report.

Item 4: This item requires that emergency procedures, based on the guidelines developed under Item 3 above, be developed and that all licensed reactor operators and senior reactor operators be retrained as required. On December 19, 1979, we audited the procedures developed for Millstone 2 as well as the training program being implemented. Future audits will be conducted by IE.

Item 5: This item relates to inadequate core cooling as discussed in NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," regarding analyses and development of guidelines and procedures. By letter dated October 31, 1979, the C-E Owners Group submitted CEN-117, "Inadequate Core Cooling - A Response to NRC IE Bulletin 79-06C, Item 5 for Combustion Engineering Nuclear Steam Supply Systems." This report has been incorporated in the docket of each operating C-E-designed plant by reference. The procedures associated with this item are required to be implemented by January 31, 1980.

4.3.3 Status of Actions Pending

IE Bulletin 79-06B

As discussed above, draft evaluations have been prepared for all C-E-designed plants. Two evaluations (Millstone 2 and Calvert Cliffs 1-2) have been issued to the licensees. It is expected that the five remaining evaluations will be issued in early 1980.

IE Bulletin 79-06C

Item 5: NRC staff approval of the revised C-E small-break guidelines, incorporating the subject of inadequate core cooling, is not required prior to implementation. Although the NRC staff will review these guidelines and the supporting analyses, no schedule has been identified at this time.

Long-Term

Item 1: By February 15, 1980, each licensee is to submit a schedule for supplying a proposed design for automatic tripping of the reactor coolant pumps. The automatic trip system is required to be installed and made operational by December 31, 1980, unless a later date can be justified, as indicated in Section 7.3.1 of NUREG-0623.

4.4 Orders

No orders were issued to operating plants with C-E-designed reactors as a result of the TMI-2 accident.

4.5 Generic Review

4.5.1 Scope

The scope of the generic review of C-E operating plants included an assessment of the following items:

(1) Comparison of C-E plant design features, i.e., a general comparison with B&W designed plants,

(2) Main feedwater systems,

(3) Auxiliary feedwater systems (generic assessment),

(4) Control systems,

(5) Safety systems,

(6) Pressurizer power-operator relief valves and safety valves,

(7) Natural circulation,

(8) Analysis,

(9) Operator training and procedures,

(10) Auxiliary feedwater systems (plant specific with generic considerations), and

(11) IE bulletins.

4.5.2 Interfaces with Short-Term Lessons Learned Items

The following items are included in NUREG-0578 as part of the short-term recommendations, but are also within the purview of and are being managed by the B&OTF:

2.1.2.b - "Instrumentation for Detection of Inadequate Core Cooling in PWRs and BWRs (analyses and procedures involved with this item, only). The requirements of this item interface with those in Recommendation 2.1.9.

2.1.7.a - "Automatic Initiation of the Auxiliary Feedwater System for PWRs.

2.1.7.b - "AFW Flow Indication to Steam Generators for PWRs.

2.1.9 - Analysis of Design and Off-Normal Transients and Accidents.

4.5.3 Actions Completed

We performed an evaluation of the auxiliary feedwater systems at each W and C-E-designed operating PWR plant. This evaluation consisted of a deterministic review and a limited reliability assessment. Based on the results of our evaluations, a number of recommendations were developed. These recommendations involved both plant-specific and generic issues pertaining to auxiliary feedwater systems designs and operating procedures. These recommendations were imposed on the licensees by letters sent to each licensee during the period of mid-September to mid-November 1979. Copies of these letters are provided in Appendix E to this report. Details of our auxiliary feedwater system evaluation are contained in NUREG-0635, "Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in Combustion Engineering-Designed Operating Plants". Additional details regarding this report are discussed below. Not all of the licensees have responded to the requirements stipulated in the aforementioned letters. Although we have not yet completed our evaluation of the responses that have been submitted, we have identified certain open items and matters requiring resolution and have transmitted these concerns to the licensees. The effort associated with the review of auxiliary feedwater systems modifications is still in progress.

In response to a number of our concerns regarding small-break LOCA analyses and corresponding guidelines for developing emergency operating procedures, as expressed in our June 5, 1979, letter to all operating C-E plant licensees, the C-E Owners Group submitted CEN-114P (Amendment 1P) "Review of Small Break Transients in Combustion Engineering Nuclear Steam Supply Systems." Our review of CEN-114P has been completed and the results are documented in NUREG-0635.

CEN-114P also contained proposed generic guidelines for developing emergency procedures for coping with small-break LOCAs at C-E-designed operating plants. Based on our review of these guidelines, they were amended to include applicable B&OTF recommendations. The amended guidelines were submitted by in a letter from the C-E Owners Group on November 7, 1979. By letter dated November 14, 1979, we approved the guidelines for plants with high pressure injection pumps with shutoff heads less than 1600 psi (i.e., all plants except Maine Yankee).

Additional information was provided by the C-E Owners Group in a letter dated December 13, 1979, to account for the Maine Yankee plant. By letter dated December 26, 1979, we approved the information contained in the December 13, 1979, letter for development of procedures at Maine Yankee subject to the C-E Owners Group providing certain confirmatory data. Copies of our letters approving the guidelines are provided in Appendix G to this report.

On December 19, 1979, representatives of the B&OTF conducted an audit of the emergency procedures and operator retraining associated with small-break LOCAs at Millstone 2. In conjunction with the plant audit, the B&OTF representatives also visited C-E's PWR Simulator located at Windsor, Connecticut to observe its responses to a small-break LOCA. The Millstone 2 plant audit and C-E PWR simulator visit are discussed further in Section 6.0 of this report.

As stated earlier in Section 4.2.2 of this report, our generic evaluation of the C-E Owners Group's response to IE Bulletin 79-06C is contained in NUREG-0623. The evaluation contained therein interfaces with our generic review of the small-break LOCA analyses and our review of the generic guidelines for developing emergency procedures for coping with small-break LOCAs.

Semiscale and LOFT Pretest Predictions - By letters dated November 19 and December 3, 1979, the C-E Owners Group provided predictions of specified LOFT and Semiscale tests as previously requested by the B&OTF. These pretest predictions were for the LOFT Small-Break Experiment L3-1 and the

Semiscale Small-Break Experiment S-07-10B, respectively. The information provided in these predictions is currently under review.

The B&OTF reviewed responses by licensees with C-E-designed operating plants to the following requirements contained in NUREG-0578:

Item 2.1.3.b - "Instrumentation for Detection of Inadequate Core Cooling in PWRs and BWRs" analyses and procedures only. This item has been completed as discussed under Item 5 of IE Bulletin 79-06C.

Item 2.1.7.a - "Automatic Initiation of the Auxiliary Feedwater System for PWRs." Our review of the licensees' responses to this item is under review. Some licensees have submitted their designs of control grade type initiation, but have indicated that this may violate some of their previous auxiliary feedwater system criteria (i.e., introducing control-grade equipment in the operation of Class IE equipment), and others have indicated that the automatic initiation of auxiliary feedwater system may introduce an unreviewed safety question (i.e., return to power concerns and high containment pressure upon a steamline break inside containment with auxiliary feedwater system flow). These matters are under review.

Item 2.1.7.b - "AFW Flow Indication to Steam Generators for PWRs." Most licensees claim that they meet our requirements regarding control-grade indication of auxiliary feedwater system flow to the steam generators (one, ANO-2, meets safety-grade requirements); however, all design information has not yet been submitted for evaluation.

Item 2.1.9 - Analysis of Design and Off-Normal Transients and Accidents." The status of our review of information regarding this item follows:

(1) Small-Break LOCAs - As stated previously, the C-E Owners Group submitted CEN-114P containing the small-break LOCA analyses and and guidelines for operating procedures. Our review of this document is reported in NUREG-0635, and our approval of the guidelines is contained in our letters to C-E Owners Group dated November 14 and December 26, 1979.

(2) Inadequate Core Cooling - In a letter dated October 31, 1979, the C-E Owners Group submitted CEN-117 which contained the required analysis. The information contained in this report will be used to develop the procedures associated with inadequate core cooling. The procedures based on these guidelines are to be implemented by January 31, 1980.

(3) Transients and Accidents - The C-E Owners Group will provide the required analyses and guidelines for procedures in early 1980 as specified in NUREG-0578.

4.5.4 Status of Actions Pending

(1) Small-Break LOCA Guidelines - NRC staff followup actions are required to audit licensee implementation of the C-E generic guidelines at the remaining C-E-designed operating plants. If significant deviations from the approved guidelines are identified, further NRC staff action may be required. It is anticipated that future audits will be conducted by IE.

(2) NUREG-0623 - The proposed designs for automatically tripping the reactor coolant pumps upon a small-break LOCA have not been received. Accordingly, this information must be submitted and reviewed by the NRC staff and followup actions will be required to ensure that the designs conform with our requirements. The schedule for this information is contained in Section 7.5.1 of NUREG-0623.

(3) NUREG-0635 - This report will be transitted to each licensee with a C-E-designed reactor for implementation of the recommendations contained therein. The schedule for implementing the recommendations of this report is contained in Chapter 3. The recommendations and implementation schedules are included as Table 4-1 of this report. Followup NRC staff actions will be required to assure that the recommendations are implemented in conformance with the stipulated requirements.

TABLE 4-1

IMPLEMENTATION OF B&OTF RECOMMENDATIONS FOR C-E-DESIGNED OPERATING PLANTS

Recommendation	Schedule
Technical Specification (TS) Time Limit on AFW System Train Outage (GS-1)	1/1/80
TS Administrative Control on Manual Valves - Lock and Verify Position (GS-2)	1/1/80
AFW System Flow Throttling - Water Hammer (GS-3)	1/1/80
Emergency Procedures for Initiating Backup Water Supplies (GS-4)	1/1/80
Emergency Procedures for Initiating AFW Flow Following Loss of All AC Power (GS-5)	1/1/80
AFW System Flow Path Verification (GS-6)	1/1/80
Automatic Initiation of AFW Systems (GS-8)	1/1/80
Primary AFW Source Low Level Alarm	1/1/80
Indication of AFW Flow to the Steam Generators	1/1/80*
AFW System Availability During Periodic Surveillance Testing	1/1/80
Automatic Initiation of AFW Systems (GL-1)	1/1/81
Single Valves in the AFW System Flow Path (GL-2)	1/1/81
Elimination of AC Power Dependency (GL-3)	1/1/81
Prevention of Multiple Pump Damage Due to Loss of Suction Resulting from Natural Phenomena (GL-4)	1/1/81
Confirmation of Small-Break LOCA Analysis Methods	
Analysis Methods Appendix K	7/1/80
Plant-Specific Appendix K Calculations	1/1/81

*Implementation modified by letter, H. Denton to all operating plant licensees, dated October 30, 1979. A copy of that letter is provided in Appendix F to this report.

