NUREG-0667

Transient Response of Babcock & Wilcox-Designed Reactors

Office of Nuclear Reactor Regulation

U.S. Nuclear Regulatory Commission

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Manuscript Completed: April 1980
Date Published: May 1980

B&W Reactor Transient Response Task Force
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

ABSTRACT

On February 26, 1980, the Crystal River Unit No. 3 Nuclear Generating Plant, designed by the Babcock & Wilcox Company (B&W), experienced an incident involving a malfunction in an instrumentation and control system power supply. This resulted in a reactor and turbine trip; the opening of the pressurizer power-operated relief valve, spray valve and a Code safety valve; decreased feedwater flow; actuation of the engineered safety features systems; and a discharge of approximately 40,000 gallons of primary coolant into the containment building.

Faced with the Crystal River Unit 3 incident and the apparently high frequency of such near similar types of transients in other B&W-designed plants, a special Task Force (i.e., B&W Reactor Transient Response Task Force) was established within the Office of Nuclear Reactor Regulation to provide an assessment of the apparent sensitivity of the B&W-designed plants to such transients and the consequences of malfunctions and failures of the integrated control system and non-nuclear instrumentation. This report provides an assessment of these issues.

Prior to its finalization, a draft of this report was informally commented upon by the NRC staff, Babcock and Wilcox, the B&W Licensees, NSAC, and both the ACRS B&W Subcommittee and ACRS full committee. Comments and suggestions received have been reflected appropriately in the report.

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TRANSIENT RESPONSE OF BABCOCK & WILCOX DESIGNED REACTORS

1. INTRODUCTION

On February 26, 1980, the Crystal River Unit 3 Nuclear Generating Plant (CR-3) experienced an incident involving an electrical malfunction in an instrumentation and control system. This resulted in: a reactor and turbine trip; the opening of the pressurizer power-operated relief valve (PORV), the pressurizer spray valve, and a code safety valve; decreased feedwater flow to the steam generators; actuation of the engineered safety features (ESF) systems; and a discharge of approximately 40,000 gallons of primary coolant into the containment building. The reactor was designed by the Babcock & Wilcox Company (B&W). Certain of the events were similar in some respects to those that took place at Three Mile Island Unit 2 (TMI-2) approximately one year ago and to those that have occurred at other B&W plants in recent years.

Faced with the CR-3 incident and the apparently high frequency of near similar types of transients in other B&W plants, a special task force (i.e., B&W Reactor Transient Response Task Force) was established by the Director, Office of Nuclear Reactor Regulation (NRR), on March 12, 1980, to assess the generic aspects of operating experiences of the B&W plants. The time estimate to complete this effort was established to be approximately two weeks. This assessment would include consideration of the apparent sensitivity of the B&W plants to transients involving overcooling and undercooling conditions, small break loss-of-coolant accidents (LOCAs), and the consequences of malfunctions and failures of the integrated control system (ICS) and non-nuclear instrumentation (NNI). The study would be made in conjunction with all of the actions already taken or proposed in response to the TMI-2 accident.

The sensitivity of B&W reactor designs to transients and accidents has been discussed previously with the Commission. Of particular concern is the plant recovery from certain anticipated transients that can lead to frequent challenges to the engineered safety features. Some preliminary findings and conclusions were presented in the April 25, 1979, NRR Status Report on Feedwater Transients in B&W Plants (Ref. 1) which served, in part, as the basis for the confirmatory shutdown orders issued in May 1979 (Ref. 2) to all B&W operating plant licensees. A more complete exposition on this subject is found in recent staff reports NUREG-0560 (Ref. 3), NUREG-0565 (Ref. 4), and NUREG-0645 (Ref. 5). Some of the design changes already accomplished have helped to reduce the sensitivity of the B&W design to certain transients (e.g., the addition of anticipatory reactor trips for loss of feedwater and turbine trip). However, the operating experience obtained recently from CR-3 requires further consideration of the NRC position on the B&W plants. A particular area that the Task Force has considered in the B&W design deals with an apparent lack of sufficient design interface requirements between the nuclear steam supply system (NSSS) and the balance-of-plant (BOP); that is, interfaces between the safety requirements for the auxiliary feedwater system and the operating requirements for the once-through steam generators (OTSG) and the ICS.

As stated in a memorandum dated October 25, 1979, from the Director, NRR, to the Commissioners (Ref. 6), the staff is continuing its review of the 'ensitivity issue and has initiated with the Office of Nuclear Regulatory Research a detailed study of risk assessment, on a relative basis, of the B&W design. The status of this work is discussed in detail in Section 6 of this report.

Because of the short time period the Task Force was given to complete its assignment, the thrust of this assessment deals with the broader aspects of the sensitivity question. Detailed analyses were not possible for this effort. The present requirements being imposed on B&W plants are those developed mainly by the Lessons Learned Task Force and the Bulletins and Orders Task Force within NRR. These requirements are tabulated in Appendix A of this report. Further actions are being considered in an overall, integrated NRC Action Plan (Ref. 7) now under development. The plan will incorporate the recommendations of the President's Commission on the Accident at Three Mile Island as well as those of the NRC's Special Inquiry Group, the Advisory Committee on Reactor Safeguards (ACRS), and other investigatory groups within the NRC and industry.

Other efforts dealing with the question of sensitivity of the B&W plants to transient response and failures of the instrumentation and control systems on a generic basis include the review of the CR-3 event and the responses to IE Bulletin 79-27 (Ref. 8). IE Bulletin 79-27 deals with an incident that occurred at Oconee Unit 3 on November 10, 1979, wherein a loss of NNI resulted in a partial loss of indication in the control room. In both of these matters, the staff is reviewing responses from the B&W licensees as well as the joint report of the Nuclear Safety Analysis Center and the Institute of Nuclear Power Operations (INPO/NSAC) dated March 11, 1980 (Ref. 9) regarding the CR-3 incident. Staff reports will be issued following completion of these reviews.

By letter dated March 6, 1980 (Ref. 10), the B&W operating plant licensees were asked a number of questions regarding the effect of the CR-3 event and actions being taken against consequential failure modes in the NNI for their plants. In addition, Florida Power Corporation (FPC, licensee for CR-3) was requested to specifically discuss the impact of the TMI-2 Lessons Learned and Bulletins and Orders modifications on its facility with respect to the February 26, 1980, event. FPC provided its response to this request by letter dated March 12, 1980 (Ref. 11). The Task Force has considered the initial response of the licensee and is in general agreement with its conclusion that some benefit to plant operations did result from the lessons learned actions. This appears to be especially true with regard to operator actions during the plant recovery sequence. Although a detailed review of the CR-3 is beyond the scope of this Task Force, it can be concluded that no known effects resulting from the event would lead the staff to suspend implementation of the Lessons Learned or Bulletins and Orders requirements. However, further consideration of variations in the CR-3 event sequence is warranted to perhaps better assess the overall aspects of the lessons learned actions.

Even following the implementation of all of the required and intended actions on the B&W plants, there will be no guarantee that transients and accidents, similar to those that occurred at the TMI-2 and Crystal River 3 facilities, can be completely prevented. However, the Task Force believes that the occurrence of such events would be less frequent and of less consequence. Thus, the effect of the actions would be to provide an increased margin of assurance to the health and safety of the public. Although the efforts of the task force tend toward consequence mitigation, it should not be implied that efforts should not continue to improve plant performance to reduce the frequency of transients (e.g., prevention of the initiating events such as control system malfunctions).

This Task Force has attempted to take a cumulative view of all of the actions and modifications that have been imposed upon the B&W operating plants as a result of the accident at TMI-2 and to couple these requirements with recent operating experience. One purpose of this broad view is to see if the modifications implemented or scheduled to be implemented go far enough to assure that the B&W operating plants can respond in an acceptable manner to transient and accident situations. As a result of this review, the Task Force has developed a number of recommendations that have been arranged into four main areas: (1) auxiliary (or emergency) feedwater systems, (2) instrumentation and control, (3) design and operational matters, and (4) general areas of improvement to enhance the safety of B&W-designed reactors. A summary of conclusions and recommendations is provided in Section 2 of this report. The bases for the conclusions and recommendations are found in the applicable sections of the report.

The Task Force has provided an estimate of the effectiveness in risk reduction potential associated with each of the recommendations along with preliminary resource estimates. A discussion of risk reduction potential is provided in Section 7 of the report.

It must be recognized that even full implementation of all of the recommendations contained in this report may not completely resolve the sensitivity of the B&W plants as compared to plants designed by other vendors. As a long-term solution to this problem, the Task Force believes that acceptance criteria for plant performance during anticipated transients, applicable to all plant designs, should be developed. Conformance to such criteria would assure a uniform and acceptable risk to the public from anticipated transients. The criteria should cover such areas as heat sink availability, transient initiators, operator information and overall plant response.

The Task Force members are listed as follows:

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2. SUMMARY AND RECOMMENDATIONS

2.1 Summary

This report contains the findings of the B&W Reactor Transient Response Task Force. The Task Force was established to assess the generic aspects of recent operating events that have occurred on the B&W-designed reactor plants, and, in particular, the Crystal River 3 (CR-3) event of February 26, 1980.

The general findings of the Task Force are in agreement with previous staff conclusions that the B&W-designed 177-FA plants show some unique levels of sensitivity in their response and recovery from anticipated transients involving overcooling and undercooling events as well as small-break loss-of-coolant accidents. The recovery from such events has often led to undesirable challenges to engineered safety features (ESF) systems. This sensitivity stems mainly from the small heat sink resulting from the operation of the once-through steam generator (OTSG), which is an inherent design feature of the B&W reactor plants. This sensitivity is further compounded by the lack of sufficient functional and design interface requirements between the nuclear steam supply system (NSSS) and balance-of-plant (BOP) systems. Because of their effect on the NSSS, better interface requirements need to be developed for the auxiliary feedwater system, integrated control system (ICS), non-nuclear instrumentation (NNI) system, and the control room process parameter indications. These aspects are discussed in greater detail in the following sections of this report and in prior staff reports (Refs. 3, 4 and 5).

The Task Force has also conducted an assessment of the effectiveness of the short- and long-term lessons learned actions resulting from the efforts of the Lessons Learned Task Force and the Bulletins and Orders Task Force. This assessment was based on the operating history of the B&W plants during the post-TMI-2 accident period. The results of this review did not reveal any major deficiency in the requirements being applied to B&W facilities. Instead, implementation of these requirements appears to have led to an overall improvement in the response of B&W-designed plants to various transient events. The Task Force supports the expeditious implementation of these requirements, since it is believed that they will contribute to improving to the safety of all operating plants.

It is clear that the OTSG is unique in terms of its capability to affect either rapid cooldown or heatup of the reactor coolant system. However, replacement of the CTSG does not appear to be practical or a necessary action for operating plants, especially when weighed against certain other safety advantages of the OTSG. Furthermore, this Task Force does not believe that complete plant shutdown of the B&W plants is either necessary or desired with regard to public health and safety.

The general findings of the Task Force may be stated as follows:

(1) Confirmation that B&W-designed plants are more responsive to secondary side perturbations than other pressurized-water reactors.

(2) The once-through steam generator design is technically sound; however, it requires a highly interactive and responsive control system (i.e., the integrated control system).

(3) A high degree of overall plant interaction is inherent in the integrated control system and the once-through steam generator.

(4) Based on the design features and the faster response of B&W plants during transients and upset conditions, the operators may be required to take more rapid action and have a better understanding of instrument response than operators on plants having other designs.

The specific recommendations of the Task Force focus on minimizing the consequences of secondary side perturbations (e.g., providing more reliable instrumentation and control systems, assuring availability of heat sink, and improving plant recovery actions). As such, these recommendations fall into four main action areas: (1) auxiliary (or emergency) feedwater, (2) instrumentation and control, (3) design and operational matters, and (4) general areas of improvement to enhance the safety of the B&W-designed reactors. This does not mean that efforts to reduce the frequency of transients should not continue.

As discussed in Section 7 of this report, to determine the effectiveness of each of the recommendations and to assess scheduling priorities, an evaluation of the risk reduction potential associated with each of these recommendations has been performed by the Probabilistic Analysis Staff. In addition, preliminary resource estimates have been provided for planning purposes.

Section 2.2 lists the specific recommendations of this Task Force. Table 2.1 provides a cross-reference listing between the recommendations listed in Section 2.2 and the corresponding recommendation and supporting material in the body of this report. In addition, Table 2.1 shows the preliminary resource estimates and links the Task Force recommendations to the related sections of Draft 3 of the TMI-2 Action Plan.

2.2 Recommendations

Auxiliary (or Emergency) Feedwater

(1) The Task Force strongly recommends that the auxiliary feedwater (AFW) systems on operating B&W plants be classified as an engineered safety feature system, and as such be upgraded as necessary to meet safety-grade requirements. As an alternative, assuming comparable reliability, consideration would be given to the addition of a dedicated AFW system (i.e., a separate train).

Note: With regard to the seismic requirements for safety-grade systems, the Task Force believes that this question warrants further study and, therefore, recommends that the issue be expeditiously resolved by the Probabilistic Analysis Staff.

(2) The AFW system should be automatically initiated and controlled by engineered safety features (safety-grade) that are independent of the ICS, NNI, and other nonsafety systems. The selection of signals used to

initiate AFW system flow should be reevaluated to permit automatic initiation of AFW in a more timely manner to preclude steam generator dryout (i.e., AFW system automatic start an anticipatory loss of feedwater). In addition, the level of secondary coolant in the steam generators should be automatically controlled by the AFW system in a manner to prevent overcooling of the reactor coolant system during recovery from feedwater transients and that an appropriate signal be provided to terminate feedwater flow to the steam generator before overfilling takes place.

(3) Installation of a diverse-drive AFW pump should be expedited at the Davis-Besse 1 facility.

(4) The steam line break detection and mitigation system should be modified as necessary to eliminate adverse interactions between it and the AFW system. In addition, to further assure heat sink availability, the steam line break detection and mitigation system should be reevaluated and modified in such a manner that it is capable of differentiating between an actual steam line break and undercooling or overcooling events caused by feedwater transients.

Instrumentation and Control

(5) B&W plants should improve the reliability of the plant control system, particularly with regard to undesirable failure modes of power source, signal source, and the integrated control system itself. Specific recommendations for improvement in the plant control system include the following:

(a) The power buses and signal paths for non-nuclear instrumentation and associated control systems should be separated and channelized to reduce the impact of failure of one bus.

(b) The power supply (including protective circuitry) logic arrangement should be reconsidered to eliminate "mid-scale" failures as a preferred failure mode for instrumentation. "Full-scale" or "down-scale" failures may be preferred in that they give the operator more positive indication of instrumentation malfunction.

(c) Multiple instrument failures, typically caused by power loss, should be unambiguously indicated to guide operator selection of alternate instrumentation that is unaffected by the failure.

(d) If control system failures or response to failed input signals can cause substantial plant upsets (e.g., required action by engineered safety features or safety valves), the control system should have provisions for detecting gross failures and taking appropriate defensive action automatically, such as reverting to manual control or some safe state.

(e) The NNI power buses should be reviewed and rearranged, as necessary, to provide redundancy of indication of each reactor coolant and secondary system loop. That is, where indicators for one loop are provided, one channel should be powered from NNI "X" and the other from NNI "Y," instead of loop "A" being powered from NNI "X" and loop "B" from NNI "Y."

(f) Prompt followup actions should be taken on the recommendations contained in BAW-1564 (Integrated Control System Reliability Analysis).

(g) NRC has reviewed the recommendations contained in NSAC-3/INPO-1 (Analysis and Evaluation of Crystal River Unit 3 Incident). The staff agrees that licensees should evaluate the effectiveness of these recommendations for their plants, especially with regard to the NNI/ICS aspects.

(h) Prompt followup actions should be taken on IE Bulletin 79-27.

(6) A minimum set of parameters should be established to enable the operator to assess plant status. The set recommended by the Task Force follows:

(a) Wide range reactor coolant system pressure,

(b) Wide range pressurizer level,

(c) Wide range reactor coolant system temperatures: hot leg (each loop), cold leg (each loop), and core outlet (two or selectable),

(d) Makeup tank level,

(e) Reactor building pressure,

(f) Wide range steam generator level (both OTSGs),

(g) Wide range steam generator pressure (both OTSGs),

(h) Source range nuclear instrumentation, and

(i) Intermediate range nuclear instrumentation,

(j) Borated-water storage tank (BWST) level.

The instrumentation for the selected parameters must meet the following requirements:

(a) The instrumentation must be reliable and redundant and should meet all applicable codes and standards for protection system instrumentation; and,

(b) In accordance with safety standards, these require a minimum of two redundant channels of all designated information. At least one channel of which shall be recorded automatically on a timely basis for use in trending, instant recall, and post-event evaluation.

(7) All B&W plants should provide the flexibility to substitute appropriate combinations of incore thermocouples for the loop resistance temperature detectors (RTDs) presently used for primary temperature input to the subcooling meter. All B&W plants should provide the capability of having

a continuous or trending display of incore thermocuples. This display need not be indicated in the control room at all times but may be called up on demand from the computer.

(8) B&W plants should provide a safety-grade containment high radiation signal to initiate containment vent and purge isolation in addition to the presently required signals (i.e., containment high-pressure and low-pressure ESFAS actuation).

Design and Operational Matters

(9) Following a reactor trip, pressurizer level should remain on scale, and system pressure should remain above the HPI actuation setpoint. The system response (e.g., secondary pressure) should be modified to meet the above two objectives. Meeting these objectives should be independent of all manual operator actions (e.g., control of feedwater, letdown isolation, and startup of a makeup pump).

(10) The B&W licensees should perform sensitivity studies of possible modifications which would reduce the response of the OTSG to secondary coolant flow perturbations. Both passive and active measures should be investigated to mitigate overcooling and undercooling events.

(11) Modifications should be made to the plant, to the extent feasible, to reduce or eliminate manual immediate actions for emergency procedures.

(12) A qualified Instrumentation and Control Technician (I&CT) should be provided on a round-the-clock basis at all operating B&W reactors.

(13) Lectures should be developed and given promptly to all licensed personnel concerning the Crystal River 3 event as well as their plant-specific loss of NNI/ICS analysis. A means to evaluate the training (e.g., quizzes) should be included. This training should be audited by the Office of Inspection and Enforcement.

(14) Licensees should develop and implement promptly plant-specific procedures concerning the loss of NNI/ICS power. These procedures should enable the operator to bring the plant to a safe shutdown condition. These procedures shall be audited by the Office of Inspection and Enforcement. Furthermore, the Task Force endorses the effort by B&W to develop abnormal transient operational guidelines and recommends full utility support be given to this program.

(15) Mandatory one-week simulator training should be required for all licensed B&W operators. The training should be oriented toward or include undercooling and overcooling events, solid system operation, and natural circulation cooling. Upgrading of simulator performance in accordance with the recommendations of the TMI-2 Action Plan (NUREG-0660) should be expedited.

(16) The NRC should review the criteria for RCP restart during recovery from non-LOCA transients as provided in B&W small-break guidelines. Restarting the RCPs provides the operator with pressurizer spray and thus greatly improves plant pressure control.

(17) In order to provide an alternative solution to PORV unreliability and safety system challenge rate concerns, the following proposal (submitted by Consumers Power Company) should receive expenditious staff review for possible consideration and backfit on all B&W operating plants:

(a) Provide a fully qualified safety-grade PORV;

(b) Provide reliable safety-grade indication of PORV position;

(c) Provide dual safety-grade PORV block valves, capable of being automatically closed if a PORV malfunction occurs;

(d) Complete a test program to demonstrate PORV operability;

(e) Install safety-grade anticipatory reactor trip on total loss of feedwater; and

(f) Reset the PORV and high-pressure trip setpoints to their original values of 2255 psig and 2355 psig, respectively.

(18) The IREP Crystal River study should be completed and thoroughly documented in an expeditious manner. When that effort is completed, the Task Force recommends the following actions:

(a) The Probabilistic Analysis Staff (PAS) should consider the need for additional Crystal River work to examine particular unresolved questions that may be evident, and reexamine the scope, methods, and format of the first IREP study so that modifications may be made prior to the initiation of further IREP work.

(b) Appropriate staff in NRR (in coordination with the PAS) should make prompt determinations with respect to the need for additional modifications to the Crystal River plant.

General Areas for Improvement

(19) Plant performance criteria for anticipated transients should be established for all light-water reactors. Industry should have a significant role in the development of these performance criteria.

(20) The criteria for tripping RCPs during small-break LOCAs should continue to be studied. To this end, the Task Force endorses the NSAC/INPO recommendation that the evaluation should be conducted jointly by both the industry and the NRC.

(21) The need to introduce AFW through the top spray sparger during anticipated transients should be reevaluated by licensees. This reevaluation should consider the reduced depressurization response if AFW could be introduced through the main feedwater nozzle and could enter the tube region from the bottom of the unit.

(22) The staff should perform an analysis of the number of Licensee Event Reports attributed to licensed personnel error to determine the significance and cause of the higher number associated with the operation of B&W facilities.

TABLE 2.1

CROSS-REFERENCE LISTING OF RECOMMENDATIONS

Recommendation Number	Subject	Number in Report	Page	Related Requirement in the TMI-2 Action Plan	NRR Staff Resources (estimated) FY80 PMY	FY80 Tech. Assist.	FY81 PMY	FY81 Tech. Assist.
Auxiliary Feedwater (AFW) System								
1.	AFW system upgrade to safety grade.	5.2.6.3(1)	5-42	II.E.1.1	0.5	-	1.2	-
2.	AFW system automatic initiation and control.	5.2.6.3(2)	5-44	II.E.1.2	0.5	25K	0.5	25K
3.	Addition of diverse drive AFW pump for Davis-Besse.	5.2.6.3(3)	5-45	II.E.1.1	0.1	-	-	-
4.	Modifications to steam line break detection and mitigation system.	5.2.7.3(1)	5-47	None	0.2	20K	0.2	50K
Instrumentation and Control								
5.	Improvements in plant control system (ICS/NNI).	5.3.5.1(1)	5-61	None	0.3	25K	0.3	25K
6.	Selected data set of principal plant parameters for operator.	5.3.5.1(2)	5-64	I.D.2	0.1	-	0.1	-
7.	Increased usage of incore thermocouples.	5.3.5.1(3)	5-65	None	0.1	-	0.1	-
8.	High radiation signal initiation of containment isolation.	5.3.5.1(4)	5-66	II.E.4.2	0.1	25K	0.1	50K

TABLE 2.1 (continued)

Recommendation Number	Subject	Number in Report	Page	Related Requirement in the TMI-2 Action Plan	NRR Staff Resources (estimated) FY80 PMY	FY80 Tech. Assist.	FY81 PMY	FY81 Tech. Assist.
Design and Operational Matters								
9.	System response to maintain pressurizer level on scale and pressure above HPI setpoint.	5.2.1.2(1)	5-14	II.E.5	0.1	-	0.1	-
10.	Sensitivity studies of operational modifications.	5.2.2.4(1)	5-19	II.E.5	0.1	-	0.1	-
11.	Modifications to eliminate immediate manual actions for emergency procedures.	5.4.2.2(1)	5-75	None	0.1	-	0.1	-
12.	Qualified I&C Technician on duty.	5.4.3.3(4)	5-83	None	0.1	-	-	-
13.	Operator training on Crystal River 3 event.	5.4.3.3(3)	5-82	None	-	-	-	-
14.	Procedures for loss of NNI/ICS.	5.4.3.3(2)	5-82	None	0.1	-	-	-
15.	Mandatory one-week simulator training for operators as part of requalification program.	5.4.3.3(1)	5-81	None	-	-	-	-
16.	Evaluation of RCP restart criteria.	5.2.4.2(1)(a)	5-31	II.K.1 Table C.1 Item 27	0.05	-	-	-
17.	Alternative solution to PORV unreliability/ safety system challenge rate concerns.	5.2.3.3(3)	5-28	None	0.2	25K	0.2	50K
18.	IREP Crystal River Study.	6.2.2(1)	6-4	II.C.1	0.5	15K	-	-

TABLE 2.1 (continued)

Recommendation Number	Subject	Number in Report	Page	Related Requirement in the TMI-2 Action Plan	NRR Staff Resources (estimated) FY80 PMY	FY80 Tech. Assist.	FY81 PMY	FY81 Tech. Assist.
General Areas for Improvement								
19.	Performance criteria for anticipated transients.	5.2.3.3(2)	5-28	None	0.2	50K	0.1	-
20.	Continued evaluation of need to trip RCPs during small break loss-of-coolant accidents.	5.2.4.2(1)(b)	5-32	II.K.3 Table C.3 Item 5	0.1	25K	0.1	-
21.	Reevaluate location of AFW injection into OTSG.	5.2.3.3(1)	5-25	None	0.05	-	-	-
22.	Staff study of personnel related LERs with respect to high number for B&W plants.	5.4.2.2(2)	5-75	I.E.8	0.05	-	-	-
				Total	3.55	210K	3.2	225K
				IE Implementation Inspection (Approx.)			5	

3. POST TMI-2 ACCIDENT REVIEWS AND REQUIREMENTS FOR B&W OPERATING PLANTS

3.1 Introduction

The accident at Three Mile Island Unit 2 (TMI-2) on March 28, 1979, involved a main feedwater (MFW) transient coupled with a stuck-open pressurizer power-operated relief valve (PORV) and a temporary failure of the auxiliary feedwater (AFW) 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: (1) assure that other reactor licensees, particularly those with plants similar in design to TMI-2, took the immediate actions necessary to substantially reduce the likelihood of a TMI-2 type accident from reoccurring, and (2) investigate the potential generic implications of this accident on other operating reactors. This section of the report summarizes the history and status of the actions required by the NRC to be taken by the holders of operating licenses for Babcock & Wilcox (B&W)-designed reactors.

3.2 IE Bulletins

During the staff's preliminary assessment of the TMI-2 accident, it became apparent that several design problems, equipment malfunctions, and human errors contributed significantly to the severity of the accident and subsequent core damage. As a result, the B&W operating plant licensees were instructed in a series of Inspection and Enforcement (IE) bulletins to take a number of immediate actions to avoid repeating the same mistakes. The TMI-2-related IE bulletins applicable to B&W-designed reactors are numbered 79-05, 79-05A, 79-05B, and 79-05C and were issued on April 1, April 5, April 21, and July 26, 1979, respectively. Some of the more significant issues dealt with in the bulletins were: (1) review and modification of many operating and emergency procedures associated with loss-of-coolant accidents (LOCAs), loss of feedwater and other transients which result in high reactor coolant system (RCS) pressure, natural circulation cooling, operation of the AFW system, maintenance, and surveillance testing; (2) criteria for the termination and override of engineered safety features (ESF) systems; (3) review and modifications to the containment isolation design and procedures; (4) provisions for manually tripping the reactor on transients resulting in high pressure in the RCS; (5) changes to decrease the setpoint of the high pressure reactor trip and increase the setpoint of the PORV; and (6) requirements to trip the operating reactor coolant pumps (RCPs) during LOCA conditions. A complete listing of the actions required by these bulletins are tabulated in Appendix A to this report (pages A-1 through A-6).

3.3 Commission Orders

In parallel with the issuance of the IE bulletins, a task group was appointed to perform a generic assessment of the feedwater transients experienced in B&W plants, including the accident at TMI-2, to determine bases for continued safe operation of these facilities in both the short term and the long term. Consideration was given by the task group to initiating events other than loss of feedwater where it was determined that such events could lead to similar transient conditions. It was based upon the preliminary principal findings of this task group that the document entitled "NRR Status Report on Feedwater Transients in B&W Plants," dated April 25, 1979 (Ref. 1), was prepared. The document concluded that there was not reasonable assurance of protection of the public health and safety in the continued operation of the B&W plants and that the plants should be shut down until certain modifications could be made to the facilities. It was determined that in order to reestablish that assurance the following items would have to be accomplished: (1) review and upgrade the reliability and performance of the AFW system; (2) review the integrated control system (ICS) and take actions to reduce its likelihood of initiating or exacerbating transients; (3) installation of anticipatory reactor trips based upon feedwater transients; (4) review detailed analyses of plant response to transients including the effects of high pressure injection (HPI) operation and natural circulation cooling; and (5) develop new standing instructions and emergency procedures for plant operators and conduct operator training in the use of those procedures. In the long term, it was recommended that either the sensitivity of the response of the B&W plants to transients be improved by design changes or substantially upgrade the instrumentation and controls available to the plant operator and substantially upgrade operator education, training, and experience. On April 26 and 27, 1979, meetings were held between the staff and representatives of the B&W reactor licensees. As a result of these meetings, the licensees agreed to shut down their facilities until certain modifications in equipment, procedures, and operator training were completed. Based upon these commitments by the licensees, the Commission directed the staff to prepare confirmatory Orders to formalize the agreements reached with the utilities. During the period May 7 through May 17, 1979, these Orders were issued to each of the B&W reactor licensees.

The Orders required that the B&W operating plants shut down (or remain shut down) until certain immediate or short-term actions were completed and found acceptable by the staff. In addition, the Orders specified certain long-term modifications to be performed to further enhance the capability and reliability of the B&W reactors to respond to various transient events. A complete listing of the short- and long-term requirements of the Orders may be found in Appendix A of this report (pages A-7 through A-14).