TABLE 4-1 (Continued)

Recommendation	Schedule
Role of Non-Safety-Equipment in Mitigating Small-Break LOCAs	
Automatic Trip of RCPs	1/1/81
Review of Reliability & Redundancy of Equipment	TMI-2 Action Plan
Michelson's Concerns	
Two-phase Natural Circulation Experiments	1/1/81
Instrumentation to Verify Natural Circulation	4/1/80
PORV Failures in C-E Plants	
Installation of Automatic Isolation of PORVs	7/1/80
Testing Automatic Isolation of PORVs	(First Refueling Outage After Installation)
C-E Report on PORVs Failure Reductions	10/1/80
Reporting Future Failures and Challenges of PORVs and SVs	Failures: Promptly Challenges: In Annual Report
Evaluation Elimination of PORV Function	TMI-2 Action Plan
Audit Calculations:	
Modification to RELAP and CEFLASH-4AS Due to Uncertainties in Heatup Calculations	RELAP: TMI-2 Action Plan CEFLASH: 7/1/80
Effects of Accumulator Injection on RELAP-4 Calculation	NRC Action
Modification of RELAP4 to Represent SG Behavior Realistically	NRC Action
Expanded Use of Simulators in Operator Training:	
Simulator Training Program	7/1/80
Simulation of Small-Break LOCAs	1/1/81
Review of Procedures (NRC)	TMI-2 Action Plan
Review of Procedures (NSSS Vendors)	TMI-2 Action Plan
Symptom-Based Emergency Procedures	TMI-2 Action Plan
Monitoring Control Board	4/1/80

5.0 BOILING WATER REACTOR PLANTS

5.1 Introduction

This section describes the status of the BWR operating plants and certain near-term operating license applications with respect to the actions taken within the purview of the B&OTF. These actions fall under three specific categories: IE bulletins, Commission Orders, and generic review. The requirements, actions completed, and actions pending are discussed for each of these categories.

5.2 List of Plants

The BWR plants evaluated by the B&OTF are listed in Table 5-1. Except for LaCrosse, which utilizes a nuclear steam supply system designed by Allis-Chalmers, all of the plants utilize nuclear steam supply systems designed by GE. Except for LaSalle 1-2, Shoreham, and Zimmer, which are near-term operating license applications, all of the plants have operating licenses.

Section 5.3 of this report, which deals with IE Bulletin 79-08, applies to all of those plants with operating licenses. Section 5.5 of this report, which deals with the B&OTF generic review, applies to all of the plants listed in Table 5-1, including LaSalle 1-2, Shoreham, and Zimmer. As indicated in Section 5.4 of this report, no Commission Orders were issued to BWR plants as a result of the TMI-2 accident.

TABLE 5-1

BOILING WATER REACTOR PLANTS EVALUATED BY THE BULLETINS AND ORDERS TASK FORCE

Plant	BWR Type	Utility	Power Level	Operating License Date
Big Rock Point	1	Consumers Power Co.	240	08/30/62
Browns Ferry 1	4	Tennessee Valley Authority	3293	06/26/73
Browns Ferry 2	4	Tennessee Valley Authority	3293	06/28/74
Browns Ferry 3	4	Tennessee Valley Authority	3293	07/02/76
Brunswick 1	4	Carolina Power & Light Co.	2436	09/08/76
Brunswick 2	4	Carolina Power & Light Co.	2436	10/27/74
Cooper	4	Nebraska Public Power District	2381	01/18/74
Dresden 1	1	Commonwealth Edison Co.	700	09/28/59
Dresden 2	3	Commonwealth Edison Co.	2527	12/22/69
Dresden 3	3	Commonwealth Edison Co.	2527	01/12/71
Duane Arnold	4	Iowa Electric Light & Power Co.	1658	02/22/74
FitzPatrick	4	Power Authority of the State of NY	2436	10/17/74
Hatch 1	4	Georgia Power Co.	2436	08/06/74
Hatch 2	4	Georgia Power Co.	2436	06/13/78
Humboldt Bay	1	Pacific Gas & Electric Co.	220	08/28/62
LaCrosse*	-	Dairyland Power Cooperative	165	07/03/67
LaSalle 1**	5	Commonwealth Edison Co.	3293	-
LaSalle 2**	5	Commonwealth Edison Co.	3293	-
Millstone 1	3	Northeast Nuclear Energy Co.	2011	10/06/70
Monticello	3	Northern States Power Co.	1670	09/08/70
Nine Mile Point 1	2	Niagra Mohawk Power Co.	1850	08/22/69
Oyster Creek	2	Jersey Central Power & Light Co.	1930	04/09/69
Peach Bottom 2	4	Philadelphia Electric Co.	3293	08/08/73
Peach Bottom 3	4	Philadelphia Electric Co.	3293	07/02/74
Pilgrim 1	3	Boston Edison Co.	1998	06/08/72
Quad Cities 1	3	Commonwealth Edison Co.	2511	10/01/71
Quad Cities 2	3	Commonwealth Edison Co.	2511	03/31/72
Shoreham**	4	Long Island Lighting Co.	2436	-
Vermont Yankee	4	Vermont Yankee Nuclear Power Co.	1593	03/21/72
Zimmer**	5	Cincinnati Gas & Electric Co.	2436	-

*Nuclear steam supply system designed by Allis-Chalmers; all others designed by GE.
**Plants with near-term operating license applications; all others have operating licenses.

5.3 Bulletins

5.3.1 Applicable Bulletin

On April 14, 1979, IE Bulletin 79-08, "Events Relevant to Boiling Water Reactor Power Reactors Identified During Three Mile Island Incident," was issued for action to all BWR plants with operating licenses and for information to all other plants with construction permits or operating licenses. IE Bulletin 79-08 contained a number of general review actions required by the licensees with operating BWR plants as well as number of actions that required specific changes in plant design or operating procedures. A copy of IE Bulletin 79-08 is provided in Appendix A to this report.

5.3.2 Actions Completed

The licensees' responses to IE Bulletin 79-08 were received in late April and early May 1979. To assure consistency in our review of the licensees' responses to IE Bulletin 79-08, a Bulletin Review Team was established. The Bulletin Review Team was composed of five members representing the B&OTF, the Division of Operating Reactors and IE. The disciplines represented by the B&OTF members of the Bulletin Review Team include project management, reactor systems and operator licensing.

As a result of its review of the licensees' responses to IE Bulletin 79-08, the Bulletin Review Team developed a number of requests for additional information. These requests for additional information were issued to the licensees in mid-July 1979. The licensees responses to these requests for additional information were received in early August 1979 and were reviewed by the Bulletin Review Team. Any outstanding problem areas in the licensees' responses were identified to the cognizant Operating Reactors Project Manager for resolution in conjunction with the licensees and the Bulletin Review Team. In addition, the Bulletin Review Team identified a number of generic concerns for further consideration by the B&OTF.

Bulletin Evaluation Reports (BEVRs), which summarize the Bulletin Review Team's evaluation of the licensees' responses to IE Bulletin 79-08, are being prepared for each BWR plant with an operating license. The status of the BEVR effort for each plant is provided in Table 5-2.

5.3.3 Status of Actions Pending

The BEVR effort for all the BWR plants except Dresden 1 is expected to be completed in early 1980. Since Dresden 1 is expected to remain shut down until November 1980 for chemical cleaning and modifications, the licensee has deferred its response to IE Bulletin 79-08. A BEVR for Dresden 1 is expected to be issued prior to restart.

5.4 Orders

No Commission Orders were issued to BWR plants as a result of the TMI-2 accident.

TABLE 5-2

IE BULLETIN 79-08 EVALUATION REPORT (BEVR) STATUS

Plant	BEVR Status*			
	A	B	C	D
Big Rock Point			X	
Browns Ferry 1-3	X			
Brunswick 1-2	X			
Cooper	X			
Dresden 1				X
Dresden 2-3		X		
Duane Arnold	X			
FitzPatrick		X		
Hatch 1-2	X			
Humboldt Bay	X			
LaCrosse			X	
Millstone 1	X			
Monticello	X			
Nine Mile Point 1		X		
Oyster Creek	X			
Peach Bottom 2-3		X		
Pilgrim 1	X			
Quad Cities 1-2		X		
Vermont Yankee	X			

*Legend:
A - BEVR already issued or issuance expected momentarily.
B - BEVR draft under review by Bulletin Review Team.
C - BEVR draft not yet prepared due to manpower limitations.
D - Bulletin response deferred by licensee due to extended shutdown; to be completed prior to restart.

5.5 Generic Review

5.5.1 Scope

The scope of the B&OTF generic review of BWR plants is limited to loss-of-feedwater and small-break loss-of-coolant events. The specific areas of review related to these events are as follows:

(1) Reliability of systems,

(2) Analyses,

(3) Guidelines for the preparation of emergency operating procedures, and

(4) Retraining of operators in emergency operating procedures.

All of the boiling water reactor plants listed in Table 5-1, including LaSalle 1-2, Shoreham, and Zimmer, were included in the generic review.

5.5.2 Interfaces with Short-Term Lessons Learned Items

In conducting its activities in connection with BWR plants, the B&OTF had an overlapping responsibility with certain short-term Lessons Learned items as follows:

Item 2.1.3b (Part 1 only)	-	"Instrumentation for Inadequate Core Cooling" (Develop Procedures and Describe Existing Instrumentation)
Item 2.1.9	-	"Transient and Accident Analysis"

These items, which are described in detail in NUREG-0578, "TMI-2 Lessons Learned Task Force Status and Short-Term Recommendations," were reviewed separately by the B&OTF independent of the parallel efforts in the Division of Operating Reactors related to the other short-term Lessons Learned items. This was done since these requirements originated from the generic reviews which were conducted by the B&OTF. Items 2.1.7a, "Auto Initiation of Auxiliary Feed," and 2.1.7b, "Auxiliary Feed Flow Indication," while reviewed separately by the B&OTF for PWR plants, are not applicable to BWR plants.