3.4 NUREG-0560

The special task group (discussed in Section 3.3) assessing B&W feedwater transients completed its work in May 1979. The findings of this task group are found in the report entitled "Staff Report on the Generic Assessment of Feedwater Transients in Pressurized Water Reactors Designed by the Babcock and Wilcox Company" (NUREG-0560, May 1979, Ref. 3). In its report, the task group concluded that certain design improvements and other actions being implemented on the B&W plants in accordance with the Commission Orders were necessary before plant operation could be resumed. The actions specified in the Orders that resulted from the generic review included: (1) reactor trip upon upsets in the secondary system (loss of feedwater and turbine trip), (2) additional operator training, (3) improvements in AFW system reliability, and (4) further analyses of small break LOCAs. Additional recommendations from this review are found in Section 8.0 of NUREG-0560. In general, these recommendations include the short-term actions taken in connection with IE Bulletins and the May 1979 Commission Orders. Other recommendations from NUREG-0560 served as a basis for work carried out by certain other task forces which were created within the Office of Nuclear Reactor Regulation (NRR).

3.5 Bulletins & Orders Task Force

The Bulletins & Orders Task Force (B&OTF) was established within NRR during early May 1979 and continued in operation until 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 LOCAs 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 the operating plant licensees on plant specific matters. For the review of B&W generic matters, a working relationship was established with the B&W Owners' Group, which was comprised of representatives of each of the B&W operating plants. In some cases, work was conducted directly with the Power Generation Group of the Babcock & Wilcox Company.

At the onset, work of the B&OTF concentrated on plants of the B&W design; as short-term actions on these plants were completed, priority was shifted to those pressurized water reactors designed by Westinghouse and Combustion Engineering and then to boiling water reactors designed by General Electric.

With respect to the B&W plants, the B&OTF evaluated responses by the B&W Licensees to the TMI-2-related IE Bulletins (79-05, 79-05A, 79-05B, and 79-05C) as well as evaluating the licensees' compliance with the short- and long-term requirements of the Commission Orders of May 1979. Between May 18 and July 6, 1979, the B&OTF completed its review of the actions taken by the B&W licensees to comply with the short-term requirements of the Orders. As a result of these evaluations, the B&W licensees were authorized to resume power operations. The activities involving the B&W-designed reactors are reflected in three documents produced by the B&OTF:

(1) "Generic Assessment of Delayed Reactor Coolant Pump Trip During Small Break Loss-of-Coolant Accidents in Pressurized Water Reactors" (NUREG-0623, November 1979, Ref. 12),

(2) "Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox Designed 177-FA Operating Plants" (NUREG-0565, January 1980, Ref. 4), and

(3) "Report of the Bulletins and Orders Task Force," Volumes I and II (NUREG-0645, January 1980, Ref. 5).

A listing of the recommendations from the B&OTF, applicable to the B&W operating plants, may be found in Appendix A to this report (pages A-15 through A-17).

3.6 Lessons Learned Task Force

In late May 1979, the Lessons Learned Task Force (LLTF) was formed. The purpose of the LLTF was to identify and evaluate those safety concerns originating from the TMI-2 accident that required licensing actions, beyond those specified in the IE Bulletins and the Commission Orders of May 1979, for all operating reactors as well as for pending operating license (OL) and construction permit (CP) applications. In developing the required actions, the LLTF considered: the review and evaluation of investigative information; staff evaluations of responses to IE Bulletins and Commission Orders; recommendations from the Commissioners, the ACRS, the NRC staff, NUREG-0560, as well as recommendations from outside the NRC. In general, the LLTF was charged with identifying, analyzing and recommending changes to licensing requirements and the licensing process for all nuclear power plants based on the lessons learned from the TMI-2 accident. The scope of the LLTF included the following seven general technical areas: (1) reactor operations, including operator training and licensing; (2) licensee technical qualifications; (3) reactor transient and accident analysis; (4) licensing requirements for safety and process equipment, instrumentation, and controls; (5) onsite emergency preparations and procedures; (6) NRR accident response role, capability and management; and (7) feedback, evaluation, and utilization of reactor operating experience. In July 1979, the LLTF issued a document entitled "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations" (NUREG-0578, Ref. 13). The recommendations contained in NUREG-0578 were narrow, specific, and urgent in nature and were intended to constitute a sufficient set of short-term requirements to ensure the safety of plants already licensed to operate and those to be licensed for operation in the near future. Following a review of NUREG-0578 by the ACRS and the Director, Office of Nuclear Reactor Regulation, all of the recommendations (except three requiring rulemaking) were directed to be implemented by all operating nuclear power plants. In addition to the recommendations specified in NUREG-0578, the requirements for remotely operable high point vents for noncondensable gas removal and three additional instrumentation requirements (designed to follow the course of an accident) were al.. directed to be implemented. Appendix A of this report lists the short-t m lessons learned requirements applicable to the B&W-designed reactors (pages A-18 through A-20).

In October 1979, the LLTF issued "TMI-2 Lessons Learned Task Force Final Report" (NUREG-0585, Ref. 14). In contrast to the short-term recommendations, the recommendations contained in NUREG-0585 dealt with safety questions of a more fundamental policy nature regarding nuclear plant operations and design as well as the regulatory process. Most of the recommendations were goal oriented rather than prescriptive in nature and if adopted would cause significant changes within the nuclear industry and the regulatory process. However, certain recommendations related to power plant operations were recommended to be initiated without delay since they would introduce a needed stepwise improvement in safety. A listing of the recommendations contained in NUREG-0585, which are applicable to B&W-designed reactors, are included in Appendix A to this report (page A-21).

3.7 Other Review Groups/TMI-2 Action Plan

In addition to the special task forces previously discussed, other groups who have investigated the accident at TMI-2 and advanced certain recommendations include the Congress, the General Accounting Office, the President's Commission on the Accident at Three Mile Island, the NRC Special Inquiry Group, the NRC Advisory Committee on Reactor Safeguards, the Special Review Group of the NRC Office of Inspection and Enforcement, the NRC staff's Siting Policy Task Force and Task Force on Emergency Planning, and the NRC Offices of Standards Development and Nuclear Regulatory Research. Each of the investigating groups organized their recommendations in a different way. In order to organize, define, and assess the recommendations of these various groups, a "TMI-2 Action Plan Steering Group" was appointed. The charter of this group was to develop a "TMI-2 Action Plan" which would provide a comprehensive and integrated plan for all actions judged necessary by the NRC to correct or improve the regulation and operation of nuclear facilities based on the experience from the accident at TMI-2 and the official studies and investigations of the accident. The first draft of this report entitled "Action Plans for Implementing Recommendations of the President's Commission and Other Studies of TMI-2 Accident" (NUREG-0660) was published on December 10, 1979. Refinements to the task descriptions and schedules contained in NUREG-0660 have been made in subsequent drafts. Draft 3 of the TMI-2 Action Plan was issued on March 5, 1980 (Ref.7). Actions to improve the safety of the B&W-designed operating plants, as well as the other vendor-designed reactors now operating, which were necessary immediately after the accident and could not be delayed until an action plan was developed are also documented in the TMI-2 Action Plan. Those requirements placed on the B&W operating plant licensees are tabulated in Appendix A to this report. They have been organized in the appendix by category source (e.g., IE Bulletins, Orders, and LLTF recommendations) and have been cross-referenced to Draft 3 of the TMI-2 Action Plan.

Throughout the development of the TMI-2 Action Plan, there has been a consensus that the accident demonstrated that additional improvements in safety are needed. There is also general agreement among the various investigations as to the causes of the accident and the failures and errors that occurred before and during the event, both in the equipment and in the organizations that built, operated, and regulated the plant. There is also general agreement as to the areas where improvements should be made. Most of the questions relate to the degree of improvement required and the best ways of achieving the improvement. The action plan serves as a collective assessment of the types and degree of improvement necessary and attempts to optimize the means of attaining this improvement.

As discussed throughout this section, many requirements, developed from various organizations within the NRC, have been issued to the B&W operating plant licensees in the year subsequent to the accident at TMI-2. Many of these requirements have been completed or are scheduled to be completed shortly. However, the TMI-2 Action Plan, being an integrated plan, addresses the entire NRC program, including short-term changes in requirements for all operating reactors and requirements for licensing new plants for operation. The action plan reveals that the present formulation of the long-term requirements, including the NRC's research program necessary to improve or confirm the adequacy of the various short-term licensing requirements, is not as detailed or specific as the short-term actions. However, at the present time, the action plan serves as the most complete listing of requirements that have been directed to be implemented on the B&W-designed operating plants and serves as the best forecast as to the requirements that may be imposed in the future.

4. SUMMARY OF OPERATING EXPERIENCE

4.1 Introduction

The purpose of reviewing operating events which have occurred is to bring to bear on current issues the knowledge gained from prior experience. This knowledge provides insight on complex performance and response characteristics inherent in plant design, power generation, control and safety systems interactions, reliability of these systems, modifications to hardware, changes in operating philosophy and procedures and the generic implications that arise from specific events or patterns of events.

The events summarized in this section deal with anticipated operating transients in pressurized water reactors. Several transients of varying levels of severity have occurred in B&W-designed reactor plants, which as a result, have been subjected to conditions of overcooling, undercooling, loss of reactor coolant inventory, or a combination thereof. Many transients have challenged safety systems. A wide variety of root causes, such as equipment failures or human errors in conjunction with system response as designed or system malfunctions, have initiated or aggravated the course of these transients. In contrast, however, the severity of many events has also been mitigated by automatic response of systems and manual intervening actions by the plant operating staff.

This section discusses sources and validity of information and observed transients in terms of initiators, integrated plant response, and resulting conditions that have aggravated or mitigated severity. The tables of events in Appendix B are described and analyzed in this section.

4.2 Sources of Information

Sources of information used in compiling the operating experience described in this report include: the NRC licensee event report (LER) file, NRC Gray Book (Ref. 15) data on forced outages, selected letters from licensees of B&W plants (Refs. 16 through 22) and from the Babcock & Wilcox Company (Ref. 23).

Appendix B to this report contains tables of reactor trip events that are relevant to the issues considered by the Task Force. The tables are not a complete listing of all reactor trip events. In its report entitled "Integrated Control System Reliability Analysis" (Ref. 35), B&W stated that a total of 310 reactor trips had occurred at B&W-designed power reactors. This number is approximately 20 percent greater than the number of reactor trips listed in other documents received by the NRC and the Gray Book (Ref. 15). The Task Force believes that the information provided is accurate and that the most significant events were included. Completeness of Appendix B tables, however, warrants qualification for other reasons discussed below.

The severity of transients affecting the reactor coolant system (RCS) heat sink is believed to vary over a wide continuum. The relatively severe transients have been identified, but the total population of transients remains unknown. Reactor trip, engineered safety features actuation, forced outage or conditions or events reportable to the NRC have not occurred in all transients. In less severe transients, control band limits have not been reached because initiators had only minor effects, or a combination of proper control system response and operator actions terminated the transient. Maneuvering transients, which are accommodated routinely by control systems or operator action, comprise the low severity end of the continuum.

The evolution of reporting requirements has led to more reports and improved report content. Maneuvering and minor transients are not reportable although the threshold of reportability has been reduced to include less severe events. Recent changes in NRC regulations have increased the number of reports received by the NRC Operations Center. Therefore, knowledge of operating events in the recent past is believed to be more complete than in prior years.

4.3 Discussion of Observed Transients

4.3.1 Transient Initiators

Conditions of overcooling, undercooling, and loss of reactor coolant inventory have resulted from a wide variety of initiators. Initial perturbations have been caused by human error in plant operations or maintenance activities, component failures or external events. These initiators have on occasion caused a loss of power to control systems or instrumentation that comprises control system inputs. In the first case, control systems have malfunctioned, whereas in the latter case, control systems may have responded as designed to incorrect failed inputs. Failures of mechanical components such as: a PORV; steam dump valve; pressurizer spray valve; feedwater pumps; condensate pumps, demineralizer or other components, or feed and condensate system valves have also led to overcooling, undercooling, or loss-of-coolant inventory conditions.

4.3.2 Integrated Plant Response

The integrated plant response to initiating events has included steam demand or feedwater transients, which have been reflected across the steam generator and have resulted in RCS pressure, temperature, and pressurizer level excursions due to undercooling and overcooling conditions. Response to root causes affecting the RCS, such as PORV opening, has led directly to RCS pressure transients.

Normal operating pressure is bounded above and below as follows:

RCS Pressure (psig)	Plant Response
2500	Pressurizer code safety valves open
2450	PORV opens (Davis-Besse 1 PORV set at 2400 psig)
2300	Reactor trip on high pressure
2155	Normal operating pressure
1900	Reactor trip on low pressure
1500-1650	Engineered safety features actuation (specific setpoints vary on B&W operating plants; e.g., high-pressure injection, etc.)

Prior to implementation of post-TMI-2 accident setpoint changes, the PORV opening setpoint was 2255 psig, and reactor trip on high pressure was set at 2355 psig.

In many cases, plant parameters have exceeded their operating band boundaries causing reactor trip, engineered safety features actuation, PORV opening, or pressurizer safety valve lifting due to the RCS pressure transient. Safety systems have been challenged during severe anticipated transients. Operators have routinely taken immediate actions on reactor trips to limit the swing in plant parameters and to facilitate recovery to a stable plant condition. Starting a second makeup pump (injecting water into the RCS through either normal charging or HPI connections) and isolating letdown flow are immediate operator actions taken in an attempt to compensate for coolant contraction and to reduce the probability that pressurizer level indication may go offscale low. Plant response to reactor trip transients, given no operator action, has not been observed. Thus, the degree to which operator action reduces severity is unknown.

4.3.3 Impact of Integrated Plant Response

Conditions of undercooling, overcooling, and loss-of-coolant inventory are distinguished from steam demand and feedwater transients because on occasion, all three of these conditions have occurred at some time during plant response to transients. Loss-of-feedwater events have characteristically reduced steam generator effectiveness as a heat sink. In many cases, steam generators have been boiled dry. The resulting power mismatch (undercooling condition or loss of heat sink) has driven RCS temperature and pressure high enough to cause a reactor trip on high pressure. Initiation of auxiliary feedwater flow to regain the heat sink, combined with steam bypass system operation, has on occasion caused overcooling and rapid depressurization of the RCS to a pressure low enough to initiate high-pressure injection. This depressurization has occurred even though a second makeup pump had been started and letdown isolated (both manual operator actions immediately after the reactor trip). Per emergency procedures after the TMI-2 accident, reactor coolant pumps have been tripped following reactor trip and HPI initiation on low pressure. Excessive reactor vessel and pressurizer cooldown rates have occurred during overcooling conditions. Feeding steam generators per procedure to the 95 percent level on the operate range for natural circulation cooling of the RCS has on occasion caused another loss or partial loss of heat sink condition. This has occurred because reduced steam pressure has been sensed by the steam line break detection and mitigation system, where installed, which has automatically isolated steam and feedwater connections to the steam generator.

During events in which the PORV opened and stuck open, impact on the plant has been aggravated by faster depressurization, lower minimum RCS pressure, slower recovery of pressurizer level by the high-pressure injection system, and a longer time to establish stable plant conditions.

Loss of instrumentation and failure-induced control system response (e.g, failure of control systems or their input signals) have contributed to transient severity or hampered operator attempts to stabilize plant conditions.

4.4 Summary of Events

Events of interest to the Task Force involve power supply failures to non-nuclear instrumentation (NNI) or the integrated control system (ICS), reactor trips, PORV actuations, and feedwater transients. An historical summary of NNI/ICS power failures in B&W plants is provided in Appendix B, Table B.1. Appendix B, Table B.2 contains (1) details of reactor trips with PORV actuation in B&W plants that occurred before the TMI-2 accident, and (2) the documented number of additional automatic reactor trips in B&W plants during which PORV opening was not reported. Table B.3 contains details of all unplanned reactor trips in B&W plants since the TMI-2 accident. Licensee estimates of the effects of the two different setpoints for high-pressure reactor trip and PORV actuation are included in Tables B.2 and B.3 (i.e., what would the effect have been during these events if present setpoints had been used before the TMI-2 accident and if pre-TMI-2 setpoints were in effect instead of present setpoints).

4.4.1 Analysis of Data in Table B.1 (Appendix B)

Since December 1974, a total of 29 NNI/ICS power failures have been identified. Twenty-one of these events caused reactor trips, 17 caused PORV actuation, and 4 resulted in engineered safeguards (high-pressure injection) actuations. A steam dump valve stuck open in one event, and feedwater transients occurred in 19 of these events. A pressurizer safety valve lifted in one event and a PORV stuck or failed open in three events. Three ICS power failures that occurred while the reactor was at power did not cause reactor trips, and in the remaining five power failures the reactor was in a shutdown condition when the events occurred.

Based on these data, NNI/ICS power failure perturbations have been severe enough, considering the B&W-integrated plant response characteristics, to cause reactor trip in almost all events (Note: in most of these instances, the reactor trip was the result of a consequential feedwater transient). Approximately 18 percent of all observed feedwater transients have been caused by NNI/ICS power failures. Approximately 10 percent of all reactor trips have been associated with NNI/ICS power failures. These percentages and the data in Tables B.2 and B.3 (discussed later) indicate that many other initiators of feedwater transients and reactor trips exist. The data in Table B.1 appear to show that, given an NNI/ICS power failure, it is very likely to result in a severe feedwater transient that will trip the reactor on high pressure (even at the pre-TMI-2 reactor trip setpoint of 2355 psig).

4.4.2 Analysis of Data in Table B.2 (Appendix B)

The data preceding the TMI-2 accident suggest a preponderance of reactor trips in response to secondary plant transient or upset conditions that were reflected across the steam generators and caused reactor coolant system (RCS) pressure excursions. Prior to the TMI-2 accident, no anticipatory reactor trip on occasion of turbine trip or loss of feedwater existed. Reacto rip with PORV actuation occurred approximately 149 times. Other automatic reactor trips, during which PORV opening was not reported, occurred 83 times in B&W plants prior to the TMI-2 accident. Notwithstanding inaccurancies which may exist in the data, it is significant to note that (1) RCS pressure excursions have been severe enough to cause not only PORV actuations (2255 psig old setpoint) but also reactor trip (2355 psig old setpoint or manual trip based on operator judgment) in approximately 149 instances, and (2) reactor trip transients with PORV actuations have occurred about twice as many times as automatic reactor trip transients in which licensees did not report PORV actuations. The number of PORV openings not accompanied by reactor trip is unknown. Therefore, the data available do not permit inferences about the ability of the PORV to counter pressure excursions and prevent reactor trip, thereby enabling the plant to "ride through" the transient without challenging safety systems. However, in approximately 149 of the most severe transients, PORV actuation did not prevent a reactor trip.

Table B.2 also indicates that a loss of feedwater, feedwater reductions, or feedwater oscillations caused or contributed to a PORV actuation and reactor trip transient in approximately 60 percent of the total number of PORV/reactor trip events preceding the TMI-2 accident. An unscheduled turbine trip resulted in a reactor trip/PORV actuation in less than 20 percent of the total number of pre-TMI-2 events presented in Table B.2.

In summary, of the events listed in Table B.2, the loss of feedwater or feedwater upsets were the major contributor to PORV actuation and reactor trip transients. The pre-TMI-2 data also show many pressure excursions were severe enough to cause reactor trips at 2355 psig even though the PORV had opened at its previous setpoint of 2255 psig.

4.4.3 Analysis of Data Table B.3 (Appendix B)

Since the TMI-2 accident, a total of 38 unscheduled reactor trips were identified. Twelve of these trips have been attributed to turbine trip and 15 due to feedwater losses or upsets. In comparison with pre-TMI-2 data (60 percent of reactor trip/PORV actuation events due to feedwater and 20 percent of these events due to turbine trip), it is apparent that the turbine trip, which now causes an automatic reactor trip, has become a more significant contributor to reactor trips.

Table B.3 also shows that PORV actuation has not occurred due to high RCS pressure during the 21 transients where those data were available. In the licensees' judgment if old setpoints were in effect, only three (one-seventh) of these 21 transients would have caused a reactor trip and 13 (two-thirds) of them would have opened the PORV. It is unknown whether these estimates of what would have happened had old setpoints been in effect are valid predictors of the long-term effects of revised setpoints. Although a higher fraction of all reactor trips are now caused by turbine trips, the fraction of turbine trips that did not cause reactor trips under the old setpoints is not known.

A total of only nine turbine trips (post-TMI-2) had occurred when licensees made the judgments that none of those turbine trips would have caused reactor trips. The inference that no future turbine trips would have caused reactor trips under old setpoints does not appear to be a valid predictor of effects of setpoint changes because pre-TMI-2 history shows that about 20 percent of all PORV actuations with a reactor trip were caused by turbine trips.

4.4.4 Effects of Revised Setpoints and Anticipatory Reactor Trips

Table 4.1 was developed by B&W and submitted by the B&W licensees (Ref. 22) in response to an NRC request. The analysis was completed in October 1979 using reactor trip data accumulated through September 1979. An updated analysis reflecting more operating time since the TMI-2 accident is enclosed in the summary of a January 29, 1980 meeting on selected responses on B&W system sensitivity (Ref. 43). The data presented in Table 4.1 show all reactor trips that have occurred at B&W facilities divided into two categories: "Pre TMI-2" and "Post TMI-2." Each of these categories is further divided into two additional categories: "A" (trips that were affected by the setpoint changes in the PORV and high-pressure reactor trip and the addition of anticipatory reactor trips for turbine trip and feedwater upsets) and "B" (total trips: Category "A" trips plus those trips not affected by the post-TMI-2 changes). The Category "B" trips include all reactor trips that, in B&W's judgment, would have occurred irrespective of setpoint changes or the addition of anticipatory reactor trips. (Note: for the specific trips listed in Tables B.2 and B.3, the effect of the revised setpoints, in B&W's judgment, is shown at the right-hand side of the tables.)

Reactor trip frequency was developed using the number of commercial days in operation converted to months. A comparison of pre- and post-TMI-2 reactor trip frequencies indicates that from March 28, 1979, through September 1979, substantial increase in average trip frequency occurred. The possible contributors to the post-TMI-2 trip frequency increase were (1) short period of operation since TMI-2 (1452 operating days or 13 percent of the 11,300 days before TMI-2), (2) many startups and shutdowns since the TMI-2 accident, (3) lack of operator experience at operating with the revised setpoints, and (4) statistical variations in pre- and post-TMI-2 samples.

In summary, the B&W data in Table 4.1 are presented as an historical summary of the number of reactor trips and trip frequencies observed through 1979. The average number of reactor trips/month of operation prior to the modifications was 0.56/facility. The average number of reactor trips/month of operation following the modifications has been 0.65/facility. The difficulties in accepting these trip frequences as valid inferences of the effects of revised setpoints are discussed in Sections 4.4.2 and 4.4.3 of this report. Thus, the reader is cautioned against using the reactor trip frequencies presented in Table 4.1 as predictors of future performance.

The data given in Table 4.1 were analyzed by the Probabilistic Analysis Staff, RES, using chi-square and likelihood ratio (small sample distribution) statistics. For the seven units with both pre- and post-TMI-2 data, analysis supports the following conclusions:

(1.) At the 95 percent confidence level, the plants behave more like one another after the modifications (revised setpoints) than before the modifications. This is true considering total trips or just those trips due to high pressure, feedwater upsets, and turbine trips. Crystal River 3 is the possible exception to the homogeneous behavior among plants after TMI-2.

(2.) At the 95 percent confidence level, Crystal River 3 and Rancho Seco show significantly higher trip frequencies after the modifications versus before the modifications for trips due to high pressure, feedwater upsets, and turbine trip.

(3.) At the 95 percent confidence level, only Rancho Seco shows a significantly higher trip frequency considering total trips.

A summary table of key points associated with the various transient events discussed throughout Section 4.4 is provided in Tablt '.2.

TABLE 4.1

EFFECT OF REVISED SETPOINTS AND ANTICIPATORY REACTOR TPIP ON TRIP FREQUENCY FOR B&W OPERATING PLANTS

B&W Plants	Pre-TMI-2 No. of Trips A	Pre-TMI-2 No. of Trips B	Pre-TMI-2 Trips/Mo. A	Pre-TMI-2 Trips/Mo. B	Post-TMI-2 No. of Trips A	Post-TMI-2 No. of Trips B	Post-TMI-2 Trips/Mo. A	Post-TMI-2 Trips/Mo. B
Oconee-1	31	52	0.45	0.76	2	5	0.24	0.60
Oconee-2	9	28	0.16	0.51	2	3	0.26	0.39
Oconee-3	17	27	0.33	0.53	2	2	0.68	0.68
Davis-Besse 1	5	23	0.31	1.42	2	4	0.31	0.61
Crystal River-3	8	28	0.33	1.14	7	7	1.18	1.18
Rancho Seco	4	16	0.08	0.34	6	8	0.66	0.88
ANO-1	6	24	0.12	0.47	2	2	0.28	0.28
TMI-1	3	6	0.05	0.11	-	-	-	-
TMI-2	1	3	0.35	1.04	-	-	-	-
TOTAL	84	207	0.23	0.56	23	31	0.48	0.65

A - Trips that are affected by setpoint changes and addition of anticipatory reactor trip (e.g., high-pressure trips, feedwater upsets, turbine trips)

B - Total trips: Category A trips plus those not affected by changes (e.g., total loss of feedwater, power to flow, test trips, etc.)

Note: Data for table through 1979.
Note: Through January 31, 1980, there have been 10 changes to the Control-Grade Anticipatory Reactor Trip System with 9 successes.

Design: 400 trips for 40-year life
10 trips/year or 0.83 trips/mo.

Pre-TMI-2 Setpoints: DB-1 and CR-3 exceeded 0.83 trips/mo.

Post-TMI-2 Setpoints: Rancho Seco and CR-3 exceeded 0.83 trips/mo.

Increase in Trip Frequency: A - 0.23 to 0.48 (approx. factor of 2)
B - 0.56 to 0.65 (approx. 15% rise)

Possible Causal Factors:

1 - Short period of time operating
2 - Many startups and shutdowns
3 - Operator familiarization
4 - Statistical Variations

TABLE 4.2

TRANSIENT EVENT SUMMARY FOR B&W OPERATING PLANTS

Event	Pre-TMI-2	Post-TMI-2
<u>Reactor Trips</u>:		
Total	232	38
Trips with documented PORV openings	149	1
<u>Reactor Trips caused by</u>:		
Feedwater transients	88	15
Turbine trip	41	12
<u>Effect of NNI/ICS Failures (29 total)</u>:		
Feedwater transients	13	6
Reactor trip	18	3
PORV opening	17	1
Engineered safety features actuation	3	1

5. B&W REACTOR SYSTEMS AND OPERATIONS

5.1 Introduction

The B&W nuclear steam supply system (NSSS) differs from other pressurized water reactors (PWRs) with regard to the design of the steam generator and with regard to the design of the control system. The once-through steam generator (OTSG) design is intended to provide the capability for the NSSS to respond acceptably to load changes when electrical grid conditions change as well as during daily load following cycles. Because of this desired responsiveness, the B&W control system design includes feed-forward features that result in immediate signals being provided to the reactor, feedwater control systems, and the turbine control valves in response to a change in the power demand signal. The control system provided in B&W plants to perform this function is the integrated control system (ICS). The basic requirement of the ICS is to match actual megawatts of electricity generated to the demand for electric power. The ICS meets this objective by controlling steam flow to the turbine, feedwater flow to the steam generator, and reactor power.

The safety features designed to mitigate the effects of transients and accidents are essentially the same for all PWRs. Namely, each PWR design includes a reactor protection system (RPS) for providing a reactor trip to limit the course of off-normal events; an emergency core cooling system (ECCS) to provide inventory in the event of a loss-of-coolant accident in the primary system. The auxiliary feedwater system (AFWS) also performs the same function in all PWRs, namely, to provide feedwater to the steam generators for the removal of decay heat when the main feedwater system is not available and to provide a method for heat removal and subsequent reactor coolant system depressurization in the event of a small loss-of-coolant accident. However, there are certain design differences in the B&W plants compared to the other PWR designs:

(1) The reactor trip signals provided in B&W designs do not include signals initiated from steam generator secondary side parameters;

(2) The high-pressure injection (HPI) pumps which make up part of the ECCS are capable of delivering coolant to the RCS at pressures exceeding the reactor coolant (RCS) safety valve setpoint. Davis-Besse 1 is the only B&W plant without this capability; and

(3) The AFWS for some B&W plants do not include initiation signals based upon steam generator secondary side parameters.

Some elements of the strengths and weaknesses of these design differences, as they affect the response characteristics of the RCS for the B&W design, are discussed in Sections 5.2 and 5.3.

The role of the operator in responding to off-normal events for B&W-designed plants is similar in nature to that performed at any PWR plant. Namely, his immediate actions are to verify that all required automatic actions have been initiated following an off-normal event and then to take those subsequent actions necessary to bring the reactor to stable hot shutdown condition.

Section 5.4 provides a discussion of the training and qualification of a B&W reactor operator considering any special requirements resulting from the design of the NSSS.

Selected figures showing both the lowered-loop (TMI-2) and raised-loop (Davis-Besse 1) designs for the 177-Fuel Assembly plants, as well as typical pressurizer and once-through steam generator designs, are included as Figures 5.1 through 5.6 of this report.