5.5.3 Actions Completed

Loss of Feedwater and Small-Break Loss-of-Coolant Events

The B&OTF's generic review of the loss of feedwater and small-break loss-of-coolant events was initiated for the BWR plants in early June 1979. In late June 1979, a meeting was held with all the BWR plant licensees to discuss the review. The licensees with BWR plants utilizing nuclear steam supply systems designed by GE formed the General Electric Operating Plant Owners Group (Owners Group) in late June 1979. The purpose of the Owners Group was to coordinate those generic activities related to the TMI-2 accident among the GE BWR plant owners, GE and the NRC. LaCrosse was handled directly with its licensee since it is the only BWR plant which utilizes a nuclear steam supply system not designed by GE (Allis-Chalmers designed the LaCrosse nuclear steam supply system).

The B&OTF's initial requests for short-term information and long-term information were issued to all BWR licensees in mid-July 1979. Subsequent requests for information were issued to the Owners Group and the LaCrosse licensee in late July, late August and early September 1979. The short-term information was to be used to prepare the B&OTF's generic report on BWRs; the long-term information was to be used to evaluate Lessons Learned Items 2.1.3b and 2.1.9 and other matters not required for the generic report. Meetings were held with the Owners Group in mid-July and early September 1979 to discuss the scope of the responses to the B&OTF's requests for short-term and long-term information, respectively.

In mid-August 1979, the Owners Group submitted GE Report NEDO-24708, "Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors." NEDO-24708 is the Owners Group's generic report responding to the B&OTF's short-term information requests.

Initial requests for additional information on NEDO-24708 were issued to the Owners Group in September 1979. It was subsequently determined that not all of the short-term information necessary to prepare the B&OTF's generic report had been requested. Therefore, requests for additional short-term information were issued to the Owners Group and the LaCrosse licensee in September, October and November 1979.

The results of the B&OTF's generic evaluation of the loss-of-feedwater and small-break loss-of-coolant events for those BWR plants utilizing nuclear steam supply systems designed by GE are documented in NUREG-0626, "Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in GE-Designed Operating Plans and Near-Term Operating License Applications."

The LaCrosse licensee's responses to the B&OTF's initial request for information and certain subsequent requests for additional information were submitted in early September and mid-November 1979. Because of their late submittal, the B&OTF was unable to review the LaCrosse licensee's responses in the same time frame as the Owners Group's responses were reviewed. Consequently, we were unable to include LaCrosse in NUREG-0626. We expect that a separate report documenting the results of our evaluation of the loss of feedwater and small-break loss-of-coolant events for LaCrosse will be issued upon the completion of the NRC staff's review of the information provided by the licensee.

Short-Term Lessons Learned Items

As discussed in Section 5.5.2 of this report, the B&OTF reviewed Lessons Learned Items 2.1.3b and 2.1.9 in conjunction with its activities associated with BWR plants. Lessons Learned Item 2.1.9 deals primarily with analyses, emergency procedures and operator retraining associated with small-break LOCAs, inadequate core cooling, and accidents and transients. The emergency procedures and operator training associated with each of these matters was to be completed by December 31, 1979, January 1980, and three months after the guidelines are established in early 1980, respectively. In addition, Lessons Learned Item 2.1.9 includes the analysis of LOFT small-break tests. In the case of the GE-designed BWRs, however, GE's Two-Loop Test Apparatus (TLTA) is being utilized in lieu of the LOFT facility. Lessons Learned Item 2.1.3b is being reviewed along with the inadequate core cooling portion of Item 2.1.9.

The small-break LOCA operator guidelines for GE-designed BWR plants are contained in Section 3.1.1.2 of NEDO-24708. As a result of our review, these operator guidelines were modified in accordance with letters from the Owners Group dated October 18, 1979, and October 31, 1979. The modified guidelines were approved by the B&OTF as documented in our letters to the Owners Group dated October 26, 1979 and November 28, 1979. Copies of these letters are provided in Appendix G to this report.

The licensees of GE-designed BWR plants proceeded with their development of the small-break LOCA emergency procedures and operator retraining based on the modified guidelines. Toward this end, the Owners Group conducted a seminar for its member licensees in St. Louis, Missouri on November 7-8, 1979, to assist the licensees in the development of their emergency procedures and operator retraining.

On December 10-13, 1979, representatives of the B&OTF conducted audits of the emergency procedures and operator retraining associated with small-break LOCAs at the Nine Mile Point 1, FitzPatrick, and Dresden 2/3 plants. In conjunction with these plant audits, a number of B&OTF representatives also visited GE's BWR Simulator at Morris, Illinois, to observe its responses to a small-break LOCA. The Nine Mile Point 1, FitzPatrick, and Dresden 2/3 plant audits and the GE BWR Simulator visit are discussed further in Section 6.0 of this report.

At a meeting with the B&OTF on October 25, 1979, the Owners Group described its approach to the development of emergency procedures to cope with inadequate core cooling. The approach is divided into mechanistic and nonmechanistic portions. The Owners Group maintained that the only realistic mechanism for inadequate core cooling involves core uncovery. The operator guidelines emergency procedures, and operator retraining associated with this aspect of inadequate core cooling were implemented along with those associated with small-break LOCAs, since the additional modifications required were not major. The Owners Group maintained that the nonmechanistic portion will require the development of completely new emergency procedures and will, therefore, not be implemented until sometime in 1980. We concluded that this two-phase approach to develop inadequate core cooling emergency procedures is acceptable.

As discussed above, the mechanistic inadequate core cooling operator guidelines were integrated with the small-break LOCA guidelines which were approved by the B&OTF as documented in our letters to the Owners Group dated October 26, 1979, and November 28, 1979. The emergency procedures and operator retraining associated with this aspect of inadequate core cooling were implemented along with those associated with small-break LOCAs. Therefore, the emergency procedures and operator retraining associated with the mechanistic aspect of inadequate core cooling were implemented by December 31, 1979, prior to the required completion date of January 1980.

It is important to note, however, that the mechanistic inadequate core cooling operator guidelines were not approved by the B&OTF. In order to assure their timely implementation, the B&OTF requested that the licensees proceed with the development of their emergency procedures and operator retraining associated with inadequate core cooling upon the establishment of the operator guidelines. Therefore, although the licensees have implemented their emergency procedures and operator retraining associated with inadequate core cooling in accordance with our requirements, it is possible that some changes to the operator guidelines and, consequently, to the emergency procedures and operator retraining may be required upon the completion of the NRC staff's review.

The analyses and operator guidelines associated with accidents and transients are scheduled to be submitted in early 1980 in accordance with our requirements. The emergency procedures and operator retraining associated with accidents and transients are required to be completed within three months after the operator guidelines are established.

As mentioned earlier, GE's TLTA facility is being utilized in lieu of the LOFT facility for the small-break LOCA tests for GE-designed BWR plants. Two TLTA tests are planned - the first with high-pressure emergency core cooling systems available; the second with the high-pressure emergency core cooling systems unavailable. For each test, a pretest prediction and a post-test calculation, including a discussion of any differences between the pretest prediction and the test results, will be performed. Pretest predictions for the first test, with high-pressure emergency core cooling systems available, have been submitted and the first test has been completed. The second test, with high-pressure emergency core cooling systems unavailable is scheduled to be completed during the first quarter of 1980.

5.5.4 Status of Actions Pending

Loss of Feedwater and Small-Break Loss-of-Coolant Events

As discussed earlier, the results of the B&OTF's generic evaluation of the loss of feedwater and small-break loss-of-coolant events for the GE-designed BWR plants are documented in NUREG-0626. NUREG-0626 contains a number of recommendations that the licensees make certain systems and procedural modifications and conduct specific additional analyses as well as a proposed implementation schedule for these recommendations. These recommendations and implementation schedule are included as Table 5-3 of this report. These recommendations will be reviewed by the Director of the Office of Nuclear Reactor Regulation and issued as documented, or modified, as requirements to be implemented. As also discussed earlier, a separate report is expected to be prepared for LaCrosse, which, because of its unique nuclear steam supply system design, is being reviewed separately. It is anticipated that analagous recommendations and requirements will be forthcoming for implementation by the LaCrosse licensee.

Short-Term Lessons Learned Items

As discussed earlier, we have not completed our review of the mechanistic inadequate core cooling operator guidelines. It is possible that some changes to the operator guidelines and, consequently, to the emergency procedures and operator retraining associated with inadequate core cooling may be required upon the completion of the NRC staff's review. As also discussed earlier, the nonmechanistic inadequate core cooling guidelines have not been completed. These operator guidelines and the attendant emergency procedures and operator retraining are expected to be implemented sometime in 1980.

TABLE 5-3

IMPLEMENTATION OF B&OTF RECOMMENDATIONS FOR GE-DESIGNED OPERATING PLANTS AND NEAR-TERM OPERATING LICENSE APPLICATIONS

Recommendation	Action Required	Schedule[(a)]
Separation of HPCI and RCIC initiation levels	(1) Analysis (2) Implementation	S L
Isolation of isolation condensers on high radiation	Modify isolation circuitry	S
Spurious isolation of HPCI and RCIC	Modify break detection circuitry	S
Reduction of challenges and failures of relief valves	(1) Feasibility study (2) System modification	S L
Identify water source prior to manual ADS	Modify guidelines and procedures	S
Report on outage of ECC systems	(1) Report submittal (2) Plant-specific technical specification changes	SS S
Modification of ADS logic	(1) Feasibility study for staff review (2) Modification to ADS logic	S L
Interlock on recirculation pump loops	Install interlocks for non-jet pump plants	S
Loss of service water for Big Rock Point	Verify acceptability of consequences	S
Restart of core spray and LPCI on low level	(1) Preliminary design (2) Modification of restart logic	S L
Revised emergency procedures	All operators must have read prior to going on duty	SSS
Revise small break LOCA model for compliance with Appendix K	(1) Revise model (2) Compare with TLTA data	S S

TABLE 5-3 (Continued)

Recommendation	Action Required	Schedule[a]
Plant-specific analysis with revised model	Submit analyses with revised model	L
No fuel failure requirement for anticipated transient with single failure	Verify compliance with requirement	S
Depressurization with other than ADS	Analyses to support other modes	S
Two operators in control room	Minimum of two operators in control room	SSS
Michelson concerns	GE address concerns	SSS
Automatic switchover of RCIC suction	(1) Verify procedures (2) Design modification	SS L
Central water level recording	Installation of recorders	L
Space cooling for HPCI and RCIC	Demonstrate minimum of two-hour capacity	L
Effect of loss of AC power on pump seals	Demonstrate adequacy of seal design	L
Use of RHR for fuel pool cooling	Risk assessment	L
Common reference for level instruments	Modify scale to obtain common reference	S
Qualification of accumulators on ADS valves	Show acceptability	L
Guidelines for symptom-based emergency procedures	Develop new guidelines	LL
Test program for SB LOCA model verification	(1) Pre-test prediction of first two tests (2) Develop test program (3) Model verification	SSS S LL
Diverse initiation signal for RCIC	Upgrade if required	L
Small-break LOCA on simulators	Upgrade simulator	L

TABLE 5-3 (Continued)

Recommendation	Action Required	Schedule[a]
Use of non-ECC systems in analysis	(1) Review system capability (2) Upgrade if needed	L
Performance of isolation condensers with noncondensibles	Demonstrate adequacy	L
Reporting of failures and challenges to SRVs	Prompt reporting of failures and annual report of challenges	N/A
Impact of B&O recommendations	Assess impact on safety and reliability	L

(a) Category S: Implement by June 30, 1980.
Category SS: Implement within 60 days of this report.
Category SSS: Implement within 30 days of this report.
Category L: Implementation by January 1, 1981.
Category LL: Implementation by January 1, 1983.

As discussed earlier, the analyses and operator guidelines associated with accidents and transients are scheduled to be submitted in early 1980. The emergency procedures and operator retraining associated with accidents and transients are required to be completed within three months after the operator guidelines are established.

As discussed earlier, the test results and post-test calculation for the first TLTA test, with high-pressure emergency core cooling systems available, have yet to be submitted. As also discussed earlier, the second TLTA test, with high-pressure emergency core cooling systems unavailable, including the attendant pretest predictions, test results, and post-test calculation remain to be accomplished.

6.0 AUDITS OF SMALL-BREAK LOCA EMERGENCY PROCEDURES AND OPERATOR RETRAINING

6.1 Background

Representatives of the B&OTF conducted audits of licensees' emergency procedures and operator retraining associated with small-break LOCA's at selected W, C-E, and GE operating plants. The plants that were audited and the dates that the audits were conducted are as follows:

Plant		Audit Date(s)
W Plant:*	Salem 1	December 10, 1979
C-E Plant:*	Millstone 2	December 19, 1979
GE Plants:	Nine Mile Point 1	December 10, 1979
	FitzPatrick	December 11, 1979
	Dresden 2/3	December 12, 1979

The purpose of the B&OTF audits was to review selected licensees emergency procedures, operator retraining, operator awareness of the emergency procedures and their bases, and systems considerations associated with small-break LOCAs. The plants selected for the B&OFT audits were based on their representation of the various nuclear steam supply system vendors' plants and licensees and on the limited time available to conduct the audits. Audits of the remaining operating plants will be conducted by the Office of Inspection and Enforcement.

*A number of other W and C-E plants were scheduled to be audited; however, these audits had to be cancelled due to inclement weather in the vicinity of the sites, or staff resource limitations.

The licensee's small-break LOCA emergency procedures and operator retraining were based on operator guidelines that were developed by the respective nuclear steam supply system vendors' owners groups and approved by the B&OTF. The owners groups and the dates of the B&OTF letters documenting its approval of the operator guidelines are as follows:

Owners Group	B&OTF Approval Letter Date(s)
W	November 5, 1979 December 6, 1979 December 27, 1979
C-E	November 14, 1979 December 26, 1979
GE	October 26, 1979 November 28, 1979

As set forth on Page 5 of Enclosure 6 to Darrell G. Eishehut's letters to all operating plants dated September 13, 1979, the small-break LOCA emergency procedures and operator retraining were to have been completed by December 31, 1979.

In conjunction with the aforementioned plant audits, a number of NRC representatives also visited Virginia Electric and Power Company's (VEPCO's) Surry Plant Simulator at Surry, Virginia; C-E's PWR Simulator at Windsor, Connecticut; and GE's BWR Simulator at Morris, Illinois. The purpose of the simulator visits was to observe the responses of the simulators to a small-break LOCA.

6.2 Plant Audits

The following matters were considered at each of the plant audits:

Emergency Procedures

The licensees' small-break LOCA emergency procedures were compared to the approved operator guidelines. The clarity of the procedures in terms of individual operator actions and cautions, and the flow of the procedures with respect to the timely initiation of operator actions, were considered.

Operator Retraining

The retraining that the operators received with respect to the small-break LOCA emergency procedures was reviewed. Informal training, formal classroom study and walk-throughs of the emergency procedures with their shift supervisors or training coordinators were considered.

Operator Awareness of the Emergency Procedures and their Bases

The operators' understanding of small-break LOCAs, the differences between small-break LOCAs and other depressurization events and the bases for the approved guidelines, and their familiarity with the small-break LOCA emergency procedures were reviewed. The effectiveness with which the emergency procedures can be carried out was considered.

Systems Considerations

Systems-related aspects of the emergency procedures were reviewed to assure that the necessary operator actions can be performed.

6.3 Findings

The most significant findings resulting from the plant audits are summarized below:

6.3.1 W-Plant Audit

Emergency Procedure

The licensee had restructured its emergency procedures to follow the approved W guidelines, i.e., a separate procedure for the operators to respond to a safety injection actuation regardless of the cause and a second emergency procedure for a LOCA. The procedures followed the guidelines reasonably well in both the immediate and subsequent actions. Several discrepancies and omissions of the procedures as compared with the guidelines were discussed with the licensee. The licensee indicated that the appropriate changes will be made. This matter will be followed up by IE.

Members of the Salem 1 plant staff used the revised procedures at the Zion simulator and found that one of the high pressure injection termination criterion, the steam generator narrow range level indication, resulted in lifting the power-operated relief valve and/or safety valves. This matter had been brought to the attention of the W Owners Group and, as discussed in Section 6.5.1 of this report, was subsequently resolved.

Operator Retraining

Since the revised emergency procedures had not yet been approved at the time of the audit, retraining of the operators had not yet begun. The licensee stated that this training would be completed prior to December 31, 1979.

Operator Awareness of the Emergency Procedures and Their Bases

Two reactor operators and two senior reactor operators were interviewed to determine the operators' knowledge of the small-break LOCA phenomenon, the bases for the procedure changes, and the TMI-2 accident sequence and its resulting impact on the Salem 1 plant. As a result of the interview, it was determined that the preliminary training of the licensee's personnel in the above areas was adequate. Several deficiencies were discussed with the facility management for incorporation in the training program. These deficiencies included:

(1) Thermodynamics and heat transfer (three of the four personnel were not able to adequately explain the theory and practical application of saturated versus subcooled or superheated water).

(2) The TMI-2 conditions and accident sequence (personnel exhibited incomplete or incorrect knowledge of the reason for pressurizer level increase while primary system pressure was decreasing).

Systems Considerations

For the most part, the instrumentation and controls associated with systems required to mitigate the effects of small-break LOCAs were located such that the necessary operator actions can be performed without undue difficulty.

6.3.2 C-E Plant Audit

Emergency Procedure

The licensee's small-break LOCA procedure contained most of the elements required by the approved generic small-break LOCA guidelines which were prepared by C-E for the C-E Owners Group. The NRC staff identified certain discrepancies that existed between the new procedure and the C-E guidelines as well as those between the new procedure and a related procedure and a licensing requirement.

The more salient recommendations made to the licensee regarding the procedure are as follows:

(1) The "Symptoms" section of the procedure included the terms "rapidly changing pressurizer level" and "rapidly decreasing pressurizer pressure." Since, for a small-break LOCA, these parameters would not change rapidly, it was recommended that the word "rapidly" be deleted from these terms.

(2) Since many of the identified symptoms may appear singularly or in conjunction with other symptoms during certain small-break LOCAs, it was recommended that the procedure should make it clear to the operator that the symptoms may not all appear simultaneously. In brief, each symptom statement should be considered to be an "or" statement.

(3) The "Immediate Action" section of the LOCA procedure includes a statement to the effect that the "Emergency Shutdown" procedure should be carried out. In reality, only the "Immediate Action" items of said procedure need be implemented; therefore, it was recommended that such a notation be made.

(4) The "Immediate Action" section also included a requirement to monitor the pressurizer pressure. Since this information would be of marginal value in the short time period associated with immediate action items, it was recommended that this action be deleted from the "Immediate Action" section.

(5) The procedure did not include a statement to the effect that high pressure safety injection should be initiated manually if it failed to be automatically initiated; therefore, it was recommended that such an action be included in the "Immediate Action" section.

(6) It was recommended that the "Subsequent Actions" sections be restructured so as to give a higher priority to the isolation of leak paths.

(7) It was recommended that a caution statement to the effect that pressurizer level, under the dynamic conditions associated with a small-break LOCA or other depressurizing event, may not be a reliable indicator of core level.

In addition to the above recommendations, a specific conflict regarding feedwater flow to the steam generators was discussed. On the one hand, the new LOCA procedure, in keeping with the bases stipulated in the approved guidelines, states that a minimum flow of 300 gallons per minute of auxiliary feedwater flow should be delivered to each steam generator. On the other hand, the existing "Emergency Shutdown" procedure, in conformance with a licensing restriction related to waterhammer, restricts the auxiliary feedwater flow to each steam generator to 168 gallons per minute under certain prescribed conditions.

Operator Retraining

At the time of the audit, the operator retraining program for the small-break LOCA emergency procedures had not been initiated. We were informed, however, that the retraining program would include a four-hour lecture session as well as a procedural walk-through in the control room. We were further informed that this training would be given to each licensed operator prior to December 31, 1979. Subject to fulfulling this commitment, we consider this to be adequate training for the small-break LOCA procedure.

Operator Awareness of the Emergency Procedure and Its Bases

Two senior reactor operators and one reactor operator were interviewed to determine the degree to which they understood the small-break LOCA procedure and its bases. As a result of these interviews, the NRC staff determined that the operators' knowledge of the procedures and their bases were adequate. However, the following general weaknesses were revealed which may reflect certain deficiencies in past operator training programs:

(1) The operators were confused on what a saturation curve represents. For example, they did not know how the plant would respond on the saturation curve to a stuck-open power-operated relief valve. In addition, they were unsure of the reasons for subcooled, saturated, and superheated plant conditions.

(2) The operators did not have a clear understanding of the reasons for tripping the reactor coolant pumps subsequent to the automatic initiation of high pressure safety injection due to low reactor coolant system pressure.

(3) The operators did not have a complete understanding of the relation between heat removal capability, natural circulation flow, and the temperature differential associated with natural circulation. Further, they did not understand the heat transfer relations across the steam generator. This matter is discussed further in Section 8.5 of this report.

Systems Considerations

For the most part, the instrumentation and controls associated with systems required to mitigate the effects of small-break LOCAs were located such that the necessary operator actions can be performed without undue difficulty. In brief, all the required actions can be performed within the control room.