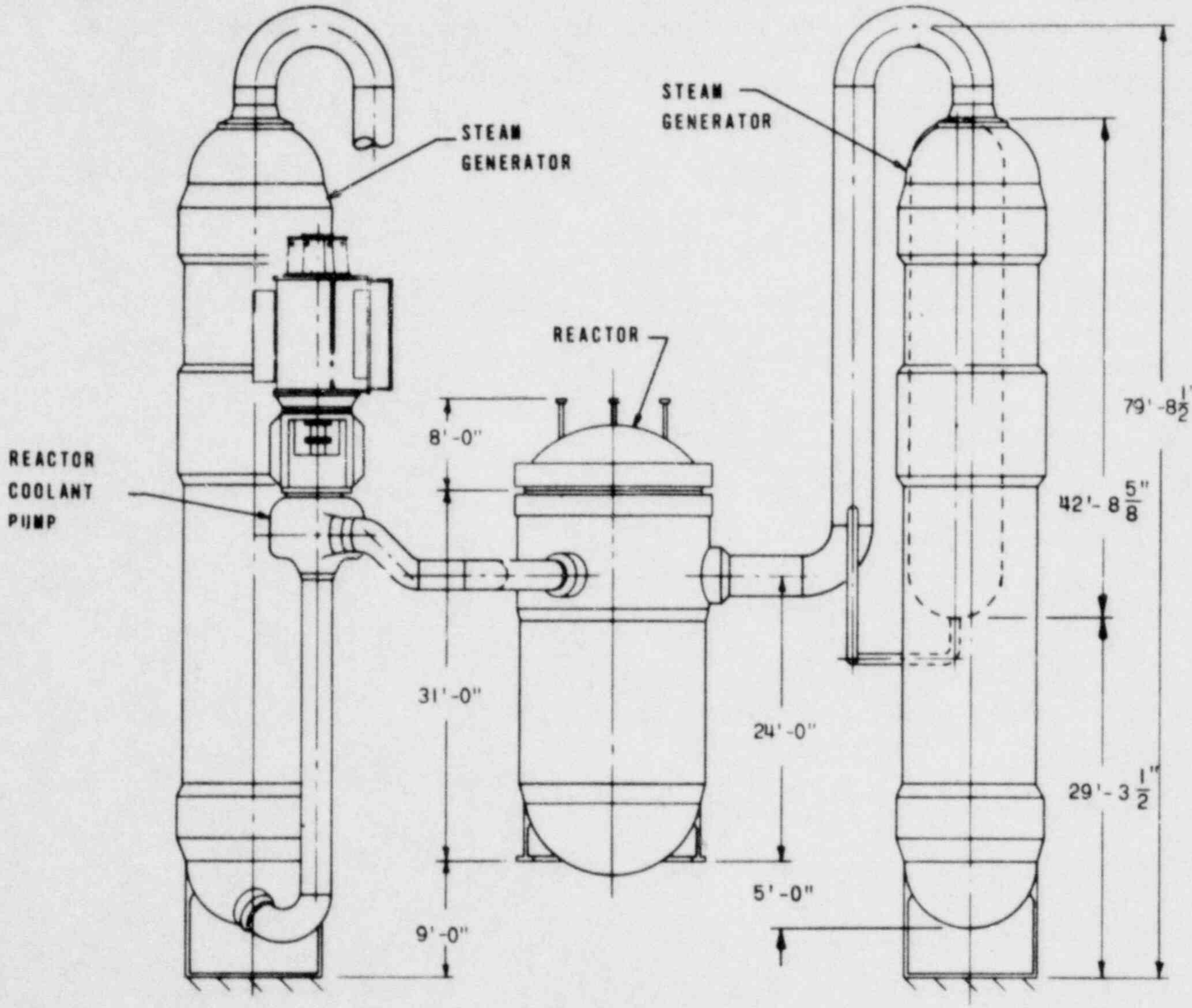

Figure 5.1. Reactor coolant system arrangeme t - elevation, from Three Mile Island, Unit 2, PWR.

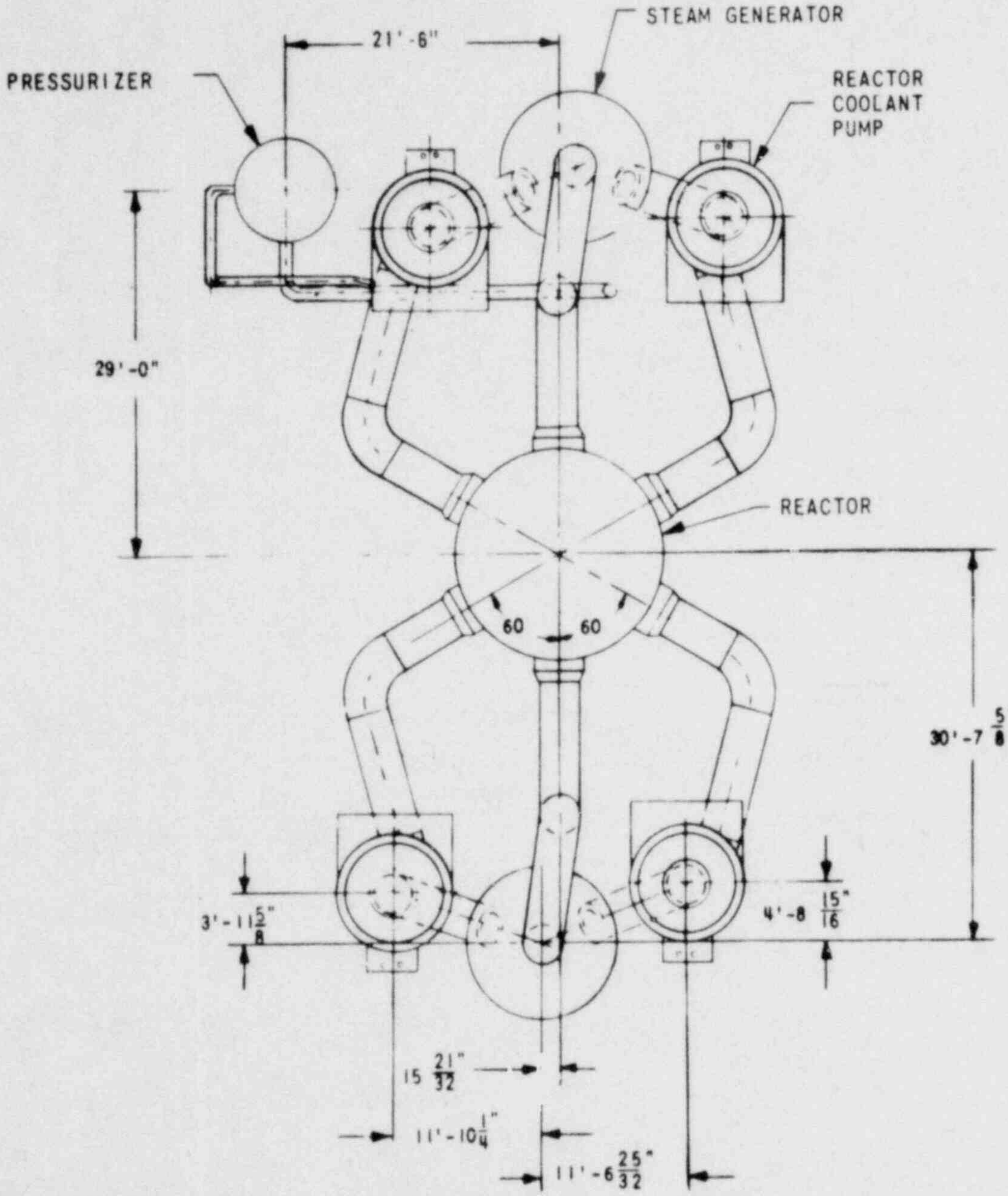

Figure 5.2. Reactor coolant system arrangement - plan, from Three Mile Island, Unit 2, FSAR.

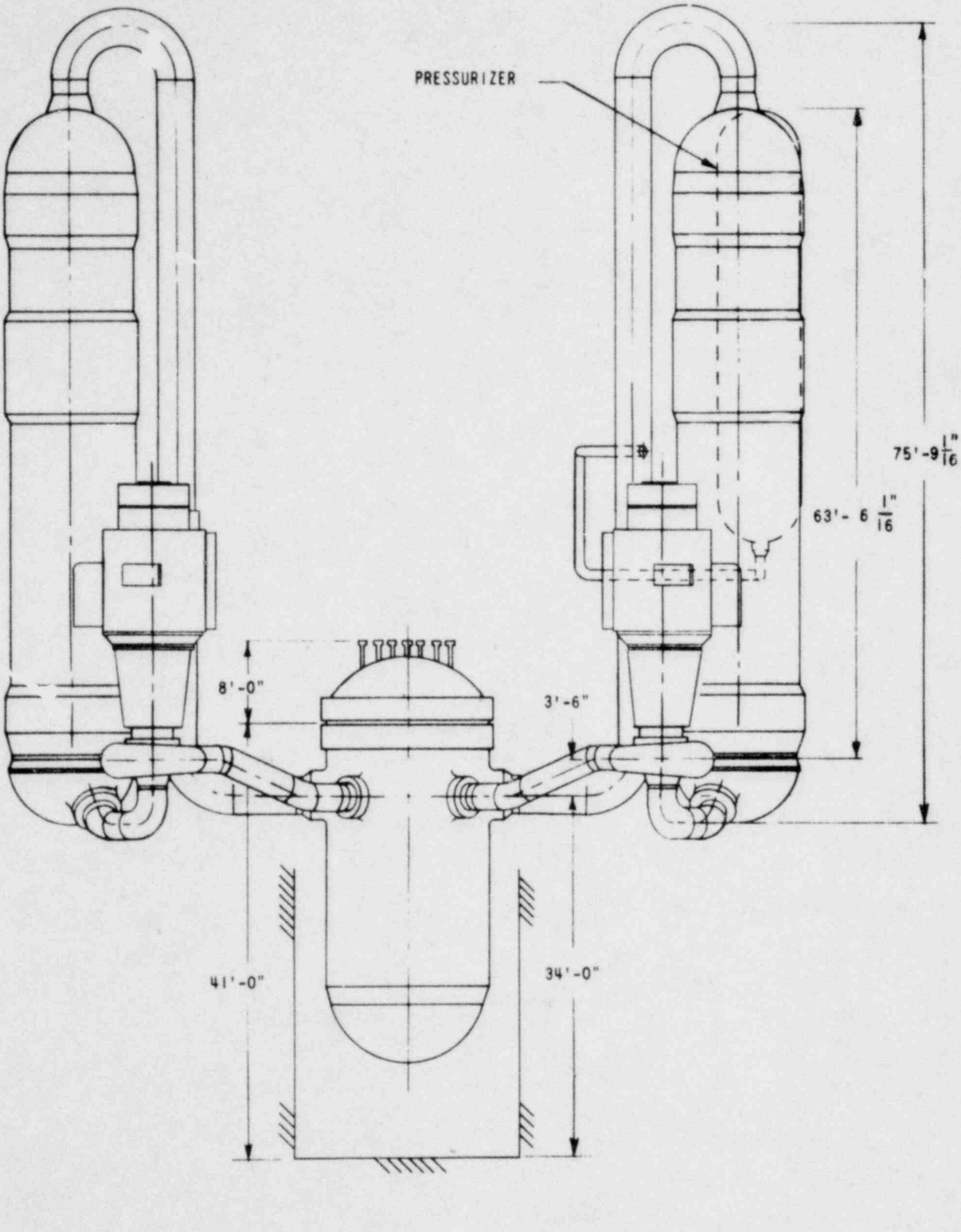

Figure 5.3. Reactor coolant system arrangement - elevation, from Davis-Besse, Unit 1, FSAR.

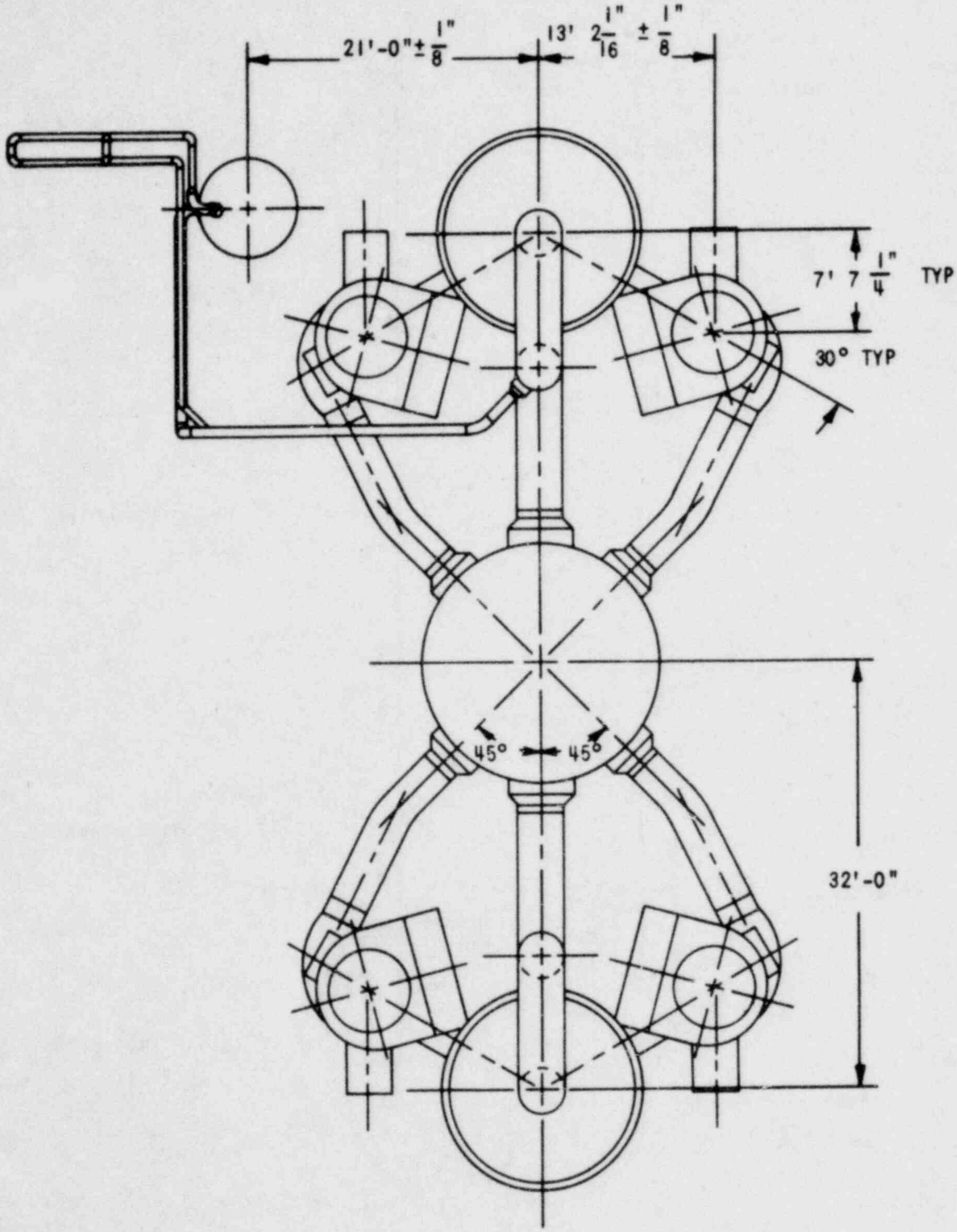

Figure 5.4. Reactor coolant system arrangement - plan, from Davis-Besse, Unit 1, FSAR.

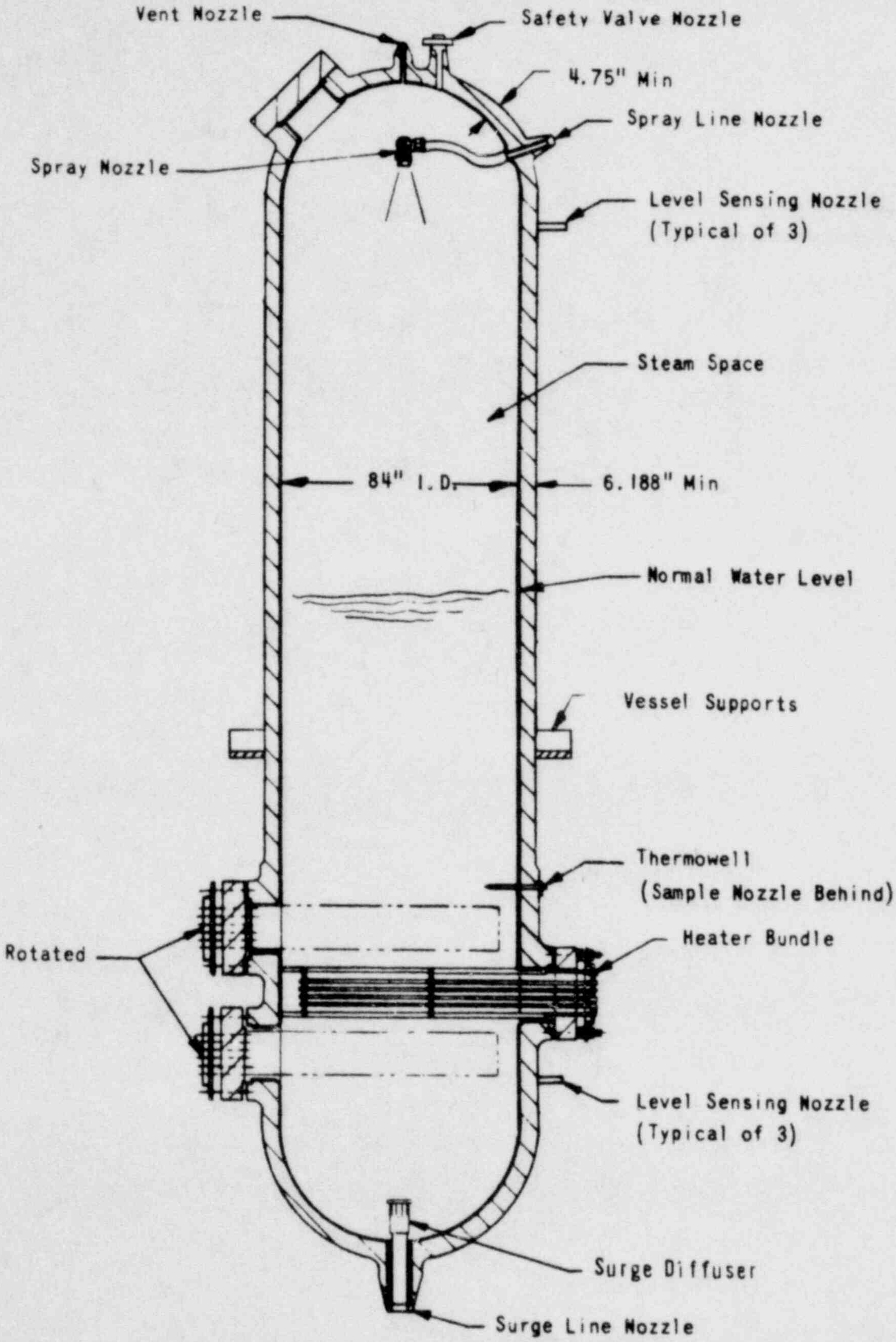

Figure 5.5. Typical B&W pressurizer.

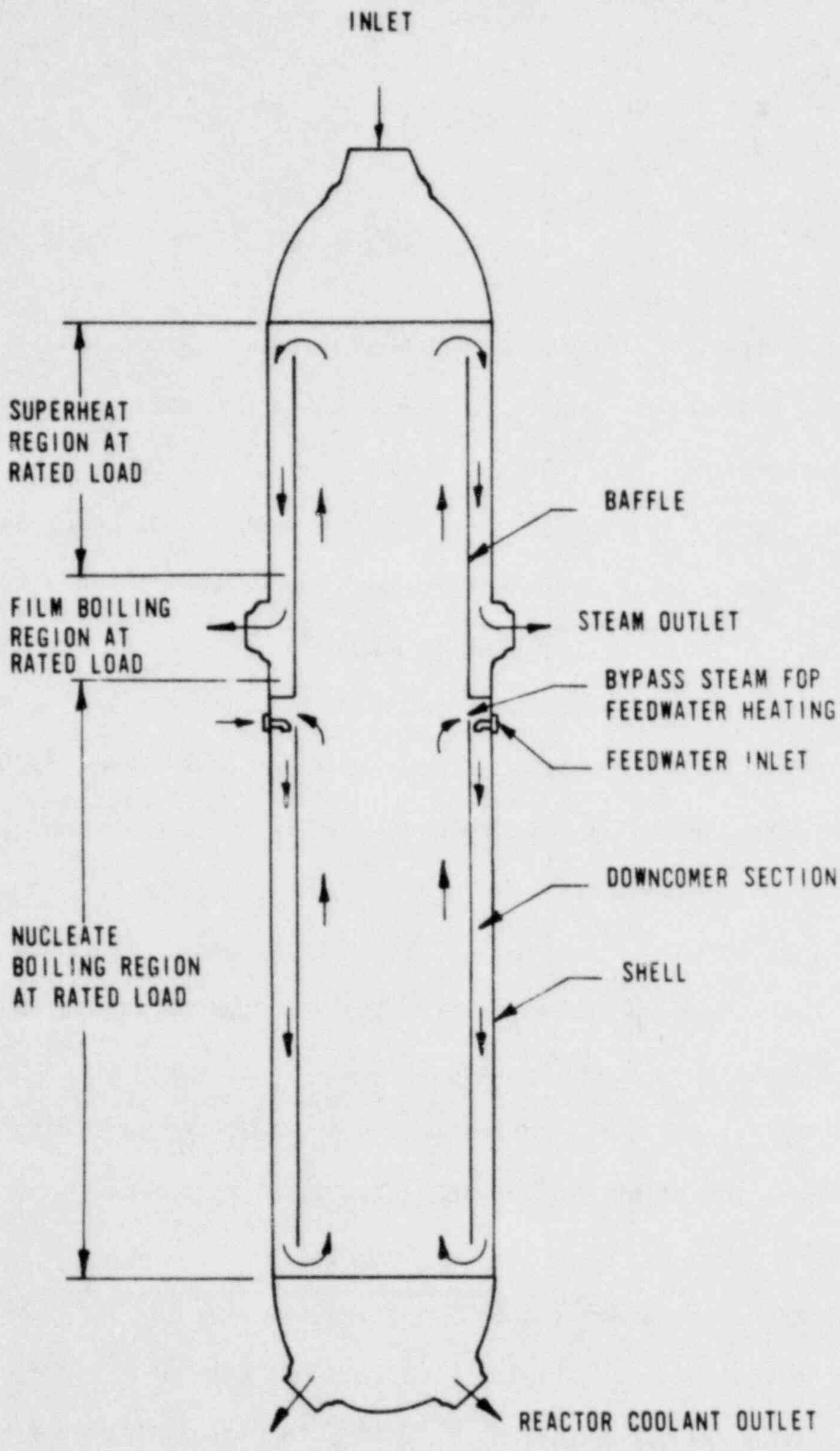

Figure 5.6. Typical B&W once-through steam generator,

5.2 Mechanical Design and Operational Considerations

5.2.1 Pressurizer Design

5.2.1.1 Discussion

Circulating systems in which temperature changes can occur usually require a gas volume to accommodate density changes in the coolant. In the pressurized water reactor design, this function is accomplished by the pressurizer (see Figure 5.5). Typically, the pressurizer is operated with about one-half of its volume filled with liquid and the remaining volume filled with steam. When the rest of the primary system is subcooled (no voids), the measured level in the pressurizer is a clear (and only) indication to the operator of the coolant inventory. The pressurizer level response after reactor trip is determined by the secondary side response as follows:

(1) Following reactor trip, steam flow to the turbine is stopped. Prior to opening the turbine bypass valves, the secondary pressure quickly builds up and opens the secondary side safety valves. This sets the secondary side temperature at the saturation temperature corresponding to the secondary side safety valve setpoint.

(2) The primary system hot leg and cold leg temperatures will collapse together once the control rods are inserted into the core and primary coolant temperature will drop to that necessary to establish the primary to secondary temperature differential for decay heat removal. This temperature is within a few degrees of the secondary temperature.

(3) The turbine bypass valves come open and begin to control steam pressure. By controlling secondary side pressure the secondary side temperature remains at the saturation temperature corresponding to this pressure, and thus primary side temperature is controlled.

(4) The pressurizer level decrease after reactor trip is primarily determined by the total temperature drop associated with the primary side (i.e., the density change of the coolant caused by the cooldown). The system pressure after reactor trip is likewise determined by the primary system shrinkage and the expansion of the steam bubble in the pressurizer associated with this shrinkage.

In Table 5.1, a comparison of some system parameters relevant to pressurizer design is made. With regard to pressurizer dynamic response, and in particular level response, it is seen from Table 5.1 that a combination of factors all tend to indicate that level response in B&W plants is expected to be more pronounced than in Westinghouse or Combustion Engineering plants. These factors are:

(1) The ratio of pressurizer volume to volume of fluid at T_H (hot leg) is smallest for B&W plants. Shrinkage of hot coolant following reactor trip will thus have a more significant effect on pressurizer level.

(2) The specific pressurizer volume is smallest for B&W plants, meaning the pressurizer level decrease will be greatest per cubic foot of coolant volume shrinkage for B&W plants.

Table 5.1

COMPARISON OF RELEVANT SYSTEMS PARAMETER

Parameter	B&W	CE	W 3-Loop
Total system volume, ft^3 (including pressurizer)	11,699.6	11,075	9360
Total thermal power, MWth	2452	2700	2785
Hot leg temperature, °F	604.8	600	618
Cold leg temperature, °F	555.2	550	554
Core temperature rise, °F	49.5	50	64
Volume of fluid at T_H, ft^3	3026.7	2780	2212
Pressurizer and surge line volume, ft^3	1525.2	1540	1482
Ratio of pressurizer volume to total system volume	0.130	0.139	0.158
Ratio of pressurizer volume to volume of fluid at T_H	0.504	0.554	0.67
Ratio of volume of fluid at T_H to total system volume	0.259	0.251	0.236
Pressurizer height, ft	~45	~37	38
Pressurizer volume, ft^3/ft	~38.5	~41.7	38

One of the characteristics of the B&W system that has recently come under scrutiny is the temporary loss of pressurizer level indication which sometimes occurs after reactor trips. This level loss is primarily caused by the contraction of primary the coolant in response to the secondary side temperature associated with the secondary side control. Level recovery is usually achieved when the coolant temperatures stabilize and then rise slightly and makeup pumps can begin to recover the system inventory. An example of this behavior is shown for a typical reactor trip transient for the Davis-Besse plant in Figure 5.7. B&W has pointed out that, for all of the cases where indicated pressurized level was lost, the pressurizer had not been calculated to completely drain in any of the cases.

5.2.1.2 Conclusions and Recommendations

(1) Conclusion:

During normal reactor trip, there is a tendency for B&W-designed plants to lose pressurizer level indication and depressurize to near or below the ESFAS (engineered safety features actuation system) setpoint.

Historically, operators at B&W plants have a tempted to dampen the system response by securing letdown flow and starting a second makeup pump. There are some indications that occasionally operators have also throttled back feedwater and/or actuated HPI (high-pressure injection) to reduce the amount of primary system depressurization. It is the belief of this Task Force that the loss of pressurizer level, along with the need for operator actions of the kind described, places the plant in an undesirable condition and should be remedied.

Recommendation:

Following a reactor trip, pressurizer level should remain on scale, and system pressure should remain above the HPI actuation setpoint. The system response (e.g., secondary pressure) should be appropriately modified in order to meet the above two objectives. Meeting these objectives should be independent of all manual operator actions (e.g., control of feedwater, letdown isolation, and startup of a makeup pump).