At the conclusion of the plant audit, the licensee's representatives were asked if the efforts involved in developing the emergency procedure and the retraining of operators have enhanced the capabilities of the plant and the operators to cope with small-break LOCAs. The answers given by the licensee's representatives were generally affirmative; however, they had reservations regarding the accelerated program associated with developing and implementing the new procedures. The affirmative aspects of their answers included the benefit derived from the collective contributions of the Owners Group in developing the guidelines and the increased attention given by the Owners Group and the licensees' management in developing the procedures and the specialized operator retraining.

6.3.3 <u>GE Plant Audits</u>

<u>Emergency Procedures</u>

For the most part, each of the licensee's emergency procedures reflected all of the essential elements of the approved operator guidelines. In fact, at one plant, the emergency procedures were little more than the guidelines themselves. Consequently, they differed significantly in both format and content from the plant's other emergency procedures. Several of the licensees' emergency procedures appeared to be somewhat too detailed, especially in the automatic actions, and immediate operator actions sections.

In several instances, caution statements were placed in the immediate operator actions sections of the emergency procedures. It was generally felt, however, that this practice should be either minimized or eliminated. There was considerable discussion concerning the most appropriate sections of the emergency procedures to place the caution statements. One licensee expressed concern that the placement of caution statements in the discussion section of the emergency procedures could result in their being overlooked. There appeared to be no consensus, however, concerning the best placement of the caution statements in the emergency procedures. In one plant's emergency procedures, the caution statements concerning the effects of drywell temperature on reactor vessel level measurement was omitted. The licensee maintained that this matter can be addressed best during operator retraining.

There was considerable discussion concerning whether certain operator actions, such as verification of the automatic transfer of high pressure coolant injection system suction from the condensate storage tank to the suppression chamber and verification of automatic containment isolation by independent means, should be placed in the immediate operator actions section or the subsequent operator actions section of the emergency procedures. Again, there appeared to be no consensus concerning the best placement of such operator actions in the emergency procedures.

Finally, one licensee expressed concern that the NRC was becoming too prescriptive in the writing of emergency procedures.

The emergency procedures had been approved for use at all three plants but had actually been implemented at only one plant at the time of the audits. At the plant at which the emergency procedures had been implemented, however, not all the operators had completed their retraining or, for that matter, had even been required to read the emergency procedures. For this plant, the existing LOCA emergency procedures were revised to incorporate the essential elements of the approved guidelines, hence, no separate small-break LOCA emergency procedures existed per se. The licensee maintained that the revisions to its existing LOCA emergency procedures were minor and, therefore, the revised emergency procedures could be implemented prior to the completion of operator retraining. Nevertheless, the potential existed for an operator to be called upon to utilize an emergency procedure that he had never seen before.

Operator Retraining

At the time of the audits, none of the licensees' operators had completed all of their retraining associated with the small-break LOCA emergency procedures. Two of the licensees had instructed their operators to review the emergency procedures on their own, however, only approximately one-half of these operators had completed their reviews at the time of the audits. Two of the licensees planned formal classroom training sessions for their operators (the other licensee planned either a brief formal classroom session or walk-throughs of the emergency procedures with their shift supervisors or training coordinators), however, only a few of these operators had attended the training sessions at the time of the audits. Two of the licensees planned walk-throughs of the emergency procedures for their operators with their shift supervisors or training coordinators. The other licensee did not plan such walk-throughs, on the basis of the similarity of its small-break LOCA emergency procedures to its present LOCA procedures. Only one of the licensees indicated that it intended to utilize a simulator as part of its operator retraining.

Operator Awareness of the Emergency Procedures and Their Bases

The operators of two of the licensees exhibited a reasonably good understanding of the emergency procedures and their bases. These operators also appeared to be capable of effectively carrying out the emergency procedures. However, the operators of the other licensee exhibited a less-than-desirable understanding of the emergency procedures and their bases. The capabilities of these operators to effectively carry out the emergency procedures also appeared to be less than desirable.

Nearly all of the operators exhibited a less-than-desirable understanding of the reactor vessel level instrumentation. Notably deficient was the operators' understanding of temperature effects on the vessel level instrumentation, how temperature compensation is achieved and the vessel locations from which the levels are being measured. A few of the operators did not know such things as the means by which safety/relief valve position indication is derived, the location of the isolation condenser isolation indicators and the purpose of the isolation condenser bypass controls. In addition, substantially different answers were received on how to verify that containment isolation had occurred.

While describing how the emergency procedures would be carried out in the control rooms, a significant number of the operators overlooked several vital operator actions, such as the verification of reactor scram. For those cases in which one emergency procedure, such as the small-break LOCA procedure, refers to another procedure, such as the reactor scram procedure, it was not clear to the operators which procedure takes precedence; e.g., does the operator complete the referenced procedure before proceeding on to the next action statement of the parent procedure?

Systems Considerations

For the most part, the instrumentation and controls associated with those systems required to mitigate the consequences of small-break LOCAs were such that the necessary operator actions can be performed without undue difficulty. With the exception of the instrumentation required to verify Group 2 containment isolation of a limited number of small lines, all instrumentation and controls were located in the control rooms on cabinet panels facing and in immediate proximity to the operators. The instrumentation required to verify Group 2 containment isolation of a limited number of small lines was also located in the control rooms but on cabinet panels behind the main control complex.

Two licensees' operators pointed out that the separation of the instrumentation and controls necessary to mitigate the consequences of small-break LOCAs contributed some difficulty to the operators in performing their necessary actions, especially in those situations in which only a single operator is available. An example of this is the separation of the reactor vessel level instrumentation from the controls used to manually actuate the automatic depressurization system. The availability of at least two operators, as was the case in one of the plants, substantially compensated for this separation.

One licensee's operator suggested that the addition of a timer which could be used in conjunction with the manual initiation of the automatic depressurization system would be helpful. Another licensee's operator, noting that the zero indications of the various reactor vessel level instrumentation often refer to different reactor vessel levels, suggested that the potential for operator error would be substantially reduced if all the reactor vessel level instrumentation were referenced to the same reactor vessel level.

At the conclusion of each of the plant audits, each licensee was asked if, in its opinion, the efforts involved in developing the emergency procedures and the retraining of its operators have enhanced the capabilities of its plant and its operators to cope with small break-LOCAs and why. The answer given by each licensee was the affirmative. The reasons given included the collective contributions of a larger-than-normal number of individuals representing various interests in the development and approval of the operator guidelines, the increased attention given by the Owners Group and the licensee in the development of the emergency procedures, and the additonal specialized operator retraining.

6.4 Simulator Visits

6.4.1 Surry Plant Simulator

The Surry Plant Simulator has been in operation for about two years. When it was initially used for training, the operators complained that the dynamic response in the pressurizer (level/pressure) was not consistent with their experience during pressurizer refill events - namely, their experience indicated that pressure recovery lagged level recovery, significantly. The simulator was modified in about September 1978 to reflect the operating experience at Surry 1.

As a result of the aforementioned modification, the Surry Plant Simulator shows a much higher level recovery than predicted by WCAP-9600 for small-break LOCA events. This increased pressurizer level produces a much more rapid primary system pressure increase as the residual compressible volume is greatly decreased. The licensee's training personnel felt that there would be inadequate time for the operator to terminate high pressure injection before the relief and safety valves would be challenged. To increase the available reaction time, the licensee was instructing the operators to reduce the high pressure injection flow to one pump (we understand that W did not concur in this alteration to the guideline) if there was 50 degrees Fahrenheit subcooling in the primary system and an increasing level in the pressurizer. The staff informed the licensee's training personnel that such actions should be included in the procedures and the actions justified by the licensee or the W Owners Group. Subsequently, the licensee's training personnel indicated that they would abide by the predictive results of WCAP-9600. We expect each utility to either use the approved guidelines or else justify need for exceptions. This policy will be reflected in our letters to licensees.

The Surry units have a five-minute delay interlock which precludes resetting the emergency core cooling system after it has been initiated. This interlock would prevent the operator from reestablishing chemical and volume control system operation if the high pressure injection termination criteria were satisfied in less than five minutes (for non-LOCA events) and this would provide potential conditions for opening the power-operated relief valve. This situation is currently being reviewed by the Surry personnel and will be considered in our policy letter to VEPCO.

6.4.2 C-E PWR Simulator

The B&OTF representatives that participated in the audit of Millstone 2 also visited C-E's PWR Simulator to observe its response to a small-break LOCA. Since the ability of the simulator to adequately represent a small-break LOCA is somewhat limited, certain modifications had to be made. The actual simulation was accomplished by simulating the rupture of several steam generator tubes. Hence, the break could be classified as a "large small-break." Nevertheless, the systems' response was observed, and since the simulation represented the case where offsite power was <u>not</u> lost, system parameters were watched closely, and when it was determined that a reactor trip had taken place and that high pressure safety injection had been initiated by low reactor coolant system pressure, the reactor coolant pumps were tripped. In brief, although this break does not represent the "worst-case" break for C-E designed reactors, the system parameters could be readily monitored during the course of the event.

6.4.3 GE BWR Simulator

A number of those NRC representatives who participated in the audit of the Dresden 2/3 plant visited GE's BWR Simulator to observe its responses to a small-break LOCA. The particular scenario represented involved a small break in a main steam line inside containment combined with a total loss of offsite power.

The capability of the simulator to adequately represent a small-break LOCA is presently limited, however. In the above scenario, the small break had to be represented by opening a safety/relief valve. A fully-opened safety/relief valve, however, represents a somewhat larger break area than is of particular interest in the study of small-break LOCAs. In addition, since the safety/relief valves discharge to the suppression chamber, the drywell pressure and temperature do not increase as they would in the event of a small-break LOCA inside containment. The increase in drywell pressure and temperature had to be initiated by turning off the drywell coolers. The GE representatives indicated, however, that in the near future, they expect to upgrade the capability of the simulator to more adequately represent small-break LOCAs. Additional features presently under consideration include the capabilities to represent various small break sizes at a number of different locations.

6.5 Conclusions

6.5.1 W Plants

Following our audit at Salem 1 and visit to the Surry Plant Simulator, we discussed our findings with the W Owners Group and with W. On December 21, 1979, the W Owners Group submitted proposed modifications to the W generic small-break LOCA guidelines which changed the high pressure injection termination criteria to correct the overpressure problems highlighted during our audit visits. By letter dated December 27, 1979, we approved the proposed modifications to the W generic guidelines.

Based on our audits at Salem 1 and visit to the Surry Plant Simulator, we have requested IE to assist us in conducting a check of procedure implementation at the remaining W-designed facilities to determine whether our limited experience was representative of the situation at W-designed operating plants in general. Based on the results of this check, no further problems are apparent at this time.