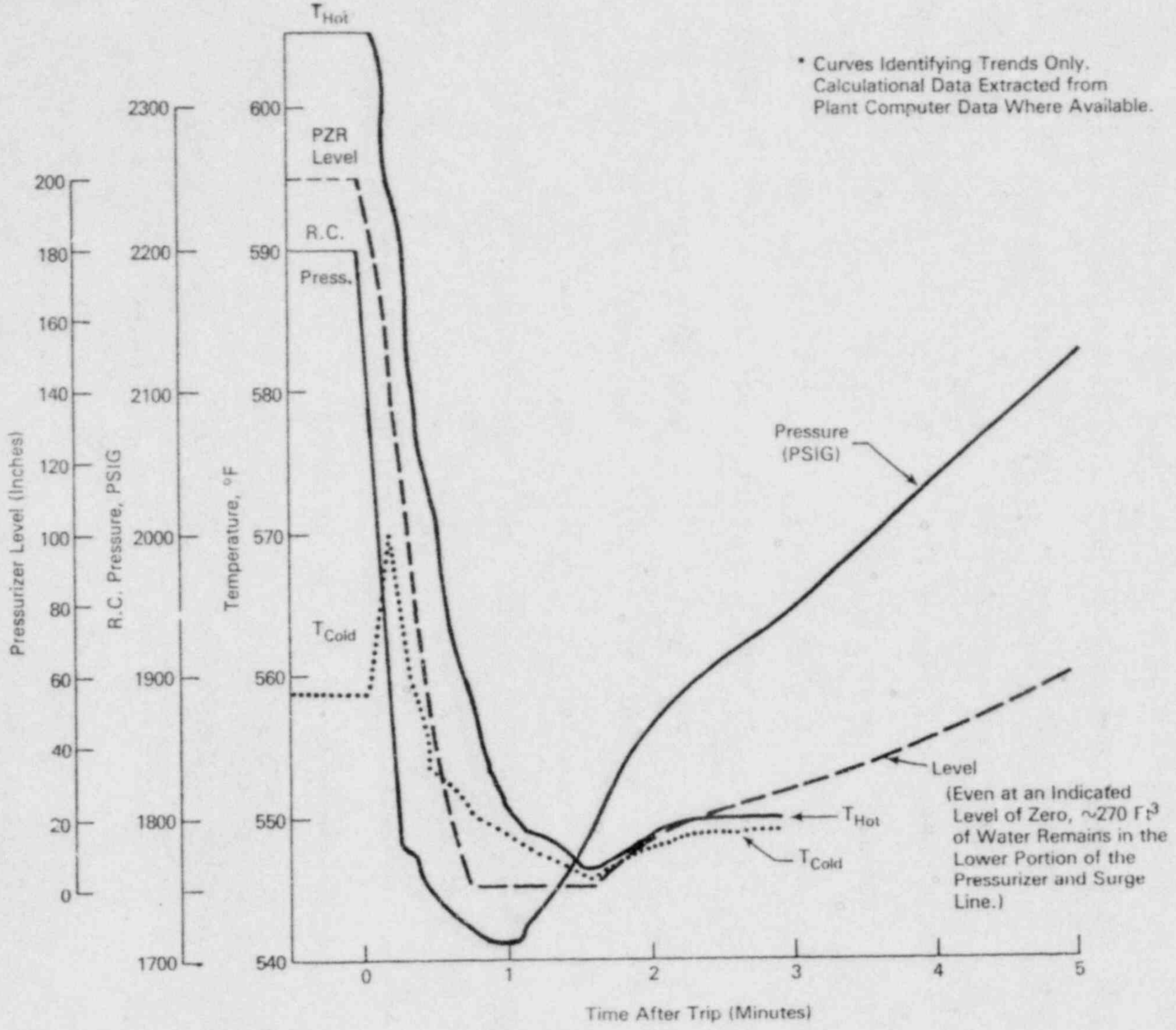

Figure 5.7. System parameters for Davis-Besse 1 reactor trip September 18, 1979.

5-15

5.2.2 Once-Through Steam Generator (OTSG)

5.2.2.1 OTSG Design

The steam generator design employed by Babcock & Wilcox is called a once-through design, referring to the fact that primary coolant rejects heat to the secondary coolant during a single pass through the unit (see Figure 5.6). In the OTSG design, the tubes are only partially covered with secondary coolant. In addition, the amount of secondary coolant in the steam generators is considerably less than in an inverted U-tube steam generator design. Because the tubes are only partially covered with secondary liquid, steam generated by the boiling of secondary liquid is superheated as it contacts the tubes in the upper 10 percent of the steam generator since that area of the tubes contains primary coolant closest to the core outlet temperature. The majority of the core heat is used to raise the feedwater to saturation temperature and to boil it to steam. Thus, most of the core heat is removed in that region of the steam generator tubes covered with liquid or a two-phase mixture, which is approximately 75 percent of the tube heat transfer area. Because the heat removed is proportional to the heat transfer area, the amount of heat removed by an OTSG is essentially directly proportional to the height of liquid on the secondary side. As such, any change in secondary coolant level directly affects the amount of heat capable of being removed. This, coupled with the relatively smaller secondary side liquid inventory, results in a fairly rapid primary system response to secondary coolant system perturbations.

5.2.2.2 U-Tube Steam Generator Design

In a U-tube steam generator, such as the kind presently used in Westinghouse and Combustion Engineering plants, the primary coolant flows through inverted U-tubes that are covered by the secondary coolant. Saturated steam is produced by the pool boiling of secondary coolant caused by heat transfer from the primary to the secondary coolant. Under normal operating conditions, the inverted U-tubes are designed to remain covered with secondary coolant.

The ability of the steam generator to transfer heat is determined by three parameters: the temperature difference between the primary and secondary coolants, the area available for heat transfer, and the coefficient of heat transfer (capability to conduct heat). In an inverted U-tube steam generator, because the tubes are covered completely, the coefficient of heat transfer is high, and both the coefficient of heat transfer and heat transfer area are relatively constant regardless of power level. Since the heat removal rate is proportional to the product of the heat transfer coefficient, heat transfer area, and temperature difference, and because the product of the heat transfer area and heat transfer coefficient is usually high, only small changes in the primary to secondary temperature difference are needed to accommodate rather large changes in the heat removal rate. Because of this and because the volume of water on the secondary side surrounding the U-tubes is large, perturbations on the secondary side of the inverted U-tube steam generator, such as feedwater flow changes or system pressure changes, do not readily affect the behavior of the primary coolant system.

5.2.2.3 Operational Advantages Associated with OTSG Design

Before discussing in detail the performance characteristics of an NSSS with an OTSG design, it is of benefit to understand the operational advantages which make the OTSG a desirable design. One of the most desirable features of an OTSG is the capability to produce superheated steam which results in lower moisture in the turbine and longer turbine life, an obvious economic advantage. Moreover, this superheat produces a slight increase in plant efficiency, also a desirable economic advantage. Operational experience has also indicated favorable tube integrity in the OTSG design compared to inverted U-tube design, and can, in part, be attributed to the low secondary side water inventory and associated lower contaminant concentration. These benefits, however, are obtained at the cost of a system highly responsive to secondary side perturbations.

5.2.2.4 Conclusions and Recommendations

(1) Conclusion:

The Task Forces recognizes that the OTSG has certain operational advantages with respect to tube integrity and steam properties that make it an attractive design. However, we conclude that other characteristics of the design result in a system that is highly responsive to secondary side flow perturbations. Specifically, the relatively small volume of the secondary coolant, together with the rapid change in heat transfer area with variations in coolant level in the OTSG, result in conditions that produce significant mismatches between the heat generated in the nuclear core and the heat removed by the OTSG during anticipated transients.

These mismatches are reflected into the RCS, and cause primary coolant volume variations and pessure change that result in unnecessary challenges to pressure relief devices or the engineered safety features. The Task Force understands that these are efforts under way in the industry to investigate means to improve the response of the OTSG to secondary coolant perturbation. We endorse and encourage these efforts.

Recommendation:

The Task Force recommends that licensees be required to perform sensitivity studies of possible modifications to reduce the response of the OTSG to secondary coolant flow perturbation. Specifically, we recommend that passive and active measures be investigated to mitigate overcooling and undercooling events.

5.2.3 NSSS Performance Characteristics with OTSGs

There are two basic types of secondary side perturbations that affect the primary system. These are events that overcool the primary system (remove more heat than is being generated in the core) or undercool the primary system (cannot remove all of the heat being generated in the core).

5.2.3.1 Undercooling Events

Undercooling events initiated from the secondary side usually involve either a reduction in or loss of feedwater flow to the OTSGs. For loss of feedwater (LOFW) events, all of the PWR vendors, prior to the TMI-2 accident, calculated on a conservative basis that power-operated relief valves (PORVs) would open due to high reactor coolant system pressure during the early stage of the transient.

Because a B&W steam generator holds about 27 to 30 full power seconds (FPS) worth of inventory compared to Westinghouse or Combustion Engineering steam generators that hold approximately 90 FPS worth of inventory, the primary system of B&W-designed plants pressurize faster and reach the PORV setpoint sooner during a LOFW event. B&W has calculated (Ref. 24) that it will take about 8 seconds after a LOFW to reach the high-pressure reactor trip setpoint. Westinghouse (W) (Ref. 25) and Combustion Engineering (CE) (Ref. 26) predict trip setpoints on low steam generator secondary side level will be reached for LOFW at about 20 seconds and 17 seconds, respectively. For Westinghouse and Combustion Engineering the primary system pressure rise prior to reaching these setpoints is negligible. These results are indicative of the more responsive nature of B&W plants to undercooling events. Since the TMI-2 accident and inversion of the PORV and reactor trip setpoints on B&W plants, the responsiveness to undercooling events leads to a lower challenge rate to overpressure relief devices (both PORVs and safety valves); however, it now reflects a high challenge rate to the plant protection system.

Undercooling events in B&W reactors, prior to TMI-2, usually resulted in lifing the PORV and discharging primary coolant to the pressurizer quench tank. Reactor trip on high pressure was usually precluded because the trip setpoint was purposely set higher than the PORV actuation setpoint to allow the plant to "ride through" loss of load events without reactor trip. Although most of the recorded challenges to PORVs are from B&W plants, these types of events have also been known to challenge the PORVs on both Westinghouse and Combustion Engineering-designed plants. Because PORV actuation is not in itself a reportable occurrence, the data base regarding the number of PORV challenges to date is incomplete (Ref. 27). Since TMI-2, the setpoints for the PORV actuation and reactor trip have been inverted on plants with B&W reactors. This action was taken to reduce the number of challenges to the PORV and hence reduce the probability of a PORV failure leading to a loss-of-coolant accident. This action however, consequently increases the number of challenges to the reactor protection system. This was recognized by the ACRS (Ref. 28), and they recommended a continued evaluation of this action on plant safety. In addition to the above action, two anticipatory reactor trips were added that will trip the reactor on turbine trip or loss of feedwater. This was also done to reduce the number of challenges to the PORVs.

Despite the actions just mentioned, there still remains an underlying problem resulting from failure related to the integrated control system (ICS). These failures involve the severe degradation of the feedwater (e.g., control valve closure, main feedwater pump runback) without producing an anticipatory reactor trip. The resultant dryout of the steam generator and loss of heat sink would produce a reactor trip on high system pressure. The startup of the auxiliary feedwater pumps and the rapid introduction of relatively cold feedwater into the upper elevations of the steam generator overcools the primary system and can produce a rapid depressurization transient which may result in actuation of ESFAS on low primary system pressure. Overcooling and its effect on primary system behavior is discussed below.

5.2.3.2 Overcooling Events

Depressurization of the primary system results from secondary side overcooling. This overcooling usually occurs because of overfeeding a steam generator, demanding too much steam from the steam generators, or introducing excessive amounts of relatively cold auxiliary feedwater into the steam generator.

The depressurization of the primary system is caused by primary coolant shrinkage due to cooldown. During normal reactor trips, this depressurization is limited by the secondary side response designed to control secondary steam pressure and maintain the core average temperature at a minimum of 547°F. For B&W-designed plants, the primary system has typically been observed to depressurize to between 1700 and 1800 psig during reactor trips that are not compounded by feedwater upsets. For events that overcool the primary system in excess of the normal cooldown experienced during a trip, this minimum pressure will decrease further and could reach the ESFAS actuation setpoint during some events. Primary system behavior for a trip at the Davis-Besse plant is shown in Figure 5.7.

Typical reactor trip transients in Westinghouse-designed plants result in a lesser primary system depressurization (about 2000 psig, or slightly below). Although newer Westinghouse plants typically operate with a higher differential temperature across the core than B&W plants (64°F versus 50°F) and at a higher core outlet temperature (~618°F versus ~605°F), the minimum pressure reached after reactor trip is typically higher.

Very little pre-TMI-2 information for plants designed either by Westinghouse or Combustion Engineering is readily available concerning plant response to events that overcooled the primary system in excess of the normal cooling expected following a reactor trip. It should be noted, however, that since TMI-2, three events that depressurized the primary system to the HPI actuation setpoint have occurred in plants with reactors designed by Westinghouse and Combustion Engineering. Two of these events involved stuck-open turbine bypass valves and one was the result of a steam generator tube rupture.

Overcooling events analyzed in safety analysis reports are only concerned with demonstrating acceptability with respect to the minimum departure from nucleate boiling ratio (MDNBR) and the overpressure limit. Because of this limited concern, these events are never carried out more than about 100 seconds into the event, and it cannot be determined from the overcooling event analyses if the HPI actuation setpoint would be reached. However, any sustained overcooling event would be expected to depressurize the primary system to the HPI actuation setpoint for both Westinghouse and Combustion Engineering designed plants.

If an overcooling event depressurizes the primary system excessively, then the HPI will be actuated and emergency core cooling water will be injected into the primary system. For all B&W operating plants, except Davis-Besse, the HPI pumps (which are the charging pumps in the injection mode alignment) are capable of injecting water at pressures above the PORV and safety valve setpoints. Thus, operator action is required to throttle the HPI flow in order to prevent the primary system from going water solid. The need for the operator to throttle the HPI pumps does not pose a direct safety concern, since the consequence of operator failure to throttle back the HPI flow does not directly challenge core integrity. However, the rapid depressurization of the primary system as a result of reactor trip, as well as steam generator overcooling events, has evolved a set of questionable operator responses having the potential to compound minor events into more serious ones. For any non-LOCA event in which HPI is actuated, the injected HPI water contains a high concentration of boron that must be removed from the primary coolant prior to restarting the plant. This is a time-consuming process, so it is highly desirable to either prevent or terminate HPI actuation as soon as possible. Thus, for reactor trips and other non-LOCA overcooling transients, the operators at B&W plants have historically secured letdown and started a second makeup pump in an attempt to mimimize the depressurization. There is also evidence that feedwater may be throttled to minimize the depressurization. Prior to the TMI-2 accident, HPI actuations were routinely terminated rapidly because the operators assumed the actuation was not due to a LOCA but rather to system response to overfeed events or even spurious actuation. Some operators would even manually actuate HPI to arrest the pressure decay, which would subject the HPI nozzles to an accountable stress cycle. Thus, although HPI actuation in itself does not result in a safety concern, the various operator actions precipitated by the actuation could ultimately place the plant in an unsafe condition.

5.2.3.3 Conclusions and Recommendations

(1) Conclusion:

Auxiliary feedwater in lowered-loop plants is delivered to the steam generators through an auxiliary feedwater spray ring in the upper elevations of the steam generator. This spray ring sprays auxiliary feedwater into the steam generator tubes and signficantly increases the heat transfer rate in the steam generator. This increased heat transfer rate enhances the ability of the auxiliary feedwater to rapidly depressurize the primary system. The reason for introduction of auxiliary feedwater through a spray sparger into the tubes is to raise the thermal center in the steam generator and promote natural circulation. The need to provide this amount of cooling for natural circulation may not be necessary during expected transients, and the excessive depressurization may be detrimental.

Recommendation:

The need to introduce auxiliary feedwater through the top spray sparger during expected transients should be reevaluated by licensees. This reevaluation should consider the reduced depressurization response if auxiliary feedwater could be introduced through the main feedwater nozzle and enter the tube region from the bottom of the unit.

(2) Conclusion:

The present acceptance criteria for anticipated transients require only that no fuel damage occur and overpressure be limited to 110 percent of design pressure. No fuel damage is demonstrated by showing that the departure from nucleate boiling ratio (DNBR) is maintained above a minimum value. Although these criteria are acceptable for providing the necessary assurance that fuel and primary system boundary integrity are maintained during the initial part of the transients, they do not provide criteria for determining acceptability of overall system response. In addition, they do not provide acceptance criteria against which the overall transient response taken to its safe conclusion (e.g., hot shutdown), including necessary operator actions, can be judged. At present, the staff is performing a generic review of transients and other accidents in accordance with Recommendation 2.1.9 of NUREG-0578 (Ref. 13). However, the Task Force does not feel this requirement goes far enough.

The Task Force concludes that a plant performance criteria for anticipated transients should be established. The purpose of developing criteria for plant performance during anticipated transients is to assure an acceptable degree of uniformity in plant transient response for all light water reactors and consequent uniformity in overall risk to public health and safety from anticipated transients. Through demonstrating conformance to these criteria, an acceptable degree of uniformity in plant transient response is attainable and quantifiable. The Task Force has identified the following four areas in which criteria should be established to meet the above goal. These areas are (1) heat sink availability, (2) frequency of transient initiators, (3) availability of operator information, and (4) overall plant response including operator action. The development of specific criteria in these areas is considered to be beyond the scope of this Task Force. We recommend that a program be established within NRR to develop these acceptance criteria. This program should be included with that given in Section V.1 of the TMI-2 Task Action Plan (Ref. 7). Other areas should be determined by the extensive use of the Interim Reliability Evaluation Program (IREP), ongoing risk assessment work, and other appropriate sources. In addition, the staff believes that a significant contribution to the development of these performance criteria can and should be made by the nuclear industry through such organizations as NSAC, INPO, and EPRI.

Although the development of performance criteria must be the product of extensive evaluation and review, the Task Force offers the following preliminary example that should be considered in order to focus attention on the overall goal to be achieved. This example is not to be considered a specific recommendation of the Task Force.

EXAMPLE

As a design basis, the following criteria for plant response during anticipated operational transients shall be met:

(a) Heat sink capacity shall be established such that availability is assured for ____ minutes following the loss of all feedwater (main and auxiliary) with no other failures,

(b) No failure of a control function should lead to the actuation of an engineered safety feature (e.g., inadvertent opening of a PORV, overfeeding of a steam generator, containment isolation, etc.), and

(c) The plant should achieve stable safe shutdown conditions following anticipated transients without reliance on operator action.

The Task Force points out that the establishment and implementation of such criteria could ultimately require extensive design or design concept changes to certain operating plants or could ultimately lead to the shutdown of some plants, and thus must be carefully and thoroughly evaluated in terms of overall reduction in risk to the public health and safety.

Recommendation:

The Task Force recommends that a performance criteria be established for anticipated transients for all light water reactors. We recommend that such criteria cover as a minimum the following four areas: (1) heat sink availability, (2) frequency of transient initiators, (3) availability of operator information, and (4) overall plant response including operator action.

(3) Conclusion:

Based on the operating history of B&W-designed reactors since the TMI-2 accident, the inversion of the reactor high pressure trip setpoint with the PORV setpoint (as required in IE Bulletin 79-05B) has resulted in an increased challenge rate to safety systems.

Recommendation:

To provide an alternative solution to PORV unreliability and safety system challenge rate concerns, the following proposal (submitted by

Consumers Power Company) should receive expeditious staff review for possible consideration and backfit on all B&W operating plants:

(a) Provide a fully qualified safety-grade PORV;

(b) Provide reliable safety-grade indication of PORV position;

(c) Provide dual safety-grade PORV block valves, capable of being automatically closed if a PORV malfunction occurs;

(d) Conduct a test program to demonstrate PORV operability;

(e) Install a safety-grade anticipatory reactor trip on total loss of feedwater; and

(f) Reset the PORV and high-pressure trip setpoints to their original values of 2255 psig and 2355 psig, respectively.

5.2.4 Effect of Reactor Coolant Pump Trip

5.2.4.1 Discussion

In July 1979, IE Bulletins 79-05C (B&W) and 79-06C (CE & W) were issued. These bulletins required, in part, that all operating reactor coolant pumps (RCPs) should be tripped upon reactor trip and actuation of the HPI on low system pressure. The reason for requiring this action was based on PWR vendor analyses which showed that, for a range of small break sizes in the primary system and a range of times into the accident in which the pumps were assumed to be tripped, severe core uncovery leading to cladding temperatures in excess of licensing limits were predicted. The basic cause of this severe core uncovery was that RCP operation continued to supply liquid to the break throughout the accident. A more complete description of the phenomenon is described in NUREG-0623 (Ref. 12). Although the analysis models themselves were suspect because they had not been verified, the analyses only showed unacceptable results when some conservatisms were imposed on the analyses. As a result, the desirability of tripping the RCPs has been continually questioned, in particular the effect of tripping the pumps for non-LOCA transients, which actuate HPI on low pressure.

On March 4, 1980, the staff presented to the ACRS the effect experienced in PWRs of tripping RCPs during non-LOCA transients which actuated HPI. The conclusions reached from this experience are as follows:

(1) During the steam generator tube rupture at Prairie Island, tripping of the RCPs eliminated the ability of the operator to utilize pressurizer spray in order to help depressurize the reactor coolant system to minimize coolant leakage out of the break.

(2) RCP restart criteria during recovery from a non-LOCA transient are needed, since pump restart restores pressurizer spray capability for pressure control. This restart criteria was being developed as part of procedure development for "Transients and Accidents" as identified in I.C.1(3) of the TMI-2 Task Action Plan (Ref. 7).

(3) Proposed signals for automatic RCP trip (low pump motor current, low system pressure) should preclude RCP trip for most non-LOCA depressurization transients.

At this meeting, the staff also presented its plan for resolving the question of the need for RCP trip during small-break LOCAs. This plan included the requirement for PWR vendors to demonstrate the capability of their analytical models to properly predict system behavior during a small-break LOCA with the RCPs running by predicting LOFT Test L3-6, to be conducted in September 1980.

5.2.4.2 Conclusions and Recommendations

(1) <u>Conclusions</u>:

The Task Force agrees with the staff conclusion in NUREG-0565 (Ref. 4) that tripping of the reactor coolant pumps is not an ideal solution and that licensees should consider other solutions to the small break problem. Since the issuance of Bulletins 79-05C and 79-06C and the staff evaluation presented in NUREG-0623 (Ref.12), there has been no new information provided by industry that would encourage the Task Force to recommend a withdrawal of the requirements of Bulletins 79-05C and 79-06C at this time. Moreover, the Task Force sees no evidence which leads it to conclude that RCP trip, which could occur during non-LOCA transients and small-break LOCAs, will place the plant in an unsafe condition.

<u>Recommendations</u>:

(a) NRC should review the RCP restart criteria during recovery from non-LOCA transients, as provided in B&W small-break guidelines. Restarting the RCPs provides the operator with pressurizer sprays and thus greatly improves plant pressure control.

(b) The criteria for tripping the RCPs during small-break LOCAs should continue to be studied. To this end, the Task Force endorses the NSAC/INPO recommendation that the evaluation should be conducted jointly by both the industry and the NRC.

5.2.5 Accommodations of Loss of All Feedwater

5.2.5.1 Discussion

As part of the Bulletins and Order Task Force review, the capability of pressurized water reactors to accommodate a loss of all feedwater was examined. For plants with reactors designed by B&W, analyses prepared by B&W (Ref. 24) concluded that approximately 20 minutes is available after loss of all feedwater for the operator to either (1) restore feedwater (either auxiliary or main), or (2) start the HPI pumps and enter into a "feed and bleed" mode of core cooling. This available time is consistent with independent staff analyses (Ref. 4).

"Feed and bleed" involves the addition of coolant to the primary system by the HPI system, and the removal of the coolant mass added, along with decay heat, through "bleeding" off by either the pressurizer safety valves or PORV. The capability to cool the core using this mode of operation is not a requirement for any PWRs, and although the potential for successful core cooling in this mode is high, the system is not specifically designed for this mode of operation. For example, the safety valves and PORV have not been designed to relieve a two-phase or liquid discharge and it has not been determined if equipment needed to operate in this mode is properly designed to appropriate safety criteria. Moreover, this form of cooling ultimately results in significant quantities of primary coolant being pumped into the containment, which is an undesirable situation to be avoided if possible.

The B&O Task Force recognized the viability of "feed and bleed" for core cooling in the event of heat sink loss. However, it was obvious that this mode of cooling could only be achieved in plants which had HPI pumps with high discharge heads (i.e., above the safety valve setpoints). Some PWRs do not have high head HPI pumps and do not have the capability to enter into "feed and bleed" unless the primary system can be depressurized, through the PORV, to a pressure below the shutoff head of the HPI pumps. Davis-Besse 1 is the only plant with a B&W reactor that does not have high-head HPI pumps and must rely on PORV operation and capability to depressurize the primary system to that pressure at which the HPI can start to inject.

Finally, it was recognized that the capability to provide core cooling in the event of a loss of all feedwater could also be accomplished with a high-pressure residual heat removal system.

Recognizing the possible options, the Bulletins and Orders Task Force reports (Refs. 4, 5, 27, 29) concluded that a diverse decay heat removal path independent of the steam generators is desirable and recommended that the TMI-2 Action Plan (Ref. 7) consider the need for a diverse decay heat removal path. More recently in a letter to Chairman Ahearne (Ref. 28), the ACRS has also indicated the desirability of a diverse heat removal path and has established an Ad Hoc Subcommittee to review this matter.

Based on the Task Force's understanding of the present NRC and ACRS actions the capability of PWRs to accommodate a loss of heat sink, we find these actions acceptable and endorse their efforts.

5.2.5.2 Conclusion

In the event of loss of all feedwater (both main and auxiliary), core cooling using "feed and bleed" has a high potential for success. However, the capability to successfully cool the core using "feed and bleed" depends on the capability of relieving devices to accommodate a single-phase liquid or two-phase flow, as well as the discharge pressure of the HPI pumps. At present, the capability to remove decay heat by this method is not a design requirement.

Present actions identified (both staff and ACRS) appear to be sufficient to provide a timely resolution to this question, and no specific recommendation by the Task Force appears necessary.

5.2.6 Auxiliary Feedwater System (AFWS)

5.2.6.1 Concept and Concerns

Following a reactor trip of a pressurized water reactor, the steam generator serves as the primary heat sink for removal of heat from the reactor coolant system. Such heat removal is accomplished by the transfer of heat across the steam generator tube boundaries to the secondary coolant on the shell side of the steam generator, and it results in the production of steam that is removed from the steam generator and released to the atmosphere or the main condenser.

The inventory of secondary coolant removed in this process must be replaced. The AFWS provides the only source for this replacement when the main feedwater system becomes inoperable. The heat removal capability must be maintained to assure adequate core cooling following all anticipated transients.

As such, the AFWS must operate reliably over the period of time the plant is to be maintained in a hot standby condition and the time required to cool the reactor system to the temperature and pressure that will permit initiation of the low-pressure decay heat removal system. Second, the AFWS must be available on demand to adequately provide the needed water in the proper quantity to assure that neither undercooling not overcooling of the steam generator occurs. This is especially important in the B&W design since the steam generators have a smaller quantity of steam generator secondary coolant (OTSG inventory) compared to other pressurized water reactor designs. Therefore, fluctuations in feed delivery can result in large mismatches between heat generated in the reactor core and heat removed by the steam generators.

Such mismatches could lead to either of the following primary system transients:

(1) Loss or reduction of feedwater leads to overheating and overpressurization of the reactor coolant system (RCS) that can lead to challenges to safety systems and pressure relief devices through lack of ability to remove the heat generated by the reactor. This type of event also forces the RCS toward a saturated condition and can possibly prevent automatic actuation of high-pressure injection equipment when required if auxiliary feedwater is not actuated in a timely manner.

(2) Excessive feedwater causes an overcooling effect on the RCS that can lead to challenges of safety systems and possible loss of RCS subcooling control ability, as well as structural stress concerns due to the entry of water into the main steam piping system, and the structural fatigue of the HPI nozzles.

Since these types of events can lead to inadequate core cooling, it is imperative that heat sink control capability be adequately maintained at all times. Therefore, an AFWS of high reliability is extremely important.

Among the operating B&W plants, there are wide design differences in the AFWS including its initiation and control features. In general, the features of the AFWS include:

(1) The capability of providing auxiliary feedwater to one or both steam generators under automatic or manual initiation and control. In the past, the initiation and control devices have not met safety-grade standards.

(2) The AFWS consists of multiple feedwater trains with a combined capacity of twice the flow of a nominal full-capacity pump.

(3) The pumps usually consist of a full-capacity turbine-driven pump and either one full-capacity or two one-half-capacity motor-driven pumps. (However, it should be noted that variations in the type of pumps do exist.)

(4) Multiple suction sources are provided including the condenser hot well or other backup water supplies.

(5) Motive power for the motor-driven pumps can be obtained from a vital bus as a result of the post-TMI-2 review.

(6) Initiation of the AFWS is accomplished, in general, through the loss of both main feed pumps and at least one other event (e.g., loss of all four reactor coolant pumps). All AFWS initiation is battery-backed. Table 5.2 tabulates the various AFW system initiation signals on B&W operating plants.

(7) In general, most plants use the ICS to control flow of auxiliary feedwater to the steam generators. Challenges to the AFW system of operating B&W plants have been frequent because of the unreliability of the main feedwater systems and their associated control and support systems.

Even though the AFWS serves the safeguards function described in the preceding paragraphs, the systems in general have not been designed and constructed to safety-grade standard. In general, the initiation and control systems for these systems do not meet IEEE Standard 279-1971 "Criteria for Protection Systems for Nuclear Power Generating Stations" (Ref. 30). On some plants, the motive power for the pumps is not provided from diverse power sources. For example at Davis-Besse 1, both pumps of the AFWS are steam-driven units and, as such, a loss of steam pressure due to dryout of the steam generators could defeat the capability to deliver adequate auxiliary feedwater. There has also

TABLE 5.2

AFW AUTOMATIC INITIATION SIGNALS

Plant	AFW Pump Drive	Two MFW Pump Trip	ESFAS	Four RCP Trip	Two MFW Pump Lo Disch Press	Two SG Lo Level	One SG Lo Level	One MFW Valve Hi Rev ΔP	2 MFW Pump Lo ΔP
Rancho Seco	Turbine	X	X	X					
	Motor	X		X					
Oconee 1, 2, 3	Turbine	X			X				
	Motor	X			X				
Crystal River 3	Turbine	X				X			
	Motor	X				X			
Davis-Besse	Turbine			X			X	X	
ANO-1	Turbine	X		X			X		
	Motor	X		X			X		
TMI-1	Turbine	X		X					X
	Motor			X					X

been a history on these plants whereby the interaction of a faulted nonsafety-grade system such as the NNI or the ICS defeated the delivery of both main and auxiliary feedwater to the steam generators.

5.2.6.2 Operating History and Corrective Actions

A discussion of the feedwater-related incidents is given in NUREG-0560 (Ref. 3). Since publication of NUREG-0560, there have been recurring events in which delivery of feedwater to the steam generators has been defeated. The most recent event occurred at Crystal River 3 on February 26, 1980, where a combination of problems including failure of the NNI "X" bus, actuation of the steam generator rupture matrix, and failure of the auxiliary feedwater pumps to automatically start caused dryout of one steam generator and near dryout of the second.