On the bases of the findings resulting from the B&OTF audit of the Salem 1 plant, it was concluded that the licensee's emergency procedures and operator retraining associated with small-break LOCAs could be implemented by December 31, 1979, as required. It was further concluded that the licensee's emergency procedures and operator retraining associated with small-break LOCAs provide added assurance that the W operating plants and their operators can accommodate a small-break LOCA in an acceptable manner.

6.5.2 C-E Plants

On the bases of the findings resulting from the B&OTF audit of the Millstone 2 plant, it was concluded that the licensee's emergency procedure and operator retraining associated with small-break LOCAs could be implemented by December 31, 1979, as required. It was further concluded that the licensee's emergency procedure for small-break LOCAs, subject to its being modified per the NRC staff's recommendations, and operator retraining provide added assurance that C-E operating plants and their operators can mitigate the effects of a small-break LOCA in an acceptable manner.

6.5.3 GE Plants

On the bases of the findings resulting from the B&OTF audits of the Nine Mile Point 1, FitzPatrick, and Dresden 2/3 plants, it was concluded that the BWR plant licensees' emergency procedures and operator retraining associated with small-break LOCAs could be implemented by December 31, 1979, as required. It was further concluded that the licensees' emergency procedures and operator retraining associated with small-break LOCAs provide added assurance that the boiling water reactor plants and their operators could accommodate a small-break LOCA in an acceptable manner.

7.0 STATUS OF LOFT L3-1 TEST PROGRAM

7.1 Discussion of Test

The nuclear loss-of-coolant experiment, LOCE L3-1, was successfully conducted on November 20, 1979. This test simulated a single-ended shear break of a small (four-inch diameter equivalent) primary system cold leg pipe in a large PWR. The LOFT configuration is shown in Figure 7-1. The system is comprised of an intact, circulating loop and a broken, stagnant loop. The broken hot leg is an inactive representation of a steam generator and primary coolant pump. The break is modeled with an orifice in the cold leg section of the broken loop and is representative of a break in the pump discharge piping. Emergency core cooling injection is into the intact loop only. A chronology of events during the blowdown transient is listed in Table 7-1.

Pretest analysis of the experiment was requested from the PWR vendors, B&W, W, and C-E. Exxon, a manufacturer of reload fuel, was also requested to perform a pretest analysis. Results of the B&W, W, and C-E calculations were received following the test. In addition, pretest analyses were performed by INEL and LASL using methods being developed for the Office of Nuclear Reactor Regulation.

Prior to the test date, the staff held meetings with the PWR vendors and Exxon to discuss the analyses and to define the proposed models to be used. During these meetings, specific input parameters and changes required to the models previously submitted to the NRC were

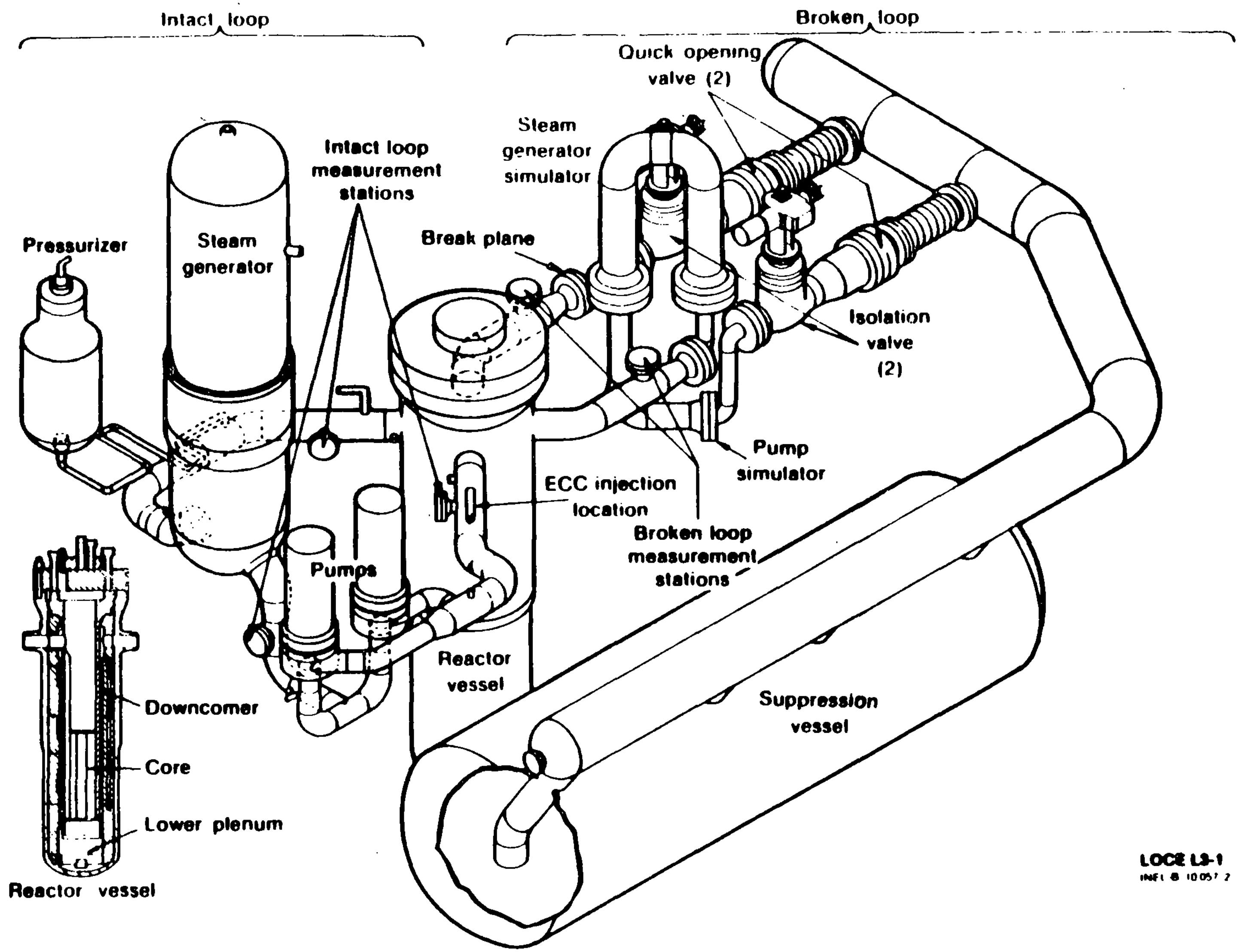


Figure 7-1 Axonometric projection of LOFT system.

TABLE 7-1

CHRONOLOGY OF EVENTS FOR LOFT L3-1

LOFT L3-1 Event	Time (seconds)
Reactor Scram	- 2.15
Control Rods Bottomed	- 0.97
QOBU's Opened	0.00
Trip Primary Coolant Pumps	0.40
HPIS Initiated	4.60
Subcooled Blowdown Completed	5.40
Pressurizer Empties	17.00
Pump Coastdown Completed	19.00
End Subcooled Break Flow	46.40
Steam Generator Feed Initiated	75.00
Accumulator Flow Initiated	633.60
Accumulator Nitrogen Enters Loop	1741.00
Steam Generator Feed Stopped	1875.00
Steam Generator Bleeding Initiated	3622.50
LPIS Initiated	4240.00
LOFT Test L3-1 Completed	4368.00

documented. This procedure was established to allow for the actual running of the analyses after the test date, if required, to compensate for the relatively short response time requested. The documentation is on file at the NRC.

The results of the pretest analyses are compared to the test data for the primary systems pressure in Figure 7-2. The break flow model and the multiplier used are identified on the figure. The Babcock and Wilcox model actually uses a 0.6 multiplier on Moody during the calculated two-phase portion of the analysis and then switches to a 0.9 multiplier on Moody when the calculated break flow quality goes to 1.0, steam flow. The Combustion Engineering analysis used the Henry-Fauske model for subcooled and two-phase flow, and the Murdock Burnam calculations for steam flow. The calculation was terminated at the time of calculated accumulator injection. Combustion Engineering is currently continuing the analysis and evaluating the results. The Westinghouse calculation was completed to 1500 seconds, and assumed the modified Zaloudek and Moody correlations with a multiplier of 1.0. The break flows are compared in Figure 7-3.*

The results of the test are currently under review at INEL. Part of this review procedure includes the identification of differences between the pretest analysis performed by INEL and the actual system performance. The following items have been identified:

*System break flow, while not measured directly, was calculated using the suppression tank liquid level and the limited TTF data.