As a result of the TMI-2 accident, the staff concluded in NUREG-0560 that a study should be made to see whether there are design deficiencies that may be corrected to reduce the frequency of feedwater transients. Further, it recommended that the reliability of the auxiliary feedwater systems should be improved and that action should be taken to upgrade the reliability of this system prior to startup of B&W operating plants.

The Commission Orders issued to each of the B&W operating plants in May 1979 (Ref. 2) required certain short-term actions of the B&W plants to upgrade the timeliness and reliability of the AFWS. These short-term actions included plant-specific hardware changes (e.g., automatic start features and ability to start pumps from a vital bus) and procedural changes (e.g., procedures to control AFW independent of the ICS). The plant-specific short-term actions are shown in Appendix A of the report. Long-term actions were also requi ed by the Orders. These long-term actions were also plant-specific and inc uded such items as (1) installation of two motor-driven AFW pumps per unit on Oconee Units 1, 2, and 3, and (2) installation of an AFW control system independent of the ICS and connecting the motor-driven AFW pump to a vital bus in ANO-1. The complete long-term actions are also listed in Appendix A of this report.

It should be noted that both the short-term and long-term actions required by the Orders did not result in an upgrade of the AFWS on the B&W plants that meets the standards specified in the staff's Branch Technical Position ASB 10-1 (Ref. 31). The staff recognized that further work w s needed to upgrade AFWS reliability. As such, the staff is conducting an ongoing study and has requested that all B&W operating plants consider additional means for upgrading the reliability of the AFWS. The objective of this study is to (1) identify necessary changes in AFWS design cr related procedures at these plants in order to assure their continued safe operation, and (2) to identify other system characteristics in the design of AFWS which on a long-term basis may require modification. In order to accomplish these objectives, the following work is being undertaken:

(1) An assessment of the relative reliability of each B&W operating plant's AFWS under various postulated potential failure conditions by determining the potential for AFWS failure due to common causes, single point vulnerabilities, and human error; and

(2) A review of plant-specific AFW designs in light of current regulatory requirements.

The Rancho Seco AFWS review has been completed by the staff, and staff positions for system modification have been identified (Ref. 32). The remaining B&W operating plants are currently under review.

5.2.6.3 Conclusions and Recommendations

(1) Conclusion:

Although the task force encourages efforts to strengthen the reliability of the main feedwater systems in order to reduce the challenges to the AFW systems, it still believes that because of the demonstrated sensitivity of B&W plants to a loss-of-feedwater transient, the AFWS design must be extremely reliable.

At the present time, an ongoing review of B&W operating plant auxiliary feedwater system reliability analyses is being accomplished. Part of this review will include a comparison of these systems with present staff guidance given in BTP ASB 10-1 (Ref. 31). This guidance provides for a diverse safety-grade system. The Task Force concludes that this review should be carried out in an expeditious manner in order to provide early identification of those aspects of the auxiliary feedwater system that do not meet present staff guidance.

The most effective method of assuring this high reliability is to provide a safety-grade auxiliary feedwater system which will be automatically initiated and controlled independent of the ICS, NNI and other non-safety systems. Thus, in staff judgment, a safety-grade AFWS gives added assurance of high reliability by providing the following:

(a) Electrical systems that meet IEEE 279-1971 Standards;

(b) Piping arrangements that take into account pipe failures, active component failures or control failures that could prevent system function;

(c) System design having suitable redundancy to offset the consequences of any single active component failure;

(d) System arrangements to assure the capability to supply necessary feedwater to the steam generators despite postulated rupture of any high energy section of the system, assuming a concurrent single active failure; and

(e) Equipment and systems that meet seismic Category I requirement (see proposed resolution of this question under the recommendation stated in this section).

Recommendation:

The Task Force strongly recommends that the AFWS on operating B&W plants be classified as an engineered safety feature system, and as such be upgraded, as necessary, to meet safety-grade requirements.

As an alternative, assuming comparable reliability, consideration would be given to the addition of a dedicated AFWS (i.e., separate train).

Note: With regard to the seismic requirements for safety-grade systems, time limitations did not permit us to resolve the matter regarding the imposition of this requirement to presently operating B&W plants. The Task Force believes that this question warrants further study and therefore recommends that the issue be expeditiously resolved by the Probabilistic Analysis Staff as a part of its ongoing risk assessment study of the AFWS.

(2) Conclusion:

The Task Force believes that appropriate protective features should be provided in the secondary systems that can quickly and reliably detect malfunctions and initiate action to prevent either undercooling or overcooling the RCS as well as overfilling of the steam generators. With regard to undercooling events caused by loss of main feedwater flow, the Task Force notes that NUREG-0578 (Ref. 13) recommends automatic initiation of the AFWS. We endorse this recommendation. However, the Task Force does not believe the actuation signals presently used to automatically initiate the AFWS are capable of detecting potential main feedwater delivery problems sufficiently to prevent steam generator dryout in all cases.

Recommendation:

The AFWS should be automatically initiated and controlled by engineered safety features (safety-grade) which are independent of the ICS, NNI, and other nonsafety systems.

The selection of signals used to initiate AFWS flow should be reevaluated to permit automatic initiation of AFWS in a more timely manner to preclude steam generator dryout (i.e., AFWS automatic start on anticipatory loss of feedwater). In addition, the level of secondary coolant in the steam generators should be automatically controlled by the AFWS in a manner to prevent overcooling of the RCS during recovery from feedwater transients and that an appropriate signal be provided to terminate feedwater flow to the steam generators before overfilling takes place.

(3) Conclusion:

Branch Technical Position ASB 10-1 (Ref. 31) states that the auxiliary feedwater system should consist of at least two full-capacity, independent systems that include diverse power sources. At the present time, the Davis-Besse 1 facility has two steam-driven AFW pumps, and therefore does not meet the diversity guidance provided in ASB 10-1. The staff has stated in its "Evaluation of Licensee's Compliance with the NRC Order Dated May 16, 1979" (Ref. 33) that it would require the licensee to provide the greater degree of diversity offered by a 100-percent-capacity motor-driven AFW pump or an acceptable alternative. The Task Force endorses the requirement for a 100-percent-capacity diverse-drive pump.

Recommendation:

The Task Force recommends that a diverse-drive auxiliary feedwater pump be expeditiously installed at the Davis-Besse 1 facility.

5.2.7 Steam Line Break Detection and Mitigation System

5.2.7.1 Design Concepts

A steam line break detection and mitigation system is provided on all operating B&W plants except the three Oconee Units. This system is known by a variety of names such as "Steam Line Break Instrument and Control" on ANO-1; "Steam Line Failure Logic" on Rancho Seco; "Steam Line Break Matrix" on Crystal River 3; and "Steam and Feedwater Line Rupture Control System" (SRFCS) on Davis-Besse 1. Table 5.3 provides a tabulation of the characteristics of the various steam line break detection and mitigation systems. Basically, these systems are designed to protect against the consequences of simultaneous blowdown of both steam generators because of a failure of the steam or feed piping. In general, upon detection of a steam line break, the system automatically initiates action to isolate each steam generator by closing the main steam valves and/or feedwater isolation valves. In the event of a steam line break, a reactor trip is followed by turbine stop valve closure. Low steam line pressure initiates feedwater isolation of the affected steam generator and allows this steam generator to blow dry. Steam pressurization in the unaffected steam generator causes the turbine bypass valve to open on increasing steam pressure. Continued reactor coolant system cooldown and decay heat removal is achieved by auxiliary feedwater flow to the unaffected steam generator with steam relief through the turbine bypass valves.

5.2.7.2 Operating History

During the recent Crystal River 3 event, the steam line rupture matrix isolated both the "A" and "B" steam generators at different times during the event. The "A" steam generator was isolated because of decreasing main steam pressure (due to dryout of the "A" generator). It should be noted, at that time, the "B" steam generator came close to being isolated because of near dryout. At a later time in the event, the "B" steam generator was isolated because of overcooling brought about by excessive injection of auxiliary feedwater. A similar actuation of the SFRCS occurred at Davis-Besse on September 14, 1977. In this case, a spurious SFRCS trip caused a reduction in heat removal from the primary system and a corresponding temperature/pressure rise in the RCS. The PORV lifted, rapidly oscillated open and closed, and finally stuck in the full-open position.

5.2.7.3 Conclusions and Recommendations

(1) Conclusion:

We conclude that the actuation of the steam line break and detection system during a normal loss of feedwater transient, or as the initiator of such a transient, can aggravate the primary system upset by removing the steam generator as a heat sink at a time when steam generator heat removal may be required.

Recommendation:

The steam line break detection and mitigation systems should be modified as necessary to eliminate adverse interactions between it and the auxiliary feedwater system. The Task Force also recommends that, in order to further assure heat sink availability, the steam line break detection and mitigation system should be reevaluated and modified so that it is capable of differentiating between an actual steam line or main feed line break and undercooling or overcooling events caused by feedwater transients.

TABLE 5.3 - OPERATIONAL CHARACTERISTICS OF STEAM LINE BREAK DETECTION & MITIGATION SYSTEMS

Plant	System	Characteristics
Oconee	None Provided.	
ANO-1	Steam Line Break Instrumentation & Control System (SLBIC)	Upon detection of a steam line break, SLBIC automatically initiates action to isolate each steam generator by closing the main steam block valve and the feedwater isolation valve on the affected generator. Actuation occurs when main steam pressure falls below 600 psig. Initiate aux feed to the unaffected steam generator
Rancho Seco	Steam Line Failure Logic	Low steam line pressure initiates automatic feedwater isolation by closure of the main feedwater control valves. Feedwater addition to the unaffected steam generator is through the auxiliary feedwater control valve. Steam isolation is by closure of the turbine stop valves.
Crystal River 3	Steam Line Break Matrix	Feedwater block valves close on affected S.G. upon low main steam line pressure (600 psi). Turbine stop valves close. Auxiliary feedwater system initiated if both main feed pumps are not in operation
Davis-Besse 1	Steam & Feedwater Line Rupture Control System (SFRCS)	SFRCS trips both main steam isolation valves when steam line pressure falls below 600 psig. All main feedwater control and stop valves close. Auxiliary Feedwater initiated and aligned to the unaffected steam generator
TMI-1	Steam Line Rupture Detection System	The steam line rupture detection system actuates when steamline pressure falls below 600 psig. As a result, main feed and auxiliary feed to the affected steam generator are isolated.
Midland* WPPSS*	Feed Only Good Generator (FOGG)	Auxiliary Feedwater is delivered to the intact steam generator following a main steam or main feed line break.

* - Not operating plants.

5.3 Instrumentation and Control Design and Operational Considerations

5.3.1 Integrated Control System Design

The integrated control system (ICS) has as its basic requirement the matching of generated megawatts with megawatt demand. The system philosophy is that control of the unit is achieved through feed-forward control from the unit load demand which in turn produces demands for parallel control of the turbine, reactor, and steam generators. By coordinating the flow of steam to the turbine and the rate of steam production, the ICS can match the generated megawatts to the megawatt demand.

The flow of steam to the turbine is controlled by the turbine throttle valves. The turbine header pressure is used as an index to determine whether the steam flow rate and the steam production are equal.

The rate of steam generation is controlled by varying the total amount of feedwater and reactor power and maintaining a proper ratio between the two so that the proper steam conditions exist. The feedwater flow is controlled by the feedwater valves and pumps, and the reactor power is controlled by moving the control rods in the reactor.

The ICS maintains constant average reactor coolant temperature between 15 and 100 percent rated power and constant steam pressure at all load conditions. Optimum unit performance is maintained by limiting steam pressure variations; by limiting the imbalance between the steam generator, turbine, and the reactor; and by limiting the total unit load demand on loss of capability of the steam generator feed system, the reactor, or the turbine generator. The control system provides limiting actions to ensure proper relationships between the generated power, turbine header pressure, feedwater flow, and reactor power.

The ICS was designed to be able to prevent a reactor trip for many anticipated plant upsets ranging from minor upsets, such as small load changes or small feedwater heating upsets, to major upsets, such as loss of one reactor coolant pump, loss of one main feedwater pump, or turbine trip from 100 percent power.

Following a reactor trip, the ICS controls steam generator level at a minimum level setpoint with the startup feedwater valves to provide decay heat removal. Upon loss of both main feedwater pumps, this minimum level control is accomplished with the auxiliary feedwater valves. Should loss of all four reactor coolant pumps occur, the level is controlled at a higher level in the steam generator (i.e., 50 percent on the operating range indication) to help promote natural circulation. Following a reactor trip, the ICS also provides control of the steam pressure with the turbine bypass valves or the atmosphere dump valves (depending on the availability of the condenser and circulating water).

5.3.2 Non-Nuclear Instrumentation Design

B&W plants utilize a set of instrumentation classified as the non-nuclear instrumentation (NNI) to provide a significant amount of the input information to various plant control systems, including the ICS. Additional input information for plant control is obtained from the reactor protection system (RPS) and nuclear instrumentation (NI) system. In addition to providing information to the plant control systems, the NNI supplies control room information for the main control board, plant computer, alarm annunicator, and display information from the RPS and the engineered safety features (ESF) systems. Therefore, the NNI is the major source of information from which the operator can determine conditions in the primary and secondary systems.

The NNI also plays a large part in reactor coolant system pressure control by providing control signals for the pressurizer spray valve, heaters, and PORV. Pressurizer level is controlled by the makeup and purification system. NNI provides the pressurizer level signal for automatic control of the makeup and purification system, low pressurizer level heater cutoff, and control room indication.

Typical B&W NNI systems (and integrated control systems) consist of two subsystems, "X" and "Y." This would imply a certain degree of channelization and/or redundancy. More channelization occurs in some plants than others. For certain parameters such as steam generator level, pressurizer level and steam pressure, redundant sensing instrumentation exist with switches on the main control board that are used to select one of the redundant sensors for both control and indication. However, the full advantage of redundancy is often compromised by the use of one indicator, and, furthermore, operating experience indicates that the sensors and signal conditioning are not channelized in a balanced manner. For example, it appears that one subsystem ("X" or "Y") contains a proportionately larger amount of equipment. In addition, one subsystem may contain signal conditioning (e.g., temperature compensation networks) for sensors in both subsystems. Therefore, there is no true redundancy for the NNI.

5.3.3 ICS and NNI Power Supply Design

The power for the NNI/ICS systems has not been required to be on Class IE (safety-grade) power sources because the systems were not designated as part of the plant protection systems (RPS or ESF). However, in order to obtain the most reliable source of power for these systems (to prevent plant unavailability) most of the plants provide the source of power from the Class IE vital buses. In some plants, the vital buses utilized are not classified as the Class IE buses; however, they do have similar reliability because they can receive power from a battery source in case of loss of offsite ac power Regardless of the "quality or reliability" of sources, power supplies do fail. Most notably these have been caused by inverter failures or transfer switch failures. Therefore, events such as reactor trip can be initiated by a single channel power supply failure.

During such events, the operator may also lose a significant amount of control room information <u>on which he would normally rely</u>. This is clearly an unacceptable situation. The above emphasis is added to point out that arguments can be made that the operator does have sufficient information to safely shut down the plant; however, this may require special procedures and unfamiliar operating modes. In addition, the operator must be able to recognize the need to revert to these procedures, which can be complicated if the operator does not readily recognize the failure mode (e.g., because of "mid-scale" failures) and, therefore, does not utilize the proper procedures immediately.

5.3.4 Reactor Protection System and Engineered Safety Features Instrumentation Design

The principal function of reactor protection system (RPS) and engineered safety features (ESF) instrumentation is covered concisely in IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations" (Ref. 30). The requirements of IEEE Standard 279-1971 are included in the Code of Federal Regulations 10 CFR 50, Paragraph 50.55(h), "Codes and Standards - Protection Systems," and are therefore to be enforced by the NRC as are other standards such as the ASME Pressure Vessel Code.

The purpose of this discussion is to present some insight into how nuclear steam system suppliers (NSSS), architect-engineers, utilities, and the Nuclear Regulatory Commission have been interpreting and implementing safety system requirements and to identify the areas where most emphasis was placed by these organizations. In addition, an attempt will be made to explain the evolution and relationship of control room instrumentation including control systems, and the nonsafety instrumentation such as the annunciator panels, the computer, and the sequence-of-events recorder.

Important control systems such as the B&W integrated control system are not required to be designed to satisfy the single-failure criterion of IEEE Standard 279. Likewise, controls and process system indicators and alarms that monitor normal plant operating and transient parameter are not required to be redundant. IEEE Standard 279 has not required that instrumentation be provided to allow for comfortable operation by the control room operator. Instead, instrumentation meeting IEEE Standard 279 (henceforth called "safety system instrumentation") was provided to perform explicit important safety functions. These functions include immediate automatic shutdown of the reactor upon occurrence of a transient regardless of the cause (e.g., equipment malfunction, failure of a control system, or operator error). Each nuclear steam system supplier has determined that certain plant parameters (e.g., pressure, flux) should be monitored and a limiting safety system setting provided to mitigate the consequences of the anticipated transients. This aspect of the analysis and design of safety systems has also been dealt with thoroughly. Similarly, the industry and the NRC have considered the design basis accident scenarios [i.e., loss-of-coolant accident (LOCA)]. For the LOCA and the steam line/feedwater line break accident, as with the reactor transient, sets of parameters were selected to insure that safety systems such as emergency core cooling and containment cooling and isolation would be initiated as required to insure that fuel design limits would not be exceeded and that other barriers such as the containment building would be automatically isolated and cooled to the extent required and described in the accident analysis. This area of safety system design has also been treated adequately for the major LOCA event (i.e., large pipe breaks). In general, we believe that safety system instrumentation for mitigation of reactor transient and reactor accidents events is adequate.

Safety system instrumentation normally includes four independent measurements of each parameter that is required to trip the reactor and/or to initiate emergency core cooling systems and other engineered safety features. These channels, including indicators, are located in RPS cabinets and ESF cabinets. In most plants, the information provided in the RPS and ESF cabinets is also made available to the control room operator. The displays are buffered signals taken directly from protection system instrumentation. In a few plants, this information from the RPS and ESF is not available in the control room.

The amount of information available to the control room operator is very much plant dependent. The NSSS vendors recommend utilization of safety system instrumentation to interface with control systems and with the operators in a variety of ways. It is important to understand that in the past the only regulatory restriction regarding the use of the safety system instrumentation, as an input to control systems, was that the safety function not be compromised by using the protection signal for these other purposes. Therefore, the extent to which the operator has safety system instrumentation available at the station where routine operation takes place varies widely. The quantity of safety-grade and nonsafety-grade instrumentation displayed on operating panels, available in the computer, in the annunciator system, and in the sequence-of-events recorder is different for each operating plant. In recognition of this situation, the NRC has accelerated the review and planned implementation of Regulatory Guide 1.97, Revision 2, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant and Environs Conditions During and Following an Accident" (Ref. 34). This regulatory guide addresses the need to provide a minimum amount of safety-related displays for the operator not only to mitigate the consequences of transients and accidents but also to monitor the performance of safety systems.

In response to an event which occurred at the Oconee Nuclear Station, Unit 3, on November 10, 1979, NRC issued IE Bulletin 79-27 (Ref. 8). This bulletin requested licensees to review buses supplying power to safety and nonsafety-related instrumentation and control systems that could affect the ability to achieve a cold shutdown condition. The bulletin also requested that emergency procedures be revised or prepared to include identification of alternate indication and control circuits that would be available in the event of loss of each power bus.

Other events before and after this Oconee event (i.e., Rancho Seco and Crystal River 3) indicate that response to the IE Bulletin alone may not adequately resolve the problem of inadequate operator information. The Task Force is proposing a requirement for a set of safety-related indicators to be located in the control room to provide the operator with adequate information to assess plant conditions during and following anticipated operational transients.

5.3.5 Instrumentation and Control Operational Considerations

Prior to the TMI-2 accident, B&W plants totaled about 35 years of combined operating data. In that time period, as reported in the B&W "Integrated Control System Reliability Analysis" (BAW-1564, Ref. 35), there were 310 reactor trips, about one-third of which are attributed to the ICS or ICS-related equipment. Comparative information submitted by B&W, in a letter from J. H. Taylor (B&W) to D.F. Ross (NRC) dated October 18, 1979 (Ref. 36), would indicate that B&W plants have a lower trip rate than other PWR vendors.

Prior to TMI-2, there had been at least one very significant event at a B&W plant involving almost all of the control sytstem aspects discussed above. This was the March 20, 1978 Rancho Seco "light bulb incident." The event was initiated by a dropped light bulb that caused a short, which in turn caused loss of power to the NNI-Y subsystem. This caused the control systems to receive erroneous information and caused a large amount of the faulty information to be presented on the main control board. The event resulted in a reactor trip, high-pressure injection actuation, and an excessive cooldown rate for the plant. It could be conjectured that the event may have been significantly different were it not for the fact that the PORV block valve was manually closed and that secondary cooling (to at least one OTSG) was available at all times.

The TMI-2 accident did not involve the failure of any of the previously discussed control systems. It did, however, involve a loss of main feedwater which could have been initiated by an ICS failure. It involved the temporary loss of auxiliary feedwater, which could also have been caused by the ICS. It also involved problems related to instrumentation, although not attributable to any instrument failure but rather to the lack of direct information, misinterpretation of information, and too much emphasis on one plant parameter (i.e., pressurizer level). It also involved a stuck-open PORV that could have been caused by its control system failure.

After the TMI-2 accident, a series of further events occurred at B&W plants. Most notable were the increase in reactor trips (as reported in an August 23, 1979 meeting between the staff and the B&W licensees, Ref. 37) and the Oconee 3 and Crystal River 3 loss of NNI/ICS power supply events of November 10, 1979, and February 26, 1980, respectively.

Subsequent to the TMI-2 accident, the NRC and the licensees took a number of actions on the B&W plants in the area of control systems. In an attempt to avoid challenges to the PORV, the staff issued IE Bulletin 79-05B requiring the licensees to raise the setpoint of control system actuation of the PORV from 2255 psig to 2450 psig. The reactor protection system trip point for high pressure was simultaneously lowered from 2355 to 2300 psig. In conjunction with these changes, it was also proposed by the licensees (and confirmed by Commission Order) that hard-wired, anticipatory reactor trips would be installed for turbine trip and loss of feedwater (as sensed by the loss of both pumps). These trips were implemented in the short term as part of the plant control system (hence, the term control-grade) and are to be implemented in the long term as part of the reactor protection system (i.e., safety-grade).

The TMI-2-related concerns with the ICS were addressed in a number of ways. The short-term portion of the Orders required that the plants have procedures to initiate and control the auxiliary feedwater flow independent of the ICS. To determine the (potential) contributions of the ICS in plant upsets, B&W proposed to perform a reliability analysis including a failure mode and effects analysis (FMEA) of the ICS. This was confirmed by the long-term portion of the Commission Orders.

Simultaneously, the Lessons Learned Task Force made related recommendations in the area of auxiliary feedwater automatic initiation. Requirement 2.1.7.a of NUREG-0578 (Ref. 13) required control-grade automatic initiation and in the long term required that this actuation system be upgraded to a safety system. The implementation of this requirement on B&W plants would effectively remove this function from the ICS, since the ICS is not a safety system.

The staff's review of the ICS Reliability Analysis (BAW-1564) (Ref. 35) was initiated with the aid of consultants from Oak Ridge National Laboratory. The

B&W report makes recommendations to be evaluated on a plant-specific basis. The recommendations highlight areas in which B&W believes improvements could potentially contribute to significant improved overall operation of the B&W facilities. The majority of the recommendations involved areas outside the ICS itself and were general in nature because of design differences in these areas at the various plants. The recommendations did involve two directly related areas, specifically the NNI/ICS power supplies and the ICS input signals. By letter from R.W. Reid to all B&W operating plants (Ref. 38), the staff requested the licensees to address what actions they were taking to implement these recommendations.

It should be noted that the B&W analysis (BAW-1564) concentrated on the control system (ICS in particular) and recommended improvements intended to reduce reactor trips and/or ECCS actuation. It should also be noted that it did not address the very significant control board information problem encountered at Oconee 3 and Crystal River 3.

As reviewed in Section 5.3.4, on November 10, 1979, Oconee Unit 3 experienced a loss of power to a non-Class IE 120-Volt ac power panel that supplied power to the ICS and the NNI. This loss of power resulted in control system malfunctions and significant loss of information to the control room operator. The loss was corrected in less than 3 minutes. Subsequent to this event, IE Bulletin 79-27, "Loss of Non-Class IE Instrumentation and Control Power System Bus During Operation" (Ref. 8), was issued.

The bulletin was intended to (1) obtain information on what the condition of each plant would be in the event of this type occurrence, (2) have licensees prepare procedures for such type occurrences, (3) have licensees review other inverter failures, and (4) have licensees propose any necessary modifications. The bulletin was issued November 30, 1979, with a requirement for response within 90 days.

The Crystal River 3 event occurred February 26, 1980, and involved the loss of a power bus; control system response leading to adverse actions which, in turn, led to reactor trip and high-pressure injection; and loss of a significant amount of control board information.

Because of the implications of the Crystal River 3 event and potential adverse effects on the public health and safety that could result from future events of this type, a letter to all operating B&W reactor licensees was issued on March 6, 1980 (Ref. 10). This letter requested information in six areas: (1) summarize power event upsets on NNI/ICS that have occurred; (2) address the susceptibility of each B&W plant to a Crystal River 3 event; (3) address information available to the operator following various NNI/ICS power upset events; (4) address the feasibility of performing a test to verify reliable information that remains following various NNI/ICS power upsets; (5) address the Crystal River 3 proposed corrective actions; and (6) expand the review of IE Bulletin 79-27 to include the implications of the Crystal River 3 event. Responses to this request are presently undergoing staff review.

As previously discussed, there are a number of ongoing activities related to solving the problems of assuring that adequate instrumentation is available to the operator. The most significant include the implementation of Regulatory Guide 1.97, Revision 2 (Ref. 34), and the Long-Term Lessons Learned Task Force

Recommendation 7.2 (Ref. 14) regarding a safety parameter display console. In addition, the Interim Reliability Evaluation Program (discussed in Section 6 of this report) may give further insight into necessary modifications to the ICS and/or other instrumentation and control systems.

5.3.5.1 Conclusions and Recommendations

(1) Conclusion:

Based on B&W operating plant transient information, we conclude that it is highly desirable to reduce dependence on high-pressure injection (HPI) actuation and safety valve relief for the mitigation of anticipated operational occurrences. The steam generator is the preferred route (heat sink) for heat removal during these types of events. To accomplish this objective, consideration must be given to several areas:

First, the auxiliary feedwater system (AFWS) must be highly reliable and independent of the normal control system. Second, the control of the steam flow (heat removal) must also be highly reliable. These two concerns are addressed in Sections 5.2.6 and 5.2.7 of this report. Third, the normal control system should be improved to reduce the number of challenges to the safety systems.

Recommendation:

Improve the reliability of the plant control system, particularly with regard to undesirable failure modes of power source, signal source, and the integrated control system itself. Specific recommendations for improvement in the plant control system include the following:

(a) The power buses and signal paths for non-nuclear instrumentation and associated control systems should be separated and channelized to reduce the impact of failure of one bus.

(b) The power supply (including protective circuitry) logic arrangement should be reconsidered to eliminate "mid-scale" failures as a preferred failure mode for instrumentation. "Full-scale" or "down-scale" failures may be preferred in that they give the operator more positive indication of instrumentation malfunction.

(c) Multiple instrument failures, typically caused by power loss, should be unambiguously indicated to guide the operator to select alternate instrumentation that is unaffected by the failure.

(d) If control system failures or response to failed input signals can cause substantial plant upsets (e.g., requiring action by engineered safety features or safety valves in addition to reactor trip), the control system should have provisions for detecting gross failures and taking appropriate defensive action automatically such as reverting to manual control or some safe state.

(e) The NNI power buses should be reviewed and rearranged as necessary to provide some redundancy of indication on each reactor coolant and secondary system loop. That is, where redundant indicators for one loop are provided, one channel should be powered from NNI "X" and the other from NNI "Y," instead of loop "A" being powered from NNI "X" and loop "B" from NNI "Y."

(f) Prompt followup actions should be taken on the recommendations contained in BAW-1564 (Integrated Control System Reliability Analysis).

(g) Prompt followup actions should be taken on the recommendations contained in NSAC-3/INPO-1 (Analysis and Evaluation of Crystal River-Unit 3 Incident).

(h) Prompt followup actions should be taken on IE Bulletin 79-27.

(2) Conclusion:

In order to provide the operator with reliable information to accurately assess plant conditions, it is necessary to provide certain selected parameters for display in the control room. These displays must be highly reliable.

The Task Force recognizes that other ongoing work addresses itself to similar types of requirements; namely, implementation of Regulatory Guide 1.97, Revision 2 (Task Action Plan II.F.3), and implementation of the plant safety parameter display console or safety state vector (Task Action Plan I.D.2). However, the Task Force believes that there is a more urgent need for a minimum display of critical plant parameters for B&W plants.

Recommendation:

Establish a minimum set of parameters that will enable the operator to assess plant status. This Task Force recommends a minimum set that includes:

(a) Wide-range reactor coolant system pressure,

(b) Wide-range pressurizer level,

(c) Wide-range reactor coolant system temperatures: hot leg (each loop), cold leg (each loop), and core outlet (two or selectable),

(d) Makeup tank level,

(e) Reactor building pressure,

(f) Wide-range steam generator level (each steam generator),

(g) Wide-range steam generator pressure (each steam generator),

(h) Source-range nuclear instrumentation, and

(i) Intermediate range nuclear instrumentation.

(j) Borated water storage tank (BWST) level.

The instrumentation for the selected parameters must meet the following requirements:

(a) The instrumentation must be reliable and redundant and should meet all applicable codes and standards for protection system instrumentation; and

(b) In accordance with safety standards, this requires a minimum of two redundant channels of all designated information, at least one channel of which shall be recorded automatically on a timely basis for use in trending, instant recall, and post-event evaluation;

Note: Display and recording requirements are flexible in accordance with human factors considerations and may be accomplished with modern technology (e.g., computers and video displays), provided that these devices also meet applicable standards for protective-grade equipment. Instrumentation, power, display, and recording equipment must be independent of the plant control system and plant computer to the extent specified in paragraph 4.7 of IEEE Standard 279-1971.

(3) Conclusion:

The Task Force believes that it is important for the operator to have an accurate and meaningful indication of subcooling margin including periods where forced circulation is not provided. In addition, the value of having a continuous or trending display of incore thermocouples has been amply demonstrated at TMI-2, Crystal River 3, and during operator requalification training at the B&W simulator.

Recommendation:

Provide the flexibility to substitute appropriate combinations of core thermocouples for the loop resistance temperature detectors (RTDs) presently used for primary temperature input to the subcooling meters. In addition, consideration should be given to providing a continuous or trending display capability for the incore thermocouples. This display need not be indicated in the control room at all times but could be called up on demand from the computer.

(4) Conclusion:

The operating history of B&W plants has shown that as a result of operational transients, such as loss of feedwater events or overcooling transients that lead to high-pressure injection actuation, a high probability exists that reactor coolant inventory may be released from the system through the pressurizer PORV or safety valves. During the course of this event, radioactive gases, which collect in the top of the pressurizer, would be expelled. If during this transient a containment vent and purge operation were in progress, it would be important to isolate containment as quickly as possible to minimize the release of radioactivity to the environment. Although all plants provide a safety-grade high containment pressure isolation signal to perform this function, this signal may prove inadequate during small or intermediate releases of coolant to the containment building, because a buildup of containment pressure to the actuation setpoint may not occur with the valves open. It appears that the most effective way of providing containment atmospheric isolation, as required, would be through the use of containment high radiation signals.

Recommendation:

Provide safety-grade containment high radiation signals to initiate containment vent and purge isolation in addition to the presently required signals (i.e., containment high-pressure and low-pressure ESFAS actuation).

5.4 Operator Training and Qualifications

5.4.1 Minimum Qualifications and Training Including Simulator Training

Operators and senior operators at B&W facilities typically receive similar quantitative training as do licensed personnel at all other vendor-designed power plants. Their training can be divided into three types: cold, hot, and requalification. Cold license training is given to the initial plant staff prior to fuel loading. The training program is submitted in Section 13.2 of PSARs and FSARs (Preliminary and Final Safety Analysis Reports). The programs generally consist of four phases: fundamentals, observation, design lecture series, and onsite training. The observation phase includes a simulator course that is typically 4 to 6 weeks in duration. The design lecture series is conducted by the NSSS vendor.

The hot license training programs are similar to the cold license programs but shorter in duration, since the majority of it can be conducted at the operating plant. Successful participation in a requalification program is required for an individual to maintain a license. It consists of facility administered lectures, on-the-job-training and tests that are audited by the NRC.

The duration of the training program is generally as follows: for cold training, 2 to 4 years; for hot training, 1 to 2 years on site as auxiliary operator plus one year in training programs; for requalification, continuous over duration of license period with annual written examination.

The education and experience requirements of licensed personnel at the time of both the TMI-2 accident and Crystal River 3 incident were set forth in ANSI N18.1-1971 ("Selection and Training of Nuclear Power Plant Personnel," Ref. 39). Operators were required to have a high school diploma or equivalent and two years of power plant experience of which one year was nuclear. Senior operators were required to have four years of responsible power plant experience and the same education requirements as an operator.

As indicated in the TMI-2 Action Plan (Ref. 7), numerous changes will be made in the training and qualification of licensed personnel. Among the more notable are increased experience levels, expanded and mandatory use of improved simulators, and NRC-administered requalification exams.

Simulator training is an integral part of all of the training programs. Personnel with no previous nuclear experience must participate in an NRC-approved simulator training program to meet cold eligibility requirements. This is not a requirement for hot license applicants since they may obtain the experience while working at their plant. The requalification program requires 10 significant reactivity manipulations during a two-year period which may be on the plant or a simulator. In practice, most of the required operating experience for hot training and requalification is met in approved simulator programs.

The simulator training of operators at the four different vendor type plants varies considerably for the following reasons: (1) number of vendor type simulators; (2) variety of plant design (e.g., two loop, three loop), and (3) NRC-approved programs.

At the present time there is only one operational simulator for each of the B&W and CE vendor designs. The B&W simulator is located in Lynchburg, Virginia, and is representative of the Rancho Seco control room. The CE simulator is located at Windsor Locks, Connecticut, and is representative of the Calvert Cliffs control room. There are two boiling water reactor (BWR) simulators (Dresden 2 and Brown's Ferry) and five Westinghouse simulators (Zion, Indian Point, Surry, McGuire, and Sequoyah).

The simulator training that B&W operators receive is unique in that there is much more uniformity in plant systems, instrumentation, and operation than any other vendor design. The single exception is Davis-Besse 1 which has the raised-loop design and low-head high-pressure injection pumps. Otherwise, the primary reactor coolant system (RCS), once-through steam generator (OTSG), ICS, emergency core cooling systems (ECCS), and plant systems including the makeup and purification system are very much the same from plant to plant. The trainees therefore do not encounter the degree of unfamiliarity that sometimes occurs in training conducted on other vendor-type simulators. The staff believes that this is a distinct advantage in the training of B&W operators. Also, the startup certifications of operators at B&W plants must be conducted at the B&W simulator (requirements for GE are similar). However, allowance has been made in the past for some Westinghouse facility operators to obtain startup certifications on the CE simulator and vice versa.

The disadvantages of the B&W simulator training are (1) age and fidelity of the simulator, and (2) counter-productive simulation for Davis-Besse operators. Because the B&W simulator was one of the first of its kind, there is a distinct lack of fidelity in some areas. For example, modifications had to be made to simulate both the TMI-2 and Crystal River 3 events. Two-phase conditions in areas of the RCS other than the pressurizer and multiple failures of instrumentation were not part of the computational model.

The "feed and bleed" method of core cooling for some small-break LOCAs and loss of feedwater events is not applicable for the Davis-Besse facility. Also, the auxiliary feedwater and steam generator level control system are distinctly different from the simulator. These differences are mentioned because the Crystal River 3 operators credit the simulator training on solid system operation and natural circulation as being very beneficial in their response to the loss of NNI event on February 26, 1980.

5.4.2 Assessment of Licensed Operators at B&W Facilities

5.4.2.1 Discussion

The logical method of assessing the role of B&W operators with regard to their ability to cope with transient events, particularly secondary side upsets, is to make a comparison with operators of other vendor-type reactors. Unfortunately, no solid data base exists for such a comparison. The only applicable data that exist for comparison are the Licensee Event Reports (LERs) attributed to licensed personnel error sorted by vendor types. One must be careful in attaching a great deal of significance to these data. These LERs have only been categorized by licensed personnel error since January 1978 and different criteria have been applied in differentiating such events. For example, a Westinghouse four-loop plant has reported the most (24) licensed personnel errors from 1978 to the present with another Westinghouse plant and a BWR close behind (21 each).

However, three other Westinghouse plants and a BWR were the only ones reporting no LERs attributable to the same cause. Also, the reported LERs decrease significantly with the age of the plant, those having already undergone the first several years break-in period generally submitting the fewest LERs.

The results of a computer search of the NRC's LER File are shown on Table 5.4. It is apparent that the personnel errors committed at B&W-designed plants are above the norm; higher than other vendor-designed facilities, however, it is unknown if the statistical variation is significant. A comparison was made of total LERs submitted by PWRs in 1978 to determine if any vendor-type was out of proportion. The results are shown in Table 5.5. A comparison of the two tables shows B&W to have a slightly higher proportion of licensed personnel error LERs to total LERs reported.

TABLE 5.4

LER OUTPUT ON LICENSED OPERATOR EVENTS
FROM 1978 TO THE PRESENT

Vendor	Total Plants	LERs	Average/Plant
B&W	9	58	6.44
GE	25	142	5.68
CE	8	45	5.63
Westinghouse	25	131	5.24

TABLE 5.5

TOTAL LERS PER PWR FACILITY IN 1978

PWR NSSS	Total LERs
B&W	40.5
Westinghouse	41.0
CE	41.4

The results of this basic analysis are inconclusive but the data reveals a trend that should be explored further. It must also be remembered that personnel error-related LERs are not all operational events. In fact, the data from 1978 show that 46.5 percent of the LERs reported from PWRs were operational events and 41.4 percent were from routine tests or inspections. A vendor comparison of this type of data should also be further investigated.

Another basis for comparison is the required immediate actions that an operator must take while carrying out an emergency procedure. Again, before making a comparison one must carefully examine the background information. The number of emergency procedures generally follows the recommendations of Regulatory Guide 1.33 ("Quality Assurance Program Requirements (Operation)," Ref. 40). There are 26 suggested "Procedures for Combating Emergencies and Other Signficant Events," some of which are only applicable to BWRs and others only to PWRs. Nonetheless, of two B&W plants chosen at random, one had 17 written and approved emergency procedures and the other had 35. A Westinghouse plant also randomly chosen had 45 emergency procedures.

In addition, the content of emergency procedures varies considerably. One loss-of-coolant accident (LOCA) procedure for small leaks at a B&W plant has only three required immediate actions, whereas a BWR has a procedure that requires 19 immediate actions for loss of compressed air supply to the main generator output breakers.

With this in mind, a comparison of emergency procedures generally shows B&W plants require more manual immediate actions on the part of the operator than other vendor types. Required immediate actions should be limited to verifications in so far as practical. Verification, in this context, means to ensure an automatic action has occurred or manually perform it if necessary. The most common transient condition that nuclear power plant operators are faced with is a reactor trip. Often this is preceded by alarms or indications that may or may not permit the operator to diagnose the cause. He is usually faced with performing the immediate operator actions of his reactor trip procedure without knowing the exact cause of the trip. For a B&W plant, two of the first immediate actions are to isolate letdown and start another makeup (HPI) pump, if required. These actions are necessary to reduce the magnitude of the primary system volume shrink due to the temperature decrease. Shutting a valve and starting a makeup pump are not unreasonable demands to make of operators following a trip.

However, as a result of the small-break loss-of-coolant analysis, it was determined that for a certain size break spectrum the stopping/loss of coolant pumps at certain time intervals into the transient could result in the ECCS not providing sufficient cooling to meet acceptable criteria. Thus, new demands were placed on the B&W operators. Following a reactor trip and ESFAS actuation on low pressure, the reactor coolant pumps must be tripped immediately (similar requirement for all other PWRs) and an evaluation made as to the adequacy of HPI flow. If failure of one train is experienced, the operators have to direct plant operators to open several sets of valves in the auxiliary building. Clearly, the demands on the operators are becoming excessive. The staff believes that to the extent feasible, few if any manual actions should be required by operators during an emergency. Accordingly, modifications should be made to the plant to lessen this burden on the operator.

The B&W licensees have undertaken, through their Owners' Group, the development of abnormal transient operating guidelines (ATOG). The objective of the program is to simplify the operator's problem of identifying and treating abnormal transients. The guidelines will enable each B&W facility to develop a standardized set of abnormal and emergency procedures and will be based on input data from each of the facilities and transient analyses performed by B&W. This Task Force endorses this program and believes full utility support should be given to its expeditious completion.

5.4.2.2 Conclusions and Recommendations

(1) <u>Conclusion</u>:

The manual actions required of an operator as part of his immediate actions for an emergency procedure should be minimized. Immediate actions should be limited to verifications insofar as practical.

Recommendation:

Modifications should be made to the plant, to the extent feasible, to reduce or eliminate manual immediate actions for emergency procedures.

(2) Conclusion:

The number of LERs attributed to licensed personnel error is slightly higher at B&W facilities than other nuclear power plants.

Recommendation:

Based on the higher number of LERs attributed to licensed personnel error at B&W facilities as compared to other vendor-designed plants, the NRC should conduct further analysis and investigation to determine the significance and cause of this apparent finding.

5.4.3 Crystal River Incident of February 26, 1980

5.4.3.1 Response of Crystal River 3 Operators

The comments in this section are based on the following information; the meeting of March 4, 1980 in Bethesda, Maryland (Ref. 41), the Florida Power Corporation (FPC) submittal of March 12, 1980 (Ref. 11), the "Analysis and Evaluation of Crystal River Unit 3 Incident" (NSAC-3/INPO-1, Ref. 9), and the observations of an NRC Operator Licensing Branch Examiner who was conducting an oral examination in the control room at the time of the incident. The overall conclusion of the Task Force is that the operating crew took prudent actions and responded well to the incident in spite of the lack of instrumentation and applicable procedures. It should also be noted that apparently little attention was given to Recommendation 2.2.2a of NUREG-0578, "Control Room Access" (Ref. 13); at one point during the event, 17 people were observed within the restricted access area of the control room.

The sequence of events is sufficiently documented in the report of the licensee (Ref. 11) and therefore will not be repeated here. An assessment of the response of the operators based on the above information will be made. The adequacy of the training and procedures will also be addressed.

Within 14 seconds of the loss of NNI "X" power supply (NSAC-3/INPO-1 states 24 seconds), the reactor tripped and the operators began performing the reactor trip procedure. Prior to the trip, the operators recognized that some failure had affected the NNI and ICS. Shortly after the trip, at least four other station members arrived in the control room to provide assistance.

Less than half a minute after the trip, a computer alarm indicated less than the required (50°F) subcooling. Most of the B&W plants require the manual initiation of high-pressure injection if the subcooling margin is lost. This was not a procedural requirement at Crystal River.

In a little over three minutes, HPI initiated due to low reactor coolant system pressure. The operators began performing the actions required by emergency procedure EP-106, "Loss of Reactor Coolant or Reactor Coolant Pressure," in addition to Short-Term Instruction 80-17. The latter instruction required

10 containment isolation valves to be closed to comply with Recommendation 2.1.4 in NUREG-0578. Within 20 seconds after the initiation of HPI, the reactor coolant pumps were tripped. The B&W small-break LOCA analysis shows that for the most limiting condition the pumps must be tripped within two minutes of HPI initiation.

There is a discrepancy between the NSAC/INPO report and the FPC March 12, 1980 submittal as to when the PORV was isolated. The latter states the block valve was closed 2½ minutes after the event initiated and about 50 seconds prior to HPI automatic initiation. It is difficult to see why pressure would continue to decrease to the HPI setpoint with the steam generators drying out and the PORV isolated. The NSAC/INPO report shows the block valve to be closed 5 to 7 minutes after the beginning of the event. The time to isolate the PORV compares favorably with the 20 minutes it took at Davis-Besse in 1977 and more than 2 hours at TMI-2.

The first subsequent action in the emergency procedures is to check alternate instrument channels to confirm key parameters. This was inserted as a requirement from the Bulletins & Orders Task Force. When instructed to do this by the assistant shift supervisor, who was reading the procedures, the reply was that they did not know what to believe. There was no procedural guidance concerning instrumentation that would be reliable.

The operating crew at this point was using applicable instructions from a number of abnormal and emergency procedures. Their training directed them to follow the most conservative approach, which was the loss-of-coolant procedure during the initial phases of the event. The NSAC/INPO report identified 13 different procedures and instructions that were or should have been followed. None of these procedures addressed the loss of NNI/ICS power. At the time of the Crystal River event, only Rancho Seco had procedures that addressed NNI/ICS power failures and the resulting impact on the plant. The Task Force believes that each B&W facility should implement procedures concerning the partial or total loss of NNI/ICS power.

Twenty-one minutes after the event began, power was restored to the NNI "X" bus and plant recovery began in accordance with the applicable procedures. The delay in restoring power was attributed to the I&C technician first being directed to the wrong power supply due to an ambiguous annunciator. It was not until power was restored that the operators were able to throttle HPI and stop the coolant release from the pressurizer safety valve. Had this event occurred on a back shift without I&C coverage, it is unknown how long the event would have continued before power to the lost instrumentation would have been regained and the event terminated. Since B&W facilities are uniquely susceptible to this type of failure, the Task Force is recommending that round-the-clock coverage by qualified I&C technicians be provided at all operating B&W reactors.

Throughout the first half-hour of the event, the operating crew was very aware of subcooling and the importance of maintaining an adequate margin to saturated conditions. They followed their training and procedural guidance for HPI termination criteria and RCP trip requirements. When NNI power was restored, they were operating under solid system pressure control. In this condition, the RCS is completely full of water with no steam bubble in the pressurizer and pressure is controlled with makeup and letdown. The Crystal River operators had received training in this mode of operation at the B&W simulator only subsequent to the TMI-2 accident. The Crystal River training staff confirmed that the simulator training the operating crew received was very beneficial in helping them to respond to the event as well as they did. The Task Force is therefore recommending that all operators licensed at B&W facilities receive additional simulator training beyond what was required by the May 1979 Commission Orders (Ref. 2).

Incore thermocouple temperatures were a parameter that was available and heavily relied upon by the Crystal River operators. The value of the information provided from this source was amply demonstrated at TMI-2. Also, a member of the NRC staff observed requalification training at the B&W simulator in which natural circulation cooling was practiced. The digital display of the incore thermocouples was of great benefit to the operators. For these reasons, the Task Force is recommending that all B&W facilities have the capability to continuously display and/or trend incore thermocouple readouts. Recommendations in this area are contained in Section 5.3.5.1 of this report.

5.4.3.2 Comments on NSAC and INPO Report on Crystal River Incident

The comments in this section will be limited to the findings, conclusions, and recommendations of the report as they relate to training and procedures. This Task Force endorses the nine recommendations made in the NSAC/INPO report (Ref. 9). Additional Task Force recommendations concerning training and procedures are contained in Sections 5.4.2.2 and 5.4.3.3 of this report.

The NSAC/INPO report recommends that the industry should further analyze and resolve with the NRC the reactor coolant pump trip requirements. This Task Force emphatically agrees with this. The pump trip requirement is explained in Section 5.2.4 of this report. This requirment was identified by the PWR vendor analyses and is necessary to meet licensing criteria for a limited spectrum of small break loss of coolant accidents. The NRC staff has recognized that tripping the reactor coolant pumps for non-LOCA events may lead to further aggravation of the incident. At the present time, the most appropriate means of resolving this dilemma is with the automatic RCP trip circuitry.

The report concluded that the post-TMI-2 generic small-break LOCA emergency procedure was not conservative in that it challenged the pressurizer safety valves and the containment barrier. The termination criterion as specified in the B&W Guidelines and Crystal River procedure EP-106, is that the HPI pumps cannot be turned off unless all hot and cold leg temperatures are at least 50°F subcooled and the action is necessary to prevent the indicated pressurizer level from going off-scale high. Until the NNI "X" power was restored, there was no indication of pressurizer level. Had the level indication not been lost, throttling of HPI could have been performed much sooner without challenging the pressurizer safety valves.

5.4.3.3 Conclusions and Recommendations

(1) Conclusion:

The simulator training received by the Crystal River operators after TMI-2 was credited as being very beneficial in combating the February 26, 1980 event. A member of the Task Force has observed such training on undercooling and overcooling events, solid system operation, and natural circulation. At present, one week of training at the simulator is optional on most B&W requalification programs and is only required for the Oconee licensed operators. In practice, nearly all licensed operators participate in the requalification simulator training.

Recommendation:

The Task Force recommends that the one-week simulator training be required for all licensed B&W operators. The training should be oriented toward or include undercooling and overcooling events, solid system operation, and natural circulation. Upgrading of simulator performance in accordance with the recommendations of the TMI Action Plan (NUREG-0660) should be expedited.

(2) Conclusion:

There were no procedures available to the operators to specifically deal with loss of NNI/ICS. Rancho Seco is the only facility having procedures that included the effect of the loss of power supply on the total plant.

For the small-break LOCA, generic guidelines were developed by B&W and successfully implemented at each of the operating facilities.

Recommendations:

Licensees should develop and implement promptly plant-specific procedures concerning the loss of NNI/ICS power. These procedures should enable the operator to bring the plant to a safe condition. These procedures should be audited by the Office of Inspection and Enforcement. Further, the Task Force endorses the effort by B&W to develop abnormal transient operating guidelines (ATOG) and recommends that full utility support be given to this program.

(3) Conclusion:

The Task Force believes it is necessary for all B&W licensed personnel to become familiar with the Crystal River event as it affects their plant. A survey of B&W training coordinators has shown that little formal instruction has been given to date.

Recommendation:

Lectures should be developed and given promptly to all licensed personnel concerning the Crystal River event as well as their plant-specific loss of NNI/ICS analysis. A means to evaluate the training (e.g., quizzes) should be included. This training should be audited by IE inspectors.

(4) Conclusion:

There have been 24 instances of loss of NNI/ICS power supplies at B&W facilities while the reactor was critical or at power (see Appendix B, Table B.1). Twenty-two of those resulted in a reactor trip and four events in HPI actuation. The longest period of time before power was restored is believed to be 21 minutes. Had a qualified I&C technician not been available, it is unknown how long this condition would have continued.

Recommendation:

The Task Force recommends that qualified I&C coverage be provided on a round-the-clock basis at all operating B&W reactors.

6. INTERIM RELIABILITY EVALUATION PROGRAM (IREP)

6.1 IREP and Its Relationship to B&W Operating Plants

The Interim Reliability Evaluation Program (IREP) has been instituted by the Probabilistic Analysis Staff of the Office of Nuclear Regulatory Research. This program has as its charter the evaluation of the probabilities of accident sequences leading to core meltdown and (to a limited extent) the assessment of the consequences associated with the specific sequences for each U.S. operating reactor to identify only high-risk sequences. Techniques developed in the "Reactor Safety Study," WASH-1400 (i.e., event tree and fault tree techniques, accident radioactive release categorization), will be used on the individual plants in conjunction with data compiled in the Reactor Safety Study and in data collection projects begun since the release of WASH-1400 (Ref. 42). To date, the IREP study has concentrated on one reactor, Crystal River Unit 3, built by the Babcock & Wilcox Company.

The application of the reliability analysis methodology mentioned above will result in several specific products. These are:

(1) For a particular facility, the identification of accident sequences which are significant contributors to the risk to public health and safety and to the probability of a core meltdown. Thus, the program will identify plants which may have relatively high public risks associated with their operation; and

(2) The identification of common cause failure mechanisms important to individual plant systems or to groups of systems. The potential failure mechanisms considered include, for example, human interactions, and common support systems such as service water, component cooling water, and electric power.

Because of the need for timely results in the IREP study, some particular aspects of the plant either have not been included in the Crystal River study or have not been thoroughly investigated. These include:

(1) No consideration has been given to the potential effects of external events such as earthquakes, floods, nearby hazardous material-related accidents, etc.; and

(2) The details of control systems and plant instrumentation (such as non-nuclear instrumentation and the integrated control system) and fault propagation through these systems has been treated in only a qualitative manner.

6.2 The IREP Crystal River 3 Study

6.2.1 Discussion

As noted above, the initial plant selected for evaluation in the IREP study has been the Crystal River Unit 3 Nuclear Plant. Completion of this work is presently expected in May 1980. Based on the preliminary results of this study, it has been shown that:

(1) The core melt probability and public risk associated with the Crystal River plant are dominated by transient-initiated accidents;

(2) The total loss of all feedwater (main and auxiliary feedwater) is a highly significant accident; and

(3) The plant's auxiliary cooling water systems couple many of the plant's engineered safety features systems, indicating a potential highly significant common cause failure mechanism.

Because of the limitations in IREP discussed in Section 6.1, certain issues have not been resolved or failure mechanisms identified. These include:

(1) The common cause failure potential resulting from ICS failures and interactions has not been quantitatively determined; and

(2) The specific common cause failure mechanism exhibited in the February 26, 1980 event at Crystal River (loss of the 24 Volt dc power supply to NNI "X") was not identified prior to the incident.

6.2.2 Conclusions and Recommendations

(1) <u>Conclusion</u>:

The IREP Crystal River study has tentatively concluded that transient-induced accidents are highly significant contributors to the likelihood of core meltdown in the Crystal River plant. Implicit in this conclusion is the determination that support systems to vital equipment (e.g., cooling water, ac and dc power) couple this equipment to a degree that common cause failure potentials are significant.

Common cause failure modes resulting from NNI/ICS faults have been qualitatively identified, but the probabilities of such common-cause failures have not been calculated. We agree with the IREP staff judgement that a rigorous quantification of these probabilities would require a large-scale effort, exceeding the constraints placed upon the IREP study.

Recommendation:

The Task Force recommends that the IREP Crystal River study be completed and thoroughly documented in an expeditious manner, with the results provided to appropriate parts of the agency in a clear manner. At that time, we recommend the action listed below:

(a) The Probabilistic Analysis Staff consider the need for additional Crystal River work to examine particular unresolved questions which may be evident, and reexamine the scope, methods, and format of the first IREP study so that modifications may be made prior to the initiation of further IREP work; and

(b) Appropriate staff within the Office of Nuclear Reactor Regulation (in coordination with the Probabilistic Analysis Staff) make prompt determinations with respect to the need for modifications to the Crystal River plant.

7. RISK REDUCTION POTENTIAL

7.1 Introduction

The Probabilistic Analysis Staff was asked to evaluate the effectiveness of the Task Force recommendations using perspectives derived from probabilistic safety analysis and risk assessment. This chapter reports this review.

It is not possible to obtain a quantitative measure of risk reduction effectiveness for the recommendations. To do so would have required a thorough knowledge of the likelihood and consequences of the many competing accident scenarios in the plants before the alterations and a thorough knowledge of the implementation and effects of the recommendations. Such knowledge is not available at this time.

On the other hand, many qualitative insights that shed some light on the potential value of the recommendations can be developed against the background of past attempts at realistic analyses of the likelihood and consequences of nuclear accidents using probabilistic risk assessment methods. These include relationships between B&W plant characteristics and the likelihood of accidents, and judgments of the range of benefits and disadvantages of the recommendations. In many cases the recommendations suggest studies and directions in which to look for improvements rather than prescriptive fixes. The risk-based perspectives add another dimension to the definition of these suggestions. The observations about B&W safety issues and about the recommendations reported here originated in the professional judgment of experienced nuclear risk assessment engineers. They are not based on probabilistic safety analyses performed for this specific purpose.

The technique employed to arrive at these observations was to develop several tables (7.1, 7.2, and 7.3). The entries in the tables were arrived at by consensus. The assumptions, observations, and arguments that surfaced in the course of arriving at this consensus became the source for the footnotes and text.

In Section 7.2 the broad outlines of the risk picture are sketched for Babcock & Wilcox reactors. The study addresses B&W plants as they are being operated since TMI but before the recommendations contained herein are implemented. We find that the characteristics of the B&W nuclear steam supply system design and operation makes these plants much more prone to minor incidents, somewhat more prone to core damage, and no more prone to severe accidents than are other PWR designs.

In Section 7.3, the twenty-two recommendations discussed in Section 2.0 of this report are evaluated for their range of effects on the frequency of a number of particular accident scenarios and for their influence on the likelihood of incidents, minor accidents, and severe accidents.

It should be clearly understood that these observations reflect the opinions of risk assessment engineers and not the results of detailed calculations or a formal research program. As such, they should be regarded as uncertain.

7.2 Risk Perspectives for B&W Plants

A number of studies have been performed or are under way which address the realistic consequences of core melt accidents at pressurized water reactors having large dry containment buildings. These studies include the Reactor Safety Study, WASH-1400 (ref. 42), the alternate sequence and consequence analyses done in conjunction with the Kemeny and Rogovin inquiries into the accident at TMI, and some studies currently in progress on Indian Point, Zion, Calvert Cliffs and Oconee.

These studies suggest that there is a "natural" classification for accidents in dry containment PWRs. In this scheme, lines of demarcation in accident consequences correspond with lines of demarcation in terms of the functional failure of systems. There are three levels of severity in this classification. We might call them:

1. Severe Accidents,
2. Accidents, and
3. Incidents.

The basis for the distinctions are as follows: all accidents that produce any acute fatalities beyond the site boundary are predicted to entail both severe core damage or meltdown <u>and</u> gross, early containment failure. Accidents of this kind are also the only ones to produce substantial ground contamination by fallout. Such accidents dominate the risk as measured by public health and safety criteria and by offsite property damage.

Accidents - the intermediate class of incidents - may entail core damage or meltdown but do not entail gross, early containment failure. The accident at the Three Mile Island is an example. Also belonging in this class of incidents are design basis LOCA events with gross containment failure. Such accidents do not cause acute fatalities. They will not cause fallout that severely contaminates offsite land. They may - in their more serious variants - cause atent cancer casualties or groundwater contamination warranting interdiction. Accidents like these are not irrelevant to public health and safety, but they are very much less severe than the ones we have called "Severe Accidents." Unless these accidents were to be - and were to remain for a long time - very much more probable than severe accidents, they would be overshadowed in public health risk significance by the severe accidents. These accidents are, however, the dominant contributor to the economic risk borne by the plant owners relating to on-site equipment damage, as the accident at Three Mile Island indicates.