Figure 7-2

LOFT L3-1 TEST DATA COMPARISONS

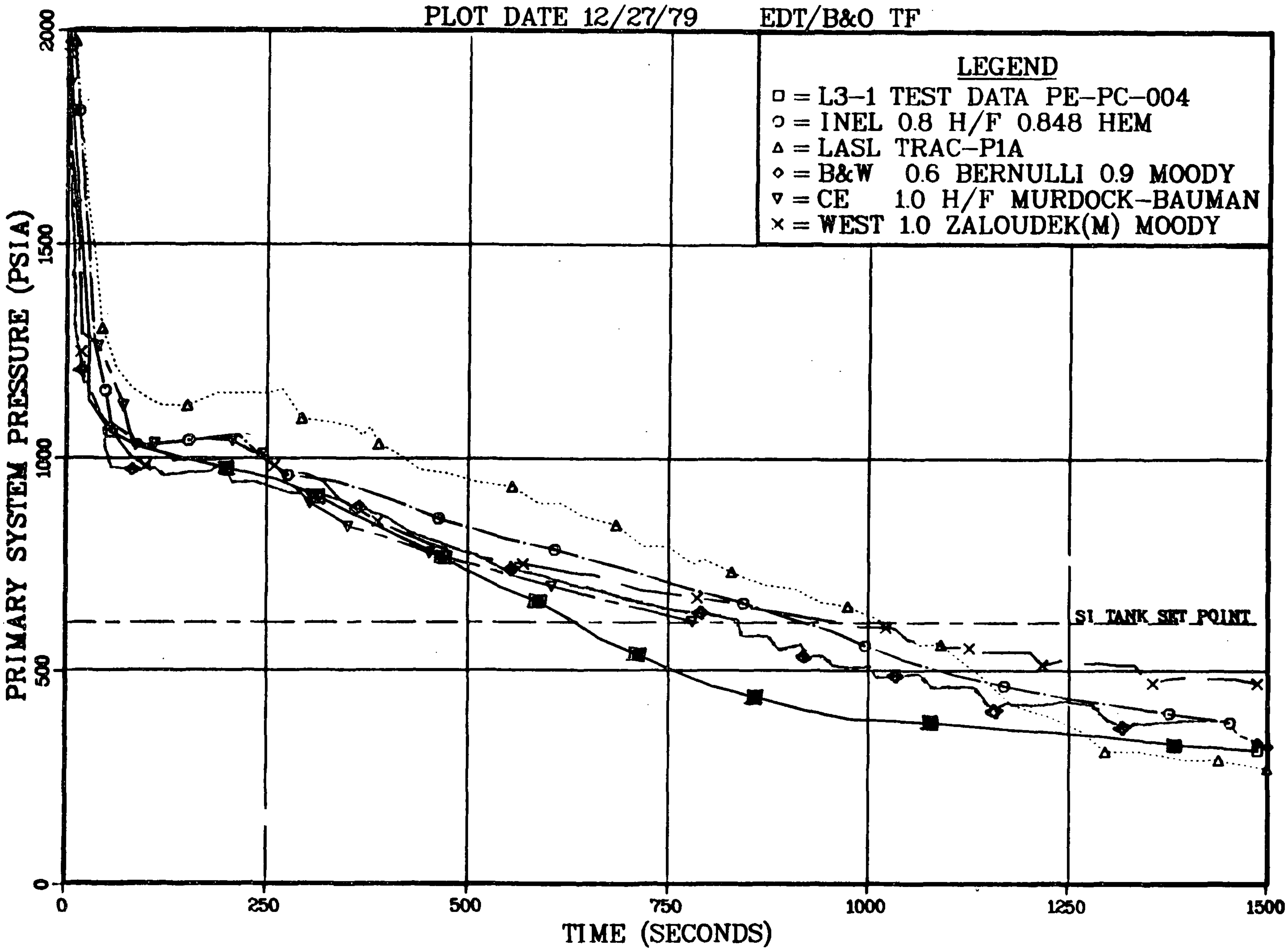
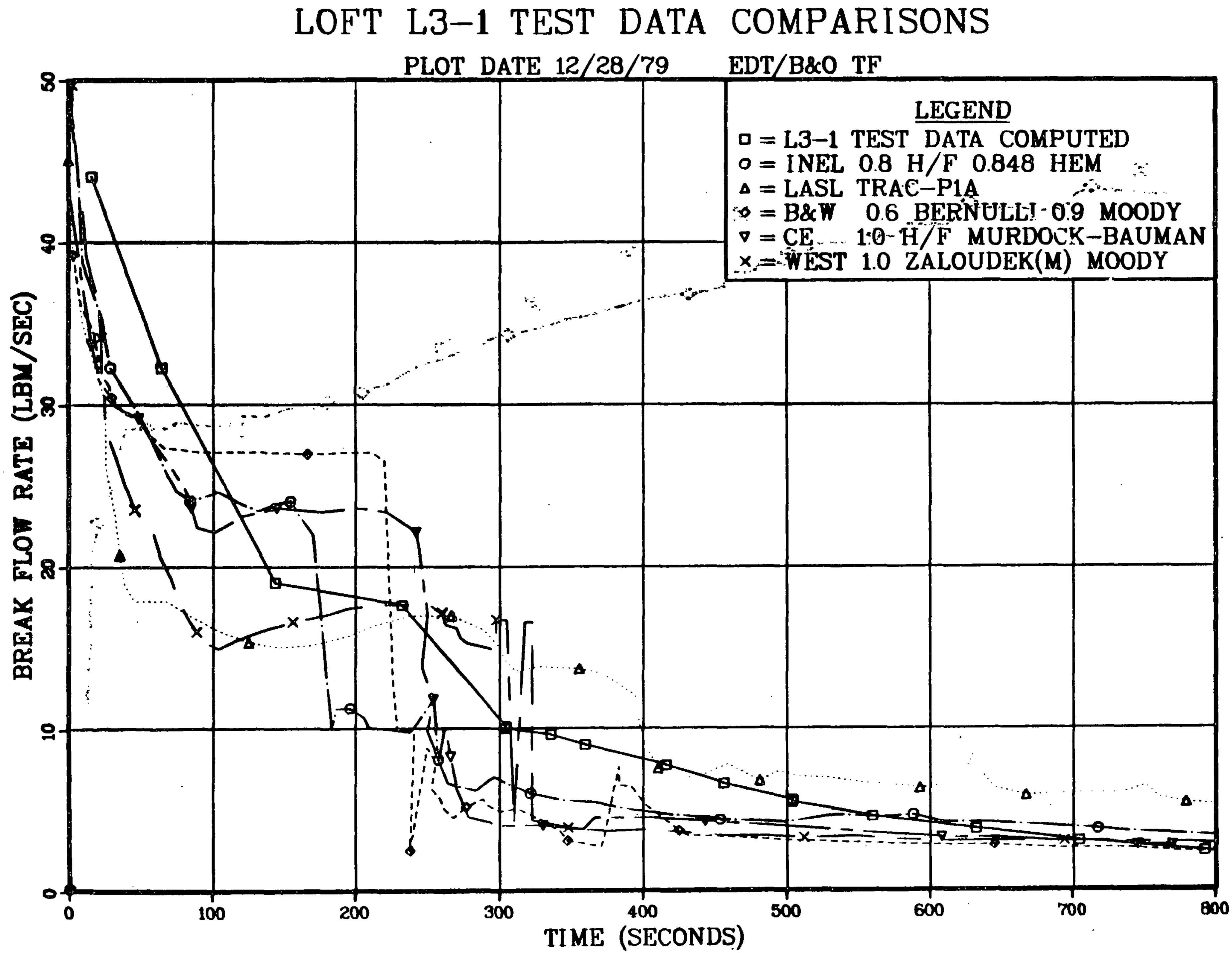


Figure 7-3


LOFT L3-1 TEST DATA COMPARISONS
PLOT DATE 12/28/79 EDT/B&O TF
LEGEND
□ = L3-1 TEST DATA COMPUTED
○ = INEL 0.8 H/F 0.848 HEM
△ = LASL TRAC-P1A
◇ = B&W 0.6 BERNULLI 0.9 MOODY
▽ = CE 1.0 H/F MURDOCK-BAUMAN
× = WEST 1.0 ZALOUDEK(M) MOODY
BREAK FLOW RATE (LBM/SEC)
TIME (SECONDS)
0
10
20
30
40
50
0
100
200
300
400
500
600
700
800

Accumulator Injection

Although the accumulator injection initially increased the depressurization rate, accumulator effects appear to have subsequently reduced the primary depressurization rate resulting in an elongated transient. This elongation of the transient is postulated to have been caused initially by increasing the water level above the break orifice and then by the accumulator nitrogen isothermally expanding and pressurizing upon injection into the system.

The accumulator nitrogen expansion was basically isothermal. Pressure and volume measurements initially and at the end of the accumulator injection indicate k of approximately 1.01 (where k is defined by pv^k = constant).

The pretest analysis performed by INEL used a k of 1.401 and, because RELAP4 cannot model nitrogen injection, the accumlator injection was terminated when the calculated liquid level indicated nitrogen injection would occur.

The use of the 1.401 value for k was an oversight in the analysis. It had been anticipated that the nitrogen expansion would be isothermal.

Intact Loop Behavior

The loop seal in the intact loop remained partially filled with liquid preventing flow in either direction. Preliminary evaluation of the flow data in the intact loop indicates no measureable flow beyond 60 seconds, whereas the pretest calculations indicated a small positive flow would be maintained in the intact loop.

This analysis indicated that the loop seal would clear, and provided a path for the steam generated in the inner vessel to reach the break.

Since the loop seal did not clear and the system did depressurize, it is likely that an alternate path existed which was not included in the analysis model. In the LOFT system, there are three significant leakage paths that might contribute to this effect. These are leakage through the pressurizer spray line mini-flow valve, leakage through the reflood assist bypass valve (these valves are located in the line connecting the break and loop cold leg and hot leg nozzles), and leakage through the gap between the downcomer and the upper plenum. There is also a leakage path from the upper to lower plenum through the filter region; however, this path was sealed since the core did not uncover. This path was included in the calculation but had no effect. The total leakage for these three paths* is estimated to be equivalent to between 5 and 15 percent of the steady-state intact loop flow rate. The distribution of the leakage among these three bypass paths is now known. Two of these paths, the bypass valves and the gaps, provide direct communication between the upper plenum and the break. NRC estimates indicate that a bypass area of 0.25 square inch is sufficient to prevent clearing of the loop seal.

*Similar paths exist in commercial PWRs.

The large, observed bypass flow is an indication that bypass areas larger than 0.25 square inch do exist in LOFT.

Analyses to determine the effect of bypass paths is continuing.

Steam Generator Heat Transfer

During the first 130 seconds of the transient, the steam generator heat transfer was primary-to-secondary. After this period, the heat transfer was secondary-to-primary. The inlet and outlet temperatures measured in the steam generator were substantially higher (100 to 120 degrees Fahrenheit) than the primary coolant saturation temperature. This indicated the steam generator primary side had voided and was filled with superheated steam. This temperature difference was substantially larger than the pretest calculated value. The observed difference is a consequence of the stagnant intact loop.

During the later portion of the test, an attempt was made to use a steam generator feed and bleed procedure to reduce the primary system pressure. The test was terminated before the secondary side temperature was less than the primary side temperature and the cooling was never established.

7.2 Summary and Recommendations

The experimental data obtained from the LOCE L3-1 test are still under review. The differences between the test data and the pretest analysis performed by INEL are being evaluated.

The failure of the loop seal to clear and the matter of steam generator heat transfer are under investigation.

A post-test analysis of the LOCE L3-1 test will be performed by INEL. This analysis will use the isothermal nitrogen expansion model; the initial condition will reflect the measured data at steady-state; bypass path will be represented in the calculations; and the break flow model will be modified to reflect the recently completed analyses of the blowdown orifice pretest program.

It is recommended that: (1) INEL establish the flow area associated with each of the three bypass paths; (2) the effect of the bypass path be evaluated on L3-1 as well as on future LOFT tests; and (3) an additional test be performed which arrives at similar conditions prior to the feed and bleed attempt, and that the test be carried out until cooling is observed to see how difficult it is to cool down the plant from this condition.

8.0 OTHER MATTERS

8.1 Introduction

During the B&OTF generic review of operating plants, a number of recommendations did not appear in the generic reports, i.e., NUREG-0565, NUREG-0611, NUREG-0626, and NUREG-0635, for one of the following reasons:

(1) The recommendation did not follow from findings and conclusions,

(2) The recommendation was not timely,

(3) The recommendation was beyond the B&OTF scope, or

(4) Insufficient information was available to develop a final recommendation.