Incidents have virtually no offsite radiological consequences associated with them. Their contribution to public risk - as measured by health effects or offsite property damage - is negligible. The economic risk for the utility and its rate-payers associated with incidents tend to be smaller than or comparable to that associated with accidents. They include anticipated transients, events like the Browns Ferry fire, design basis LOCAs, etc. They do not entail significant core damage nor do they include LOCA in conjunction with abnormal post-accident containment leakage.

Accidents fall into the "severe" category only if the containment fails and the core releases much of its radioactivity. The causes of such accidents may be described as follows:

1. External missiles (e.g., heavy airplane crash) or internal missiles (e.g., the reactor vessel head) that breach the reactor coolant system, disable emergency core cooling systems and breach containment.

2. Structural collapse of the containment building which defeats the core cooling systems.

3. Loss of coolant accidents that are not isolatable and which bypass containment. (Event V in WASH-1400)

4. Failure of core cooling, failure of containment sprays, and failure of containment fan coolers.

5. A borderline case is failure of core cooling and failure of containment isolation with operable containment sprays and coolers. Such scenarios may fall in either the "severe accident" or "accident" spectra of consequences.

Accident scenarios of the first two kinds (missiles and structural collapse) have been extensively analyzed in nuclear power plants. They are believed to be extremely improbable. Probabilistic risk assessment suggests that the third kind of scenario, the interfacing system LOCA that blows down outside containment, may be among the dominant contributors to the risk from any PWR. The susceptibility of a plant depends upon the design, administrative controls, and surveillance of the reactor coolant pressure boundary valves on the larger lines that attach to the reactor coolant system and penetrate containment. It does not importantly depend upon the particular reactor design.

Risk assessment studies suggest that the fourth group of severe accident scenarios may also contain dominant contributors to the risk. These are accidents entailing failure of core cooling (leading to severe damage or melt) and also failure of containment fan coolers and sprays (leading to gross containment rupture on overpressure). Many failures in the "front line" engineered safety features are required for this to happen. For example, failure of all trains of containment fan coolers, failure of all trains of containment sprays, failure of the safety injection function and either a LOCA or a failure of main and auxiliary feedwater. The coincidental or random failure of all trains of all these "front line" engineered safety features is clearly much too unlikely to affect the risk. However, common cause failures such as fires, floods, earthquakes, or the failure of support or auxiliary systems, such as AC power, DC power, control and actuation systems, auxiliary cooling water systems, etc. can produce the many functional faults in "front line" systems from one or a very few root-cause failure events.

One example of this group of accident scenarios was found to be a dominant contributor to the risk from the PWR analyzed in the Reactor Safety Study. It entails loss of offsite power, the failure of both emergency diesel generators, and the failure of the turbine-driven auxiliary feedwater pump. All feedwater is lost, leading to the boil-dry of first the steam generators and then the reactor core. Containment sprays and coolers are also defeated by the failure of AC power sources, so this scenario belongs in the group of severe accidents.

The likelihood of these severe accident scenarios is governed by the susceptibility of the front line engineered safety features to common failure mechanisms, not to the details of the design of the nuclear steam supply systems. Therefore, there is little reason to believe that B&W plants are any more or less likely to be subject to such accidents than are other PWRs.

It is well known that the once-through steam generators employed in B&W plants hold a small inventory of secondary coolant. They boil dry more quickly than other PWR designs following a loss of all feedwater flow. Dry steam generators imply an interruption in normal reactor heat dissipation but it does not mark a point of no return for core cooling. Later restoration of feedwater may restore normal cooling for some time after steam generator dryout. All B&W plants but one (Davis-Besse 1) also have HPI pumps with high shutoff head; these pumps can drive open the pressurizer safety valves. This capability is very useful in extending the time-window within which core damage or meltdown can be avoided following an interruption in primary and secondary side cooling. Thus, most B&W plants may have as long or longer points of no return for the restoration of successful core cooling than do some other PWR designs.

Undercooling transients are more likely in plants with highly responsive OTSGs than in otherwise comparable plants with recirculating steam generators. Brief interruptions in the heat sink provided by the steam generators may cause a challenge to one of the pressurizer valves (PORV or safety valves). Thus, the B&W design tends tr e more susceptible to transient-induced LOCA. The difference between B&W and other designs is confined to the case of <u>delayed</u> auxiliary feedwater starts. Prompt AFWS starts do not cause undercooling transients. Outright (sustained) failure to start is equally serious with or without responsive steam generators. Thus, B&W plants place a premium upon the reliability with which the auxiliary feedwater starts are properly timed. The penalty for late starts is an increased likelihood of transient-induced LOCA.

The most prominant common-cause failure mechanism we can identify that causes both delayed auxiliary feedwater starts and sustained ECCS failures lies in operator error. A practice of trying to avoid over-cooling incidents tends to make such errors more likely. On the other hand, the experience of having had a TMI accident, the operator retraining it spawned, and the other changes made since the accident have gone a long way to reduce the likelihood that such scenarios would start or would progress to core damage once started. Nevertheless, our event tree-fault tree studies suggest that transient induced LOCA which cannot be isolated and which occurs in conjunction with ECCS failure may be among the dominant routes to core damage, i.e., to an accident, although we think it very unlikely that such a scenario would also entail the failure of containment fan coolers as well as sprays. Thus, transient-induced LOCAs should not be prominant causes of severe accidents.

It is known that B&W plants have somewhat more frequent trips than do other PWRs, particularly since the TMI-inspired alterations to the trip setpoints. These excess trips seem to be originating from minor secondary side transients and non-safety-grade instrumentation faults. These transient initiators do not correlate with the occurrence of massive, common-cause failures in the engineered safety features - with a couple of noteworthey exceptions - so they are not expected to increase the frequency of the risk-dominant severe accidents in B&W plants above the level expected for other PWR designs.

The two exceptions deserve closer scrutiny. The Non-Nuclear Instrument (NNI) bus faults that occurred at Rancho Seco and Crystal River caused massive faulting of the instruments upon which the operators depended to understand the status of the plant. It could be postulated that such faults could lead to the kinds of operator errors that could give rise to severe accidents. For a number of reasons, severe accidents via such routes seem very unlikely: (1) In the post-TMI environment, it is unlikely that operators would override the autostart of engineered safety features while their instruments are obviously faulted; (2) It is unlikely that operators would shut off containment fan coolers, even under circumstances in which they might mistakenly shut off ECCS or containment sprays; (3) All historical instances of NNI failures have been repaired before the point of no return for a severe accident; and (4) The attention given to the recent Crystal River and other incidents has alerted operators to the symptoms, consequences, and the ways to deal with NNI failures.

Another hypothetical way that the somewhat higher transient rate at B&W plants might affect the frequency of high-risk accident sequences is through failures of offsite power. Loss of offsite power may originate outside the plant or be precipitated by a plant trip. Studies performed for WASH-1400 suggested that most instances of loss of offsite power originate outside; the overall frequency of the loss is quite insensitive to the plant trip rate according to industry statistics. There may be exceptional sites where this is not true, however. To the extent that B&W plants trip more often than other PWRs, they place a correspondingly greater safety premium upon the reliability with which the grid, the switchgear and the startup transformer picks up plant auxiliary loads. We expect for most B&W plants that the somewhat higher trip rate has a negligible effect on the likelihood of severe station blackout accidents.

In summary, then, the enhanced frequency of transients in B&W plants is not believed to importantly affect the likelihood of severe accidents.

Another concern with B&W plant design and operation is the comparatively high frequency of overcooling transients following reactor trip. In some of these transients the shrinkage of reactor coolant causes the pressurizer level to go off-scale low and/or the pressure to fall to the ECCS actuation point. Even if the pressurizer bubble is drawn into a reactor coolant loop and the reactor coolant pumps are tripped, we see no difficulty in sustaining convective circulation in the unaffected loop and sustaining or restoring it in the loop with some of the steam bubble. Frequent ECCS actuation in such events is significant in the ways it affects operator behavior. Frequent spurious ECCS actuations could tend to induce operators to disable or override actuation signals important to safety.

In the post-TMI environment, we think that operators would correct such errors long before they resulted in core damage in all but the fastest-moving accidents and would correct such errors before containment failure results in a severe release in virtually every case. Thus, the "cry wolf" effect of overcooling transient-induced spurious ECCS actuations might have some effect on the frequency of core damage (accidents) but a negligible effect on the frequency of major releases, i.e., severe accidents.

ECCS actuations in overcooling transients - apart from their effect on operator behavior - are expected to have very little effect on the likelihood of core damage. If ECCS fails to start, no harm is done as it isn't really needed in an overcooling accident. There is a very slight chance that HPI or the affected makeup pump might be critically needed before it could be repaired. On the other hand, such challenges provide experiences more closely resembling genuine demands than do surveillance tests, so these nuissance demands also help to debug the system. On balance the prospect of ECCS failures in these overcooling transients has very slight and counterbalancing effects on the likelihood of core damage and a negligible effect on severe accidents.

If ECCS does start in these overcooling transients, the operators may leave it on long enough to lift the pressurizer PORV or possibly a safety valve. This, in turn, opens the possibility of a spillage of reactor coolant and perhaps a stuck-open valve, i.e., a LOCA. In the worst case of a stuck-open, non-isolatable pressurizer valve, ECCS must work to sustain core cooling. However, ECCS will have higher-than-normal reliability under these conditions because its successful start caused the LOCA in the first place. There is no reason to believe that such incidents are likely to be coupled with ECCS failure or with the failure of containment fan coolers or sprays.

It has been suggested that a reactor trip together with a failure to throttle main feedwater in a B&W plant would rapidly fill the OTSG's and result in water in the main steam lines. No such instances have occurred but comparable upsets in the Integrated Control System have been observed. The main steam lines and valves may not be qualified for the weight or the water-hammer potential associated with this scenario; they might rupture. The characteristic range of times to fill the steam generators and main steam lines is a very few minutes, perhaps too rapid to give much confidence that the operators would consistently trip the feedwater pumps or stop valves in time to avoid main steam line breaks.

Such scenarios would affect the risk of severe accidents only if the break produced flooding that defeats support systems for essentially all of the active engineered safety features, i.e., essential DC power, AC power, or possibly essential auxiliary cooling water systems, and do so with a probability that rivals station blackout or Event V. Such scenarios would have a significant effect on the likelihood of core damage only if the flooding defeats emergency feedwater and HPI (feed and bleed cooling) and does so with a probability that rivals other common-cause or multi-fault scenarios such as loss of all feedwater and HPI failure.

In either the case of accidents or severe accidents, the significance of the water-solid main steam line break scenarios seems to rest upon the potential for massive flood damage in essential compartments of the auxiliary building. If such flooding does not take place, there appears to be little direct threat to ultimate core cooling or containment integrity.

The susceptibility of B&W plants to loss of all essential AC or DC power or loss of all HPI and EFW due to water-solid main steam line breaks and subsequent flooding should be reviewed. If a deterministic analysis suggests a real possibility of such a scenario, then a probabilistic evaluation should be performed.

These considerations of B&W plant characteristics are summarized in Table 7.1. We conclude that B&W plants are not significantly different from other PWRs in their vulnerability or susceptibility to severe accidents - those that dominate the nuclear risk.

B&W plants have a different profile of susceptibility to core damage accidents than do other PWRs. They are more likely to incur transient-induced LOCA but the ones with high head HPI pumps may be less likely to incur core damage from a loss of all feedwater. B&W plants are more likely than other PWRs to have over- or undercooling incidents, transient-induced LOCA, etc.

Table 7.1

Effect on Frequency of Incidents of B&W Plant Characteristics or Concerns

B&W Plant Characteristic or Concern	Effect on Frequency* of: Severe Accidents (large release)	Accidents (small release)	Incidents (no abnormal release)
1. Short time to SG dryout following loss of feedwater	small[1]	small[1]	large[2]
2. Frequent undercooling transients	small[3]	large[4]	large[4]
3. Heightened trip frequency	negligible (neg)	small	large[5]
4. NNI/ICS faults	neg	medium[6]	large[2]
5. Frequent overcooling transients			
a. Loss of PRZR level	neg	neg	large[2]
b. Nuissance ECCS actuation	neg	medium[7]	large[2]
6. Overfeed main steam line rupture	neg?[8]	neg?[8]	?
7. Feed and bleed capability (high head HPI)	moderate improvement[9]	large improvement[9]	large

Notes:

*Baseline of comparison is the WASH-1400 risk picture for Surry.

[1]Loss of steam pressure to drive turbine-driven emergency feedwater pumps or restore main feedwater may be more likely with the OTSG design.

[2]Faults of this kind intrinsically qualify as abnormal occurrences or disruptive events.

[3]The direct effect on the frequency of dominant sequences is negligible, however, the pronounced effect on the frequency of core damage in conjunction with coincidental containment failure might rival dominant sequences in probability.

Table 7.1 (Cont.)

[4]Delayed start of auxiliary feedwater following loss of main feedwater is more likely to lift a pressurizer valve in B&W plants. This increases the frequency of transient-induced LOCA in positive association with faults that might degrade the reliability of HPI as well as auxiliary feedwater. The Lessons of TMI have already reduced this likelihood of serious outcomes for these scenarios. Total failure of all feedwater and of HPI is equally problematic in all PWRs.

[5]Frequent trips are intrinsically a cause for concern.

[6]Effect via operator error or transient-induced LOCA.

[7]Effect via long term influence on operator behavior.

[8]Neither the possibility nor the likelihood of this hypothetical group of accidents has been verified.

[9]Feed and bleed can provide an option for core cooling in the event of a total loss of feedwater. It may also provide a later point of no return for saving the core during primary coolant boiloff.

7.3 Observations on the Task Force Recommendations

Table 7.2 reports the judgment of the review group from the Probabilistic Analysis Staff of the effect of the Task Force recommendations on the likelihood or severity of a number of accident scenarios: loss of main feedwater, loss of main feedwater due to ICS or NNI faults, loss of offsite power, small LOCA, station blackout, anticipated transient without scram, and steam generator overfill.

Table 7.3 is very much like Table 7.2 except that the columns treat incidents by the severity of outcome rather than by the kind of initiating event. In this table, we have assessed the potential of each recommendation for reducing the likelihood and/or severity of the three categories of events (incidents, accidents, and severe accidents). That is, each entry in the table may be interpreted as the potential for the specific recommendation reducing (or increasing) the likelihood of the particular event category and/or improving (or harming) the plant's capability to cope with the events in that category. Thus, some recommendations may be of high potential benefit in reducing the likelihood of a severe accident but of low potential benefit in coping with an ICS/NNI fault like that experienced at Crystal River. Others may be of some moderate benefit in reducing the frequency of overcooling incidents, of moderate benefit in reducing the likelihood that such an incident will propagate into an event causing core damage (the "accident" category), but of negligible benefit in reducing the likelihood of severe accidents.

Table 7.2

Effect of Task Force Recommendations on Particular Plant Transients

Task Force Recommendation	Loss of MFW		Loss of MFW From ICS Faults		Loss of Offsite Power		Small LOCA		Station Blackout		ATWS		OTSG Overfill	
	Pos	Neg	Pos	Neg	Pos	Neg	Pos	Neg	Pos	Neg	Pos	Neg	Pos	Neg
1. AFWS Upgrade to an ESF system														
a. Fluid System Upgrade	M-H		L		M-H		M-H		H		M-H			L
b. External Event Qualification	L		L		L		L		L		L		L	
2. ESF Automatic Initiation and Control of AFWS	H		H		H		H		H		H			L
3. Diversely-Powered Auxiliary Feedwater Pump for Davis-Besse	H		H		M-H		M-H		L		H			L
4. Modifications to the Steam and Feedwater Line Break Detection and Mitigation Systems	H		H		H		M		?	?	H			
5. Improvements to the Integrated Control System and NNI														
a. Channelizing sensors, etc.	L	L	L		L	L	L	L	L	L	L	L	L	M
b. Meter failure position	L	L	M		L	L	L	L	L	L	L	L	L	L
c. Failures clearly indicated	L		M		L		L		L		L		L	
d. Reversion to manual control	L	L	M		L	L	L	L	L	L	L	L	L	L
e. Loop indication separation	L	L	M		L	L	L	L	L	L	L	L	M	M
f. Recommendations from ICS reliability analysis	L	L	M	L	L	L	L	L	L	L	L	L	M	L
g. Recommendations from INPO/NSAC Crystal River report	L		M	L	L		L		L		L		L	
h. Follow-up to IE Bulletin 79-27	H		H	L	H		H		L		L		M	M
6. Installation of a Safety Grade Panel of Vital Instruments	H		H		H		H		H		H		H	

Table 7.2 (Cont.)

Task Force Recommendation	Loss of MFW		Loss of MFW From ICS Faults		Loss of Offsite Power		Small LOCA		Station Blackout		ATWS		OTSG Overfill	
	Pos	Neg	Pos	Neg	Pos	Neg	Pos	Neg	Pos	Neg	Pos	Neg	Pos	Neg
7. Improved Use and Display of In-Core Thermocouple Indication	L		L		L		L		L		L		L	
8. Safety Grade Vent/Purge Isolation on High Radiation Signal	L		L		L		L		L		L			
9. System Response Modifications to Prevent Pressurizer Level Loss and ECCS Actuation	L		L	?	L		M		L		L		H	
10. Study of Means to Improve the Response of the OTSG	?		?		?		?		?		?		?	
11. Elimination of Post-Reactor Trip Operator Actions	L		M		M		M		L				H	M
12. Instrumentation and Control Technician Be Assigned to All Shifts	L	L	M	L	L	L	M	L	L	L	M	L	L	L
13. Operator Training on the Crystal River Incident 14. Development of Plant Specific Procedures for Loss of ICS/NNI	M		H		M		H		L		L		M	
15. Increased Simulator Training	M	L	M	L	M	L	M	L	M	L			M	L
16. Criteria for Restarting Reactor Coolant Pumps	M		M		M		M		M		?			

Table 7.2 (Cont.)

Task Force Recommendation	Loss of MFW		Loss of MFW From ICS Faults		Loss of Offsite Power		Small LOCA		Station Blackout		ATWS		OTSG Overfill	
	Pos	Neg	Pos	Neg	Pos	Neg	Pos	Neg	Pos	Neg	Pos	Neg	Pos	Neg
17. Alternative Solution to PORV Unreliability and Safety System Challenge Rate Concerns	M		L		L		M	M	L		L	L	L	
18. Completion of IPEP Crystal River Study	H?				H?		H?		H?					
19. Performance Criteria for Anticipated Transients	?		?		?		?		?		?		?	
20. Criteria for Reactor Coolant Pump Trip in Small LOCAs	L		L		L		L	L	L		M			
21. Reevaluation of AFWS Injection Point into the Steam Generators	L	L	L	L	L	L	L	L	L	L	L	L	L	
22. Study of Operator Errors in B&W Plants	L		L		L		L		L		L		L	

Table 7.3

Effect of Task Force Recommendations on Severe Accidents, Accidents, and Incidents

	Potential Benefit SA	Potential Benefit A	Potential Benefit I	Potential Detriment SA	Potential Detriment A	Potential Detriment I
1. AFWS Upgrade to an ESF system						
a. Single Failure Criterion*	L	L	L	ε	ε	ε
b. Pedigree (N-Stamp, QA)	ε	ε	ε	ε	ε	ε
c. Safety Grade Power Supplies*	L	L	L	ε	ε	ε
d. Seismic and External Event Qual.	L	ε	ε	ε	ε	ε
e. Technical Specifications	M	M	M	ε	ε	ε
f. Main Steam and FW Line Break Criteria	ε	ε	ε	M	L	L
g. Diversity of Power Supplies	H	M	L	ε	ε	ε
h. Other Requirements (see text)	H	H	L	ε	ε	L
*Most plants already comply; improvement might be large in those (if any) that do not.						
2. ESF Automatic Initiation and Control of AFWS						
a. Safety Grade Control and Instrumentation Independent of ICS/NNI	M	H	H	ε	ε	L
b. Autostart to avoid dry steam generators	ε	M	M	ε	ε	M
c. Throttle AFWS to avoid overcooling of steam generators	ε	L	M	L	M	L
d. Feedwater termination to prevent overfill	ε	L	L	M?	H?	M?
3. Diversely-Powered Auxiliary Feedwater Pump for Davis-Besse	H	H	M	ε	ε	L
4. Modifications to the Steam and FW line Break Detection and Mitigation System	M	M	H	ε	ε	ε
5. Improvements to the ICS and NNI						
a. Channelized signals	ε	L	L	ε	ε	L
b. Evaluate mid-scale instrument failure mode	ε	L	L	ε	ε	L
c. Failures clearly indicated	ε	L	L	ε	ε	ε
d. Reversion to manual control	ε	ε	ε	ε	L	M
e. Loop indication separation	ε	L	L	ε	ε	L
f. Recommendations from ICS reliability analysis	ε	L	M	ε	ε	ε
g. Recommendations from INPO/NSAC Crystal River report	ε	L	L	ε	ε	ε
h. Follow-up to IE Bulletin 79-27	M	H	L	ε	ε	ε

Table 7.3 (Cont.)

	Potential Benefit			Potential Detriment		
	SA	A	I	SA	A	I
6. Installation of a Safety Grade Panel of Vital Instruments	M	H	H	ε	ε	ε
7. Improved Use and Display of In-Core Thermocouple Indication	ε	L	L	ε	ε	ε
8. Safety Grade Vent/Purge Isolation on High Radiation Signal	ε	L	M	ε	ε	ε
9. System Response Modifications to Prevent Pressurizer Level Loss and ECCS Actuation	ε	L	M	ε	ε	ε
10. Study of Means to Improve the Response of the OTSG	?	?	?	?	?	?
11. Elimination of Post-Reactor Trip Operator Actions	ε	L	L	ε	L?	L?
12. Instrumentation and Control Technicians Be Assigned to All Shifts	L	M	M	ε	L	L
13. Operator Training on the Crystal River Incident 14. Development of Plant-Specific Procedures for Loss of ICS/NNI	M	H	H	ε	ε	ε
15. Increased Simulator Training	ε	M	M	ε	L	L
16. Criteria for Restarting Reactor Coolant Pumps	L	M	M	ε	ε	ε
17. Alternative Solution to PORV Unreliability and Safety System Challenge Rate Concerns	ε	L	M	ε	L	L
18. Completion of the IREP Crystal River Study	?	?	?	?	?	?
19. Performance Criteria for Anticipated Transients	?	?	?	?	?	
20. Criteria for Reactor Coolant Pump Trip in Small LOCAs	ε	M	M	ε	L	ε

Table 7.3 (Cont.)

	Potential Benefit			Potential Detriment		
	SA	A	I	SA	A	I
21. Reevaluation of AFWS Injection Point into the Steam Generators	ε	ε	L	ε	L	L
22. Study of Operator Errors in B&W Plants	ε	ε	ε	ε	ε	ε

For each accident grouping, there are two columns in Table 7.2 or 7.3 labeled "Pos" and "Neg." "Positive" denotes the benefit to be expected from the sound implementation of the recommendation. "Negative" denotes the potential for increased competing risks that might arise from the recommendation. For example, an alteration to the Integrated Control System could make one failure mode less likely and other failure modes more likely. We record both effects, the improvement under "Pos," the degradation under "Neg." The comments and interpretations underlying these judgments are summarized in the text below.

The entries in the tables are interpreted as follows:

1. H - High

   The recommendation is judged to have a substantial effect on a dominant contributor to the likelihood of accidents in the group of accidents.

2. M - Medium

   The recommendation is judged to have a moderate effect on a dominant contributor or a major effect on contributors that are only moderately likely to have a significant influence on the overall frequency of accidents of the type under consideration.

3. L - Low

   The overall effect on the likelihood of accidents is judged to be low. That is, the recommendation may have little effect, or it may have a strong effect on factors not bearing directly on the dominant contributors to the class of accidents under consideration.

4. Blank or Epsilon (ε)

Negligible effect.

A discussion of each recommendation follows.

1. <u>Upgrade the Auxiliary Feedwater System (AFWS) Fluid System to an ESF System</u>

In this recommendation, the Task Force calls for the improvement of the "fluid-moving" aspects of the AFWS to ESF requirements. The actuation and control aspects are treated in recommendation 2. ESF qualification entails several facets:

a. Single failure criterion

We believe almost all B&W plants have an AFWS already meeting the single failure criterion for its mechanical aspects. Thus, we think the effect of this recommendation is small. Nonetheless, its imposition is desirable, because a violation of the single failure criterion could severely compromise the reliability of the AFWS.

b. Pedigree requirements

Safety qualification normally entails a number of quality assurance and code requirements. As applied to pipes, pumps and valves, these criteria tend to bear upon pressure boundary integrity rather than active failure reliability. Since pipe breaks are a negligible contributor to the functional unavailability of the AFWS, there is very little benefit to be gained from a retroactive requirement to upgrade the pedigree of piping, valves and pumps (presuming that the equipment now installed is already of goo, quality).

c. Class IE power supplies for motor-operated pumps and valves

We believe most plants already comply so that the effect of the recommendation will be small. Nonetheless, this recommendation is important as an instance of non-compliance could compromise system reliability.

d. Seismic Category I qualification

Seismically-induced loss of main feedwater is sufficiently probable to warrant a requirement to provide an engineered safety feature qualified to cool the core under this circumstance. However, it is not so common an initiating event that diverse as well as redundant means are needed. We recommend that licensees and applicants be given the option of selecting either primary system cooling (feed and bleed) or secondary system cooling (auxiliary feedwater) as the designated, qualified method of cooling the core following a seismically induced loss of main feedwater.

e. Technical specifications

Safety qualification implies the imposition of technical specifications and finite allowable outage times for periods during which portions of the AFWS are out of service. These can have a moderate to large effect on AFWS reliability and thus on risk.

f. Main steam and feedwater line break design bases

Main steam and feedwater line breaks have been taken as design basis challenges for the AFWS in some but not all operating PWRs. AFW must be isolated from the affected steam generator and yet AFW must be supplied to the surviving steam generator(s) despite a single active failure.

Such accidents pose very little risk. They are rare and they do not directly threaten core cooling. We see virtually no risk reduction potential in extending these requirements to all PWRs, and the requirements might safely be relaxed where the provisions for automatic isolation of the "affected" steam generator or the valving necessary to satisfy the single failure criterion is found to degrade AFWS functional reliability for the very much more common loss of feedwater events.

g. Diversity of power supplies

Branch Technical Position ASB 10-1 currently requires diverse power supplies for AFWS pumps. The concept of designing out the susceptibility of the AFWS to failure in the event of a common cause failure of all sources of motive power, such as all AC power or all steam, can have a very large risk reduction potential. However, the requirement needs strengthening to include not just pump power supplies but valve and support systems as well. There should be at least one train of the

AFWS that is capable of starting and running for each of the following circumstances:

1. Loss of power on all essential and non-essential switchgear buses.
2. Loss of steam pressure in both steam generators.
3. At least one train should fail on rather than off if the corresponding control power supplies (DC or AC instrument power) were to fail off.

h. Other requirements

Most B&W plants have a two train AFWS. There is a limit to the reliability improvement that can be achieved without adding a third train. Loss of main feedwater is a very frequent challenge. With two train AFWS designs - even ones of comparatively high reliability - loss of <u>all</u> feedwater is a rare but distinctly credible event. We judge that a return interval of once in a thousand reactor years is about the best one might confidently expect for loss of all feedwater in PWRs having two train AFWS designs. A case can be made for the provision of an add-on, third train of the auxiliary feedwater system which does not depend upon the same support and auxiliaries as does the principal two-train system. However, the case for such an add-on may be less compelling in B&W plants with a demonstrated feed and bleed cooling capability than it is in plants with comparatively low head HPI since they have alternate means of core cooling.

2. ESF Automatic Initiation and Control of the AFWS

This recommendation is primarily concerned with the need for a safety grade system for initiation and control of the AFW system independent of the ICS/NNI. Also included within the recommendation are: a call for an appropriate selection of initiating signals such that the undercooling and overcooling transients experienced during the transition from main to auxiliary feedwater are minimized in severity; an inclusion within the steam generator level control of an overcooling protection capability; and a feedwater termination signal to prevent overfilling of the steam generators.

The most important part of this group of recommendations deals with the provision of an AFWS autostart system that is capable of responding in the event of a loss of main feedwater and which is independent of the ICS or its power supplies. The key to a large improvement in safety is to assure that the kind of failure events that may cause a loss of main feedwater will not also disable the AFWS.

Apart from this elimination of common cause failure susceptibility - which has large risk reduction potential - the redundancy and IE qualification requirements associated with safety grade actuation is expected to produce a small improvement in system reliability.

The selection of autostart actuation points to minimize the likelihood or severity of over- or undercooling incidents is clearly desirable provided that it doesn't introduce new system failure modes. That 's, a provision to delay or disable an autostart to avoid an overcooling transient ought not to have, as a failure mode, the outright disabling of the autostart system.

The recommendation to provide throttling of the AFWS to prevent overcooling is directly related to the discussion above concerning the safety grade level control. We believe that providing such level control is desirable, will help to some degree to reduce the frequency of overcooling events, and to a lesser extent reduce the likelihood that such events propagate into accidents involving core damage.