8.2 Conservatisms in LOCA Analyses

It was recommended that the NRC staff's review of the conservatisms in the LOCA analyses be accelerated with a position on small-break LOCA analyses to be developed by June 1980. This recommendation was not included in the generic reports because it went beyond the B&OTF charter which was to assure that the analysis methods were acceptable for the development of improved small-break LOCA procedures and related operator training. In addition, this activity involves decisions on resource priorities and was recommended for inclusion in the NRC Action Plan for implementing the recommendations of the President's Commission and other studies of the TMI-2 accident.

8.3 Loss of All Feedwater in PWRs

It was recommended that the capacity of the power-operated relief valves be demonstrated to provide depressurization in the case of a complete loss of feedwater with due consideration of input uncertainties and calculational uncertainties. If such could not be demonstrated, it was recommended that continued operation of the affected plants should be conditioned to require the implementation of timely design changes.

This recommendation was not included in the generic reports on the basis that it needs more careful consideration under the Unresolved Safety Issues program, or the NRC Action Plan. In addition, the B&OTF is requiring substantial improvements in the reliability of auxiliary feed-water systems as discussed previously. In a related action, short-term Lessons Learned Item 2.1.2 requires performance verification by full scale prototypical testing for all relief and safety valves. Test conditions are to include two-phase slug flow and subcooled liquid flow calculated to occur for design basis transients and accidents.

8.4 Design Bases for Auxiliary Feedwater Systems

As a result of the B&OTF review of operating plant auxiliary feedwater systems, we concluded that the design bases and criteria used by licensees for establishing auxiliary feedwater system requirements for flow to the steam generator(s) to assure adequate removal of reactor decay heat are not well defined or documented.

The staff transmitted a request for information to each licensee (C-E and W plants) that received a letter with the auxiliary feedwater system recommendations resulting from the NRC staff's auxiliary feedwater system reliability reviews. A similar request will be sent to each B&W operating plant licensee in conjunction with the auxiliary feedwater system reliability review that is now in progress. These licensees were requested to reevaluate the auxiliary feedwater system flow requirements and pump capacities to assure that the design bases for these requirements are current and adequate with respect to the various plant transients and postulated accident conditions that each plant must be able to withstand safely. The licensees were requested to describe the analyses, assumptions, and acceptance criteria used for determining auxiliary feedwater system flow requirements for specified plant transients and accident conditions and to verify that the auxiliary feedwater pumps in the existing auxiliary feedwater systems satisfy these requirements.

In the responses received by B&OTF as of December 21, 1979, the licensees have not provided the requested auxiliary feedwater flow requirements information. These licensees indicated that the information may be evaluated on a generic basis by the respective owners, but no firm schedule commitment has been established.

The B&OTF considers that this information should be submitted in a more timely manner and that the licensees should be informed that the requested information should be submitted by February 28, 1980.

8.5 Audits of Licensed Operators

Audits of licensed operators and senior operators were conducted at seven PWR facilities during the six months following the TMI-2 accident. The audits consisted of an Operator Licensing Branch examiner or an IE inspector verbally quizzing the licensees in the following areas:

(1) TMI-2 accident,

(2) Small-break phenomenon, and

(3) Revised LOCA procedures.

The audits revealed that while many of the licensees understood the basics of thermodynamics, heat transfer, and fluid flow, additional training in these areas was warranted to increase their knowledge. Specifically, the topics where weaknesses were identified were saturation versus subcooled or superheated conditions, recognition of proper natural circulation flow, relationship of secondary system pressure and primary system temperature, and other related topics. Facility management was informed when such weaknesses were found. In the case of B&W plants, additional training was conducted, as applicable, by outside consulting organizations.

In the future, a separate category will be added to the NRC's operator and senior operator written examination containing questions regarding thermodynamics, heat transfer, fluid flow, and hydraulics. In the meantime, each licensed PWR facility should incorporate these topics in the lecture series of the requalification program. In addition, the annual facility requalification written examination should be expanded to include the additional categories.

The depth of knowledge in these areas should be at a first-year college level. Basic principles should be emphasized and then applied to practical situations. As an example, operators should be able to explain phase changes of water using pressure-temperature or temperature-volume diagrams but need not necessarily be familiar with Mollier diagrams. They should be able to use Reynold's Numbers in explaining the difference between laminar and turbulent flow. They should also be familiar with the three modes of heat transfer--conduction, convection, and radiation--and explain where they apply in the plant. As a general rule, their knowledge level in these areas should extend up to the point necessary to explain concepts consistent with the ability of operators to understand calculus.

The NRC is requiring that each licensed operator and senior operator receive formal lectures on these topics within the next three months. A minimum of 15 hours of lectures is recommended. The changes in the lecture series and annual written requalification examination will be audited as part of the normal inspection function of the requalification program.

8.6 Recommendations Resulting from Audits of BWR Plants

As a result of the B&OTF audits of the emergency procedures and operator retraining associated with small-break LOCAs at the Nine Mile Point 1, FitzPatrick, and Dresden 2/3 plants and the visit to GE's BWR Simulator at Morris, Illinois, several areas were identified in which the need for improvement is indicated. It is recognized that the need for improvement in at least some of these areas existed prior to the licensees' efforts related to small-break LOCAs or as a result of the audits being conducted several weeks before the date that the efforts were required to be completed.

The areas in which the need for improvement is indicated and the recommendations for improvement are as follows:

Emergency Procedures

On the bases of their reflection of the essential elements of the approved operator guidelines and their status of implementation, it was concluded that the licensees' small-break LOCA emergency procedures could be implemented by December 31, 1979, as required. However, on the bases of the potential problems associated with the use and placement of caution statements, placement of operator action statements, and implementation practices associated with the emergency procedures, it is recommended that the NRC, in collaboration with the nuclear industry, develop guidance, perhaps in the form of a regulatory guide, on the format, content, and implementation of emergency procedures.

Operator Retraining

On the basis of their status of completion, it was concluded that the licensees' operator retraining associated with small-break LOCAs could be completed by December 31, 1979, as required. However, on the bases of the substantial disparity in the emphasis the licensees placed on the operator retraining associated with the small-break LOCA emergency procedures, it is recommended that the NRC, in collaboration with the nuclear industry, develop guidance, perhaps in the form of a regulatory guide, on the extent of operator retraining appropriate to the implementation of new or revised emergency operating procedures.

Operator Awareness of the Emergency Procedures and Their Bases

On the bases of a significant number of operators' less-than-desirable understanding of the emergency procedures and, especially, their bases, it was concluded that the licensees' operator retraining is less than desirable. Accordingly, it is recommended that the licensees reassess their operator retraining in order to ensure that their operators thoroughly understand the emergency procedures and their bases.

8.7 Loss of All Alternating Current Power

It was recommended that plants be required to withstand a station blackout of a specified duration (two hours, for example). This recommendation was not included because the matter has already been assigned to the Unresolved Safety Issues Program as Task A-44, "Station Blackout." Little progress has been achieved toward resolution of the concern although it was designated as a high priority generic concern more than a year ago. We recommend that resolution of this concern be expedited.

Acceptance criteria should be developed (e.g., a plant should be capable of withstanding a total loss of alternating current power for at least two hours) and all light water reactors should show compliance with the criteria within a reasonable time frame, e.g., 18 months. We recognized that this may require the addition of emergency power sources at some plants.

8.8 Technical Specification Limit on Cumulative Outage Times

It was recommended that the Technical Specifications for plants be modified to include a requirement to limit the cumulative outage times of emergency core cooling systems. The cumulative outage times might differ from system to system depending upon their relative importance. This recommendation was not included in the B&OTF reports for the various plant types due to time constraints. However, this recommendation has been directed for action to the NRR Division of Operating Reactors.

PORV /SCRAM set points

NRC FORM 335 (7-77)

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER *(Assigned by DDC)*	NUREG-0645, Vol. 1
2. *(Leave blank)*	
3. RECIPIENT'S ACCESSION NO.	
4. TITLE AND SUBTITLE *(Add Volume No., if appropriate)*	Report of the Bulletins & Orders Task Force of the Office of Nuclear Reactor Regulation
5. DATE REPORT COMPLETED	MONTH: January; YEAR: 1980
DATE REPORT ISSUED	MONTH: January; YEAR: 1980
6. *(Leave blank)*	
7. AUTHOR(S)	
8. *(Leave blank)*	
9. PERFORMING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*	U.S. Nuclear Regulatory Commission Office of Nuclear Reactor Regulation Washington, D.C. 20555
10. PROJECT/TASK/WORK UNIT NO.	
11. CONTRACT NO.	
12. SPONSORING ORGANIZATION NAME AND MAILING ADDRESS *(Include Zip Code)*	
13. TYPE OF REPORT	
PERIOD COVERED *(Inclusive dates)*	
14. *(Leave blank)*	
15. SUPPLEMENTARY NOTES	

16. ABSTRACT *(200 words or less)*

The results of the Bulletins & Orders review of the Office of Inspection and Enforcement bulletins, Commission Orders, and the Office of Nuclear Reactor Regulation generic evaluation of feedwater transients, small-break loss-of-coolant accidents, and other Three Mile Island Unit 2 related events in operating plants to confirm or establish the bases for their continued safe operation are summarized. As a result of its review, the Bulletins & Orders Task Force has concluded that (1) the continued operation of the operating plants is acceptable provided that certain actions related to the plants' designs and operation, and training of operators are implemented consistent with the recommended implementation schedules, and (2) the actions taken by the licensees with operating plants in response to the IE bulletins (including the actions specified in NUREG-0623, "Generic Assessment of Delayed Reactor Coolant Pump Trip During Small Break Loss-of-Coolant Accidents in Pressurized Water Reactors") provide added assurance for the protection of the health and safety of the public. In addition, the Bulletins & Orders Task Force has independently confirmed the safety significance of those related short-term and long-term actions recommended by other Office of Nuclear Reactor Regulation task forces.

17. KEY WORDS AND DOCUMENT ANALYSIS	
17a. DESCRIPTORS	
17b. IDENTIFIERS/OPEN-ENDED TERMS	
18. AVAILABILITY STATEMENT	Unlimited
19. SECURITY CLASS *(This report)*	Unclassified
20. SECURITY CLASS *(This page)*	
21. NO. OF PAGES	
22. PRICE	$

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

POSTAGE AND FEES PAID
U.S. NUCLEAR REGULATORY
COMMISSION

U.S. MAIL