The recommendation to terminate feedwater supply to prevent an overfill condition appears to be more appropriate for the case of the main feedwater system rather than the AFWS. However, even for the former system, provisions to override the ICS and trip or throttle to avoid grossly overfilling the steam generators may - through nuissance trips - degrade plant safety by as much as this proper action may increase it. If such a protective system is deemed to be necessary, great care should be employed to design it for a very low nuissance trip rate.

Provisions to throttle or trip the auxiliary feedwater system to avoid grossly overfilling the steam generators (beyond that provided by the upgraded AFWS control system) is even more subject to adverse side effects. "Protective" systems that have the effect of isolating a reactor from its heat sink - as these do - should be avoided if possible, and entered into only with great care, thorough reliability analysis, and a careful investigation of adverse side effects. We expect that a system to trip or throttle the AFWS on very high steam generator level may have the net effect of increasing the risk.

3. Diversely-Powered Auxiliary Feedwater Pump for Davis-Besse

In this recommendation, the Task Force has noted and addressed their concern about a unique feature of the present Davis-Besse AFWS. In this plant, both AFWS pumps are driven by steam drawn from the main steam lines. The Task Force concern about this configuration was that temporary interruptions in feedwater flow to the steam generators can result in dry-out; subsequent attempts to initiate the AFWS may then be compromised by lack of motive steam. Potentially aggrevating this problem is the failure to isolate the steam lines. During the February 26, 1980 Crystal River incident and the March 20, 1978 Rancho Seco "light bulb" incident, the steam generators dried out. The remaining steam mass trapped within the steam generators was depleted by the continued operation of the main feedwater pump turbines, although the feedwater discharge valves were closed so there was no water mass replenishment by feedwater injection.

Other recommendations of the Task Force address the reduction in frequency of events which would result in steam generator dry-out. However, because such events cannot be eliminated completely and because the AFWS is a critical feature for coping with feedwater transients and some small LOCAs, we believe that a diversely-powered AFW pump for Davis-Besse is of high value in reducing the likelihood of severe accidents and accidents, and moderate value for incidents. This is further reinforced by the more limited capability of the Davis-Besse plant to cope with a total loss of feedwater because of the relatively low shutoff head of their HPI pumps.

4. <u>Modifications to the Steam and Feedwater Line Break Detection and Mitigation Systems</u>

Installed in most of the B&W plants are systems intended to cope with the effects of a main steam line break inside the reactor building. These detection and mitigation systems are designed to detect the affected steam generator and isolate feedwater flow to it. Licensing calculations indicate that, for the assumed conditions, continued flow of feedwater presents the possibility of reactor building pressure exceeding its design pressure and a possible return to criticality in the core (due to the severe RCS overcooling combined with a stuck-out control rod). This recommendation of the Task Force addresses the concern that such systems can initiate feedwater transients (by spurious operation) and, under certain circumstances, prevent feedwater delivery during a (non-steam line break) transient.

We believe that these detection and mitigation systems can be highly significant common-cause failure mechanisms, being both the cause of a feedwater transient and interfering with the subsequent necessary delivery of emergency feedwater (as occurred during the September 24, 1977 Davis-Besse transient). For this reason we believe that this Task Force recommendation is of moderate value in reducing the likelihood of severe accidents and accidents, and high value for incidents. We note, however, that the goal of the recommendation, to eliminate the potential for adverse interactions resulting from these detection and mitigation systems, may be very difficult to accomplish. We believe that it is important not only to consider design changes for these systems but to also reconsider the actual need for such systems. If the requirement for automatic isolation of the auxiliary feedwater system (<u>vis a vis</u> operator intervention) is an artifact of conservative reactor building pressure calculations, it may be preferable to remove the detection and mitigation system's control of the <u>AFWS</u>, rather than attempting to design a more sophisticated system.

5. <u>Improvements to the Integrated Control System and Non-Nuclear Instrumentation</u>

It is clearly evident from the Crystal River incident and other similar events that the ICS and NNI in B&W plants can be both the initiator of a transient event and a compromising agent in the plant's and operators' attempts to mitigate the transient's effect. While other Task Force recommendations deal with ways to improve the mitigating capabilities of the plant and its operators, this recommendation addresses means for improving the reliability of the ICS/NNI so that its frequency of failure is reduced and its failure not so severe.

Because this recommendation deals strictly with means to improve the ICS/NNI, we believe that it can provide significant benefit only for transient events initiated by faults in these systems. Thus (as Table 7.2 illustrates), we feel that these recommendations are, in general, of relatively low merit for events such as "normal" losses of the main feedwater system, small LOCAs, etc. In some cases, we also believe that specific recommended modifications might have slight negative implications. For example, modifications in meter failure position may impede operator actions in other events (until such time that the operators become thoroughly familiar with the new indications and the altered system is debugged).

For the case of ICS/NNI-initiated transients, we believe that the specific Task Force recommendations are generally of low to moderate importance in reducing the likelihood of incidents, while of generally low value for accidents, and negligible value for severe accidents. Again, since instrumentation and control equipment modifications will inevitably require some time for adjustment on the part of the operators and the I&C technicians, some increased likelihood in human error and frequency of ICS/NNI failures can be expected for some time.

We also believe that certain recommendations are of relatively more importance for the ICS/NNI-initiated type of transient. Specifically, we believe to be more important the capability for bus transfer in the event of power supply faults and the follow-up actions to IE Bulletin 79-27, which addresses on a plant-specific basis the capability to cope with power-failures to the ICS/NNI. We also note that recommendation 5d (reversion to manual control) could be of some low to moderate value (for accidents) if this change were to remove the possibility that faults could disable both automatic and manual control of the plant secondary side. If the recommendation does not accomplish this, then we believe it to have negligible importance.

6. Installation of a Safety Grade Panel of Vital Instruments

This Task Force recommendation is similar to the Lessons Learned Task Force recommendation 7.2 and calls for a safety-grade panel of instruments in the control room which is independent of other instruments, their power supplies, etc. and their associated potential for common-cause failures.

The installation of such a safety-grade panel would provide the operating crew with a credible source of information during events which affect other plant instrumentation. Other Task Force recommendations have as a goal the reduction in frequency of such losses of instrumentation; however, since such losses cannot be eliminated (or even substantially reduced in frequency), we believe that such a safety panel is important. Since it is a virtual certainty that operating crews will in the future be faced with faulted non-nuclear instrumentation during a transient, such a safety panel can significantly improve the likelihood that the operators will correctly diagnose and cope with the transpiring events (presuming that these instruments are powered from appropriate supplies, e.g., batteries). For this reason, we believe that this recommendation has high value for incidents, high value for accidents, and moderate value for severe accidents.

7. <u>Improved Use and Display of In-Core Thermocouple Indication</u>

This Task Force recommendation has two aspects: the improvement in the capability to use the in-core thermocouples (as one input to the subcooling meter); and the improvement in the display capability of the thermocouple indications, so that trend information in core outlet temperature (temporal behavior, regional variations, etc.) is available to the operators. Apparently, thermocouple indications were used by the Crystal River operators during the February 26, 1980 incident while much of the other instrumentation was failed or of questionable credibility.

As we have discussed above, it is highly likely that instances of large-scale instrumentation failures will in the future be experienced by operating crews, so that reliable information from diverse sources such as the in-core thermocouples will be important to the operator response to the events. In this sense, this recommendation is coupled with Task Force recommendation 6 (Safety-Grade Vital Instrument Panel). Because the latter recommendation calls for the provision of several indications of RCS status, we believe that it overshadows the potential benefit resulting from the improved use and display of the thermocouple indication. Thus, while we feel that better use and display of the thermocouple indication would be a desirable capability, we believe that the installation of the "safety panel" is distinctly more important. In this context, this recommendation appears to be of low importance for incident and accident mitigation and of negligible importance for severe accidents.

8. Safety-Grade Vent/Purge Isolation on a High Radiation Signal

This Task Force recommendation calls for the installation of safety-grade isolation equipment on the reactor building vent/purge system which would be actuated on high radiation levels in the reactor building. This is of concern because, for some events, isolation of the vent/purge system on high building pressure or low RCS pressure might not occur until after the release of some radioactive material. For example, for a total loss of feedwater accident (i.e., both main and auxiliary feedwater fail), RCS pressures would climb rather than drop sufficiently to cause the building isolation on low RCS pressure. Further, the operation of the purge might prevent building pressures from reaching the other isolation setpoint; thus, automatic isolation might not occur. Under such circumstances, operator actions to isolate the vent/purge system might not occur until some material (e.g., radioactive gases released from the expelled coolant) has escaped through the system. To cope with such a situation, a vent/purge system isolation on high radiation level in the reactor building has been recommended.

In essence, the intent of this recommendation is to substitute automatic isolations (on high radiation) for operator-initiated isolations for that class of accidents where the "normal" isolation-initiating signals would not be received. The consequences of not providing such an isolation can be thought of as the difference in the magnitude of release if an automatic isolation were to occur and if the isolation were dependent on operator action. Since the concentration of radioactive material in coolant is relatively low, we believe that the increased time required for human actuation of the vent/purge system isolation would result in only a small difference in the radioactive release. For this reason we believe that this recommendation is of negligible value with respect to severe accidents, and low value for accidents. We also believe, however, that it could be important (in the severe accident category) to assure that these valves fail closed on loss of power, so that isolation occurs in the event of such potentially severe accidents as station blackout.

We note that the above conclusions on the relative merit of this recommendation are based on the conclusion that small releases of radioactive material during an incident will result in negligible health effects within the surrounding public. If, however, the objective is to prevent any release of radioactive material, this recommendation clearly is more desirable; for this reason we believe it is of moderate value with respect to coping with incidents. We also note that an anticipatory trip of the containment purge isolation valves could also be triggered on high pressure in the reactor coolant drain tank.

9. System Response Modifications to Prevent Pressurizer Level Loss and ECCS Actuation

Following a reactor trip in a B&W plant, the reactor coolant undergoes significant contraction as it cools; as a result, the pressurizer level and RCS pressure drop substantially. To cope with this, operators are trained to quickly isolate letdown flow and start an additional make-up (HPI) pump, so that shrinkage is accounted for by additional coolant injection into the RCS. Even with such operator intervention, however, these plants have a history of occasional secondary side malfunctions leading to reactor trips, losses of pressurizer level, and ECCS/HPI actuations (on low RCS pressure). This Task Force recommendation calls for the examination of means to reduce the severity of the post-trip RCS transient, so that the frequency of level loss and HPI actuation is reduced.

A reduction in the frequency with which pressurizer level is lost and/or ECCS is actuated in overcooling accidents is useful in several ways. Frequent ECCS actuations due to overcooling transients may condition operators to expect all ECCS actuations to be spurious and encourage them to disable the autostart of emergency feedwater (to avoid the overcooling) or to override the ECCS start without positively determining that there is no genuine need for it. Thus, it is important to avoid or counteract (with training) this effect on operator behavior.

Apart from the effect on operator behavior, the frequency of overcooling transients leading to loss of pressurizer level or spurious ECCS actuation has little bearing on the likelihood of core damage and still less on public health and safety. The failure of ECCS under such challenges has almost no safety penalty since ECCS is not really needed in this scenario; it offers an opportunity to gain experience and debug the system. The success of ECCS under such challenges may lead to increased challenges to pressurizer relief and safety valves, which might then fail open. However, the ECCS system needed to mitigate such failures must be accorded higher-than-average reliability in such situations because its operability was responsible for the opened valve in the first place.

Thus, virtually all of the moderate significance (with respect to the incident accident category) attributed to this recommendation relates to its effect on operator behavior. We also believe it is of low value with respect to reducing the likelihood of accidents, with negligible value in the severe accident category.

10. Study of Means to Improve the Response of the Once-Through Steam Generator (OTSG)

In this recommendation, the Task Force has addressed the concern of the relationship of the relatively small OTSG secondary side coolant inventory to the overall "sensitivity" of the B&W plant. The recommendation suggests that both active and passive means to improve the OTSG response be investigated.

We recognize as the Task Force did that there are a number of ways possible to improve the OTSG responsiveness. Such design changes to the OTSG obviously have the potential for significantly improving the overall behavior of the plant during feedwater transients (or, if poorly designed, having negative impact). Equally obvious is that, since we do not now know what the study results would show, we cannot pass judgment on its relative merit. For this reason, we believe that it is sufficient that we concur on the Task Force recommendation that such a study be undertaken.

11. Elimination of Post-Reactor Trip Operator Actions

As was described in our discussion of recommendation 8 above, following a reactor trip in B&W plants, the operators are required to take certain actions to help minimize the post-trip pressurizer level and RCS pressure decrease. Additional operator actions are also required in the event of a small LOCA to balance HPI flows, etc. This Task Force recommendation calls for decreasing the burden placed on the operators during this time period by reducing or eliminating (automating) the immediate manual actions required by the emergency procedures.

By removing those requirements on the operator to act, one allows the operator the opportunity to think more broadly about his situation. For this reason, we believe that the reduction in the demands placed on the operating crew during the early phases can have an important impact on their capability to cope with the accident, i.e., reduce the likelihood of errors during the event.

Thus, we believe that this recommendation has negligible potential for reduction in the likelihood of severe accidents, and low benefit for accidents and incidents.

We note that, under certain circumstances, the automation of post-trip actions can also produce adverse effects. Care should be taken when automating certain functions (e.g., letdown isolation) to avoid potential adverse interactions with ICS/NNI. Since we do not believe it possible to eliminate the occurrence of large scale instrument failures, etc. resulting from ICS/NNI failures, prudence dictates that newly-automated functions be subject to thorough failure modes and effects, common-cause failure, and interactions analyses.

12. <u>Instrumentation and Control Technicians Be Assigned to all Shifts</u>

This recommendation addresses the Task Force concern that power faults, etc. which result in severe ICS/NNI failures can be sufficiently complex that trained instrumentation and control personnel are required to study and correct the problem. Since it is not now the practice of all plants to have such personnel on all shifts, there exists the potential for extended fault rectification times if staff must be brought in from offsite in an emergency. Because of this concern, the Task Force recommended that appropriate personnel be available on-site during all shifts.

We believe that this recommendation has both positive and negative aspects. On the positive side, we agree with the Task Force that having trained personnel available would be somewhat beneficial - probably of moderate value for incidents and accidents, and low value for severe accidents. However, consideration of the data on the causes of large scale ICS/NNI failures indicates that roughly one-half of the events were a result of errors made by these same personnel as they performed their surveillance and maintenance duties. Since presumably these personnel would be performing their routine duties during their shifts, the likelihood of experiencing an ICS/NNI failure on back shifts would be increased somewhat by requiring the appropriate personnel to be present. On balance, we believe that the positive aspects of this recommendation slightly outweigh the negative aspects; however, we also believe that the "net gain" is of low value. Recommendation 14 is more to the point.

13. Operator Training on the Crystal River Incident

14. Development of Plant-Specific Procedures for Loss of ICS/NNI

We have chosen to consolidate Task Force recommendations 13 and 14 into one for the purposes of this risk evaluation because of their similarity in intent. Recommendation 13 of the Task Force calls for specific operator training on the events of the February 26, 1980 incident at Crystal River. Recommendation 14 addresses the need for plant-specific procedures to assist operating crews when ICS/NNI failures occur in the future.

We believe that the reduction in the likelihood of operator errors during ICS/NNI-caused transients requires operator training involving both retrospective and forward-thinking views. The Task Force's recommendation on Crystal River training provides one aspect of the retrospective training; however, this specific training alone does pose questions regarding the need for training on other similar events, e.g., the Rancho Seco "light bulb" incident or others identified from LERs as having the potential to be accident precursors. We believe that this type of training could be highly valuable in "preparing" the operators for possible future accidents.

The Task Force recommendation on plant-specific procedure development addresses the need for forward-thinking training. Since it is a virtual certainty that operators will be faced with ICS/NNI failures in the future (which may be similar to or different from past events), we believe it important that more general training on coping with such events be provided.

We believe that this combination of training for ICS/NNI faults can be of relatively high effectiveness for this type of transient. Other recommendations reduce the significance of these incidents, e.g., recommendations 2 and 6. We believe that on an overall basis, these recommendations are of high value for incidents, high value for accidents, and moderate value for severe accidents.

15. Increased Simulator Training

This Task Force recommendation calls for the requirement of a one week per year simulator training course for all operators in B&W plants (this training is now optional).

We believe that this recommendation has both positive and negative aspects. On the positive side, such simulator training can be important to the understanding of plant behavior during transient events, LOCAs, etc., and thus be a useful means to reduce the likelihood of operator error during real events (e.g., Crystal River type "incidents" and TMI-2 type "accidents"). We believe that making such training mandatory, rather than optional, is of moderate value for incidents, moderate value for accidents, and negligible value for severe accidents.

The negative aspects of this recommendation result from our concern about the limitations of the available simulator capability. First, the B&W simulator is made to resemble the Rancho Seco control panels, so that operators from other plants may have difficulty in fully melding together their training with their own control room. Second, present simulators tend to have difficulty in accurately recreating some transient events, so that the training can again be somewhat counterproductive. Overall, however, we believe that these negative aspects do not overshadow the gains achievable by the simulator training, so that we agree that this training should be pursued.

16. Criteria for Restarting Reactor Coolant Pumps

This Task Force recommendation is concerned with guidelines provided to the operators of B&W plants with respect to the restart of the reactor coolant pumps during non-LOCA transients. B&W has provided these guidelines to the operators; however, the NRC staff has yet to conduct their review. The recommendation calls for the expeditious completion of the NRC review.

We believe that appropriate guidance on the restart of the reactor coolant pumps can be an important aspect in the prevention of core overheating and damage. Forced-flow cooling of the fuel can be highly advantageous during events where malfunctions have interrupted decay heat dissipation, so that clear criteria for re-establishing this flow appears to be of significant merit. Because of the potential merit of quickly re-establishing reactor coolant pump flow, we believe that the completion of the NRC's review of the restart guidelines is of moderate value for improving the capability of the plant to cope with incidents and accidents, and low value for severe accidents.

17. Alternative Solution to PORV Unreliability and Safety System Challenge Rate Concerns

This Task Force recommendation addresses the concern that, since the post-TMI switch of the PORV setpoint and the reactor trip setpoint on high RCS pressure (and other related plant modifications), the frequency of reactor trips in B&W plants has increased. It appears that transients which formerly would have been accommodated without causing a reactor trip now do result in trip. Since this increased trip frequency has some negative impact on plant safety (e.g., increased likelihood of an ATWS event), the Task Force has recommended that a proposed plant modification plan (submitted by Consumer's Power Company) which would allow a return to the pre-TMI setpoints be considered by the NRC staff. If determined to be acceptable by the staff, the Task Force recommends that such modifications be required in all B&W plants.

It is apparent that the return to the pre-TMI PORV/reactor trip setpoints has both positive and negative aspects. On the positive side, the return to the original setpoints could reduce the likelihood of ATWS events to some limited extent, and allow the plants to operate in a way more like that to which they had been originally designed. The latter aspect may help somewhat to minimize unusual behavior of the plants during transients (i.e., it allows them to respond more smoothly during such events).

On the negative side, the return to the original setpoints will increase the frequency of use of the PORV; with this increased frequency the likelihood of experiencing a stuck-open valve (a small LOCA) increases commensurately. While the installation of an automatically-closing PORV block valve may alleviate this aspect, it also presents other problems. In some accidents (e.g., a total loss of feedwater), the PORV is the only controllable means for energy removal from the RCS. In such instances, an open PORV can be advantageous, in that it permits RCS depressurization with the associated increased HPI flow. Further, for plants with relatively low-head HPI pumps (e.g., Davis-Besse), a stuck-open (or commanded open) PORV is the only means for the critical RCS depressurization. In such situations, automatic block valve closure can be distinctly counterproductive. Also, the automatic closure of the PORV block valve could, for events such as a total loss of feedwater or the Crystal River incident, result in unnecessary challenges to the (unisolable) safety valves. Thus, block valve auto-closure can increase the challenge rate of the safety valves, resulting in an increased likelihood of a <u>bona fide</u> LOCA. It is noteworthy that during the February 26, 1980 Crystal River incident, operator actions to close the PORV block valve (as required by NRC) resulted in the opening of the safety valves, with the resulting increase in coolant release to the reactor building.

The return to the original setpoints appears to have merit. Improved PORV block valve reliability is also clearly desirable. However, the automatic closure of the block valve(s) appears to have undesirable side effects. While not as critical as some other Task Force recommendations, we nonetheless believe that the resolution of this issue is still important. We believe that this recommendation is of moderate value for the incident category, low value for the accident category, and of negligible value for the category of severe accidents.

18. Completion of the IREP Crystal River Study

This Task Force recommendation relates to the Probabilistic Analysis Staff's risk evaluation of the Crystal River plant, which is the first part of the overall IREP study of all operating plants. This study has as its goal the identification of those factors of the plant design which are important to the public risk from that particular plant. The recommendation calls for the expeditious completion of the Crystal River study, with prompt consideration made by the NRC on the need for plant modifications suggested by the study.

The IREP Crystal River study has as a goal the identification of those plant faults which have the greatest potential for causing core damage and risk to the public for events initiated by transients and LOCAs. For this reason, we believe that such an identification can have high value for accident sequences resulting from "routine" losses of feedwater, station blackout, and small LOCAs. Since other initiating events have not been as thoroughly evaluated (e.g., losses of ICS/NNI, etc.), the potential frequency reduction potential for such sequences is less significant. Since the results of the study (and the subsequent regulatory actions) are not yet completely clear, we cannot now determine the importance of the study results on plant safety.

19. <u>Performance Criteria for Anticipated Transients</u>

This Task Force recommendation calls for the development of performance criteria to define the acceptable limits of plant response to anticipated transients. The purpose of the criteria is to assure that those plant functions critical to coping with transient events are designed to adequately protect the core during such events.

Without knowing what factors will be considered in the development of these performance criteria, we find it difficult to assess the relative merit of this recommendation in relation to others made by the Task Force. Development of criteria for system performance, such as reliability, human and systems interactions potential, etc. could provide significant payoff; for this reason, we agree that this relatively long-term Task Force recommendation should be pursued.

20. <u>Criteria for Reactor Coolant Pump Trip in Small LOCAs</u>

In the post-TMI reconsideration of small pipe break accidents, a concern arose that for certain sizes of pipe breaks, the running of the reactor coolant pumps might aggrevate the break flow to the extent that licensing requirements on acceptable accident fuel temperatures would be exceeded. As a result of this concern, the NRC now requires that the reactor coolant pumps be tripped under certain conditions when it is believed that a small LOCA exists. The NRC staff has acknowledged that such a requirement may impede the capability for recovery from other types of events and as such has recommended that the question of the relative merit of pump trip continue to be pursued. This Task Force recommendation endorses the previous staff and industry recommendations on this matter.

We agree that the present requirements for pump trip are less than ideal. While for some small LOCAs it may be preferable to trip the reactor coolant pumps, clear benefit in continued pump operation may be seen for other sizes of LOCAs and for non-LOCA transients which have some symptoms similar to those of LOCAs. We believe that this concern is of moderate value in the capability of the plant to cope with incidents and accidents, and of negligible value for severe accidents.

21. <u>Reevaluation of the AFWS Injection Point into the Steam Generators</u>
In general, B&W plants inject the AFWS water into the steam generators through a feedwater ring at the top of the steam generators, so that the water sprays directly onto the steam generator tubes. In contrast, Westinghouse and Combustion Engineering plants are designed

such that AFWS flow enters through the main feedwater rings, filling the steam generator from the bottom. Because of top-entry of AFWS water increases the potential for an RCS overcooling transient, the Task Force has recommended that reconsideration be given to the relative desirability of top-entry and bottom-entry of AFWS water.

We believe that both points for AFWS entry have positive and negative aspects. Top-entry has the advantage of providing a higher effective thermal center in the steam generator, so that natural circulation cooling would be enhanced. Prospects of recovering from situations entailing degraded core cooling are better with top-entry injection. It is thus important to safety not to lose this option. As noted above, this entry point does, however, have the disadvantage of increasing the likelihood of overcooling the RCS. Bottom-entry does reduce the overcooling potential, but also lowers the steam generator's thermal center. The latter entry point may also pose problems of thermal shock of the feedwater lines, nozzles, etc. We strongly recommend against eliminating the top-entry injection option. Further, the added complexity of top and bottom injection point options is probably not warranted by the small risk reduction potential in reducing overcooling events. In our judgment, we believe this recommendation to be of low value in the reduction of incident frequency, and negligible importance to the categories of accidents and severe accidents.

22. Study of Operator Errors in B&W Plants

In reviewing the operating experience of B&W plants for instances of ICS/NNI failures, it became apparent to members of the Task Force that the frequency of operator errors in these plants tended to be somewhat higher than that for other plants. This Task Force recommendation calls for an evaluation of the compiled data to assess the statistical significance of this apparent difference.

The Probabilistic Analysis Staff has determined that the differences in operator error rates in Table 5.3 of this report are not statistically significant. However, PAS has under contract a research program to study the kinds and frequencies of operator errors being reported in LERs, to relate these to plant, vendor, and circumstance. These studies may lead to insights that can be used to reduce human error contributions to the risk.

8. GENERIC IMPLEMENTATION GUIDELINES

Because of the generic nature of this report, it is not possible for this Task Force to recommend a detailed implementation schedule, since the specific actions necessary to comply with the recommendations will vary with the particular design of each plant. Therefore, Table 8.1 provides generic implementation guidelines. These guidelines address each of the 22 Task Force recommendations and assign each to a priority group and an action group.

Priorities 1 or 2 are assigned under the priority grouping. Priority 1 implies that the Task Force believes that these items should be scheduled and implementation begun as soon as possible. To accomplish this may require rescheduling of NRC staff and licensee/industry priorities and resources. Priority 2 items, on the other hand, should be scheduled for implementation consistent with existing priorities and resources. Priority 1 and 2 recommendations were assigned by the Task Force based on consideration of the following:

(1) The Probabilistic Analysis Staff's evaluation of the effectiveness of the Task Force recommendations based upon qualitative perspectives derived from probabilistic safety analysis and risk assessment (this evaluation is presented in Section 7 of the report);

(2) The Decision and Priority Group assignments of existing or closely coupled requirements contained in the TMI-2 Action Plan; and

(3) Comments received since the issuance of the draft report on April 2, 1980 from members of the NRC staff, Babcock & Wilcox, the B&W licensees, NSAC, and both the ACRS B&W Reactor Subcommittee and the ACRS full committee.

The action group is divided into four parts: (1) "A" recommendation is closely coupled with an existing requirement contained in the TMI-2 Action Plan; (2) "B" recommendation is not closely coupled with an existing requirement contained in the TMI-2 action plan and licensee/industry action is required. (3) "C" recommendation is not closely coupled with an existing requirement in the TMI-2 Action Plan and NRC staff action is required; and (4) "D" recommendation is not closely coupled with an existing requirement in the TMI-2 Action Plan and Joint NRC staff and licensee/industry action is required.

Based on discussions held during meetings between the Task Force and the B&W licensees, it is anticipated that licensees may propose alternative methods or solutions to meeting certain goals of specific Task Force recommendations. Prior to directing implementation of the recommendations, alternative solutions should be given consideration by the staff. In addition, the Task Force recognizes that licensees may propose to combine and implement certain of the Task Force recommendations in such a manner that the need to implement other Task Force recommendations may become unnecessary.

Therefore, the Task Force recommends that implementation should be accomplished by integrating the generic implementation guidelines, proposed by the Task Force, with the existing or related requirements specified in the TMI-2 Action Plan, and plant-specific comments or proposals advanced by the B&W licensees.

TABLE 8.1

GENERIC IMPLEMENTATION GUIDELINES

Key to terms in table:

Priority Group

Priority 1 - Items should be scheduled and implementation begun as soon as possible. These items may require rescheduling of NRC staff and licensee/industry priorities and resources.

Priority 2 - Items should be scheduled and implemented in accordance with existing priorities and resources.

Action Group

A - Recommendation closely coupled with existing requirement contained in the TMI-2 Action Plan.

B - Recommendation not closely coupled with existing requirements contained in the TMI-2 Action Plan, licensee/industry action is required.

C - Recommendation not closely coupled with existing requirement contained in the TMI-2 Action Plan, NRC staff action is required.

D - Recommendation not closely coupled with existing requirement contained in the TMI-2 Action Plan, joint NRC staff and licensee/industry action is required.

TABLE 8.1

GENERIC IMPLEMENTATION GUIDELINES

No.	Requirement	Priority 1	Priority 2	Action Group A	Action Group B	Action Group C	Action Group D	Similar Requirements Which Should Be Considered
1.	AFW system upgrade to ESF system	X		X				II.E.1.1 of TMI-2 Action Plan
2.	ESF automatic initiation and control of AFW	X		X				II.E.1.2 of TMI-2 Action Plan
3.	Addition of diverse-drive AFW pump for DB-1	X		X				II.E.1.1 of TMI-2 Action Plan
4.	Modifications to steam line break detection and mitigation system	X		X				None
5.	Improvements in plant control systems (NNI/ICS)		X				X	BAW-1564, NSAC-3/INPO-1, IE Bulletin 79-27
6.	Selected data set of principal plant parameters	X		X				I.D.2 of TMI-2 Action Plan
7.	Increased usage of in-core thermocouples		X		X			None
8.	High radiation signal for vent/purge isolation		X	X				II.E.4.2 of TMI-2 Action Plan
9.	System response to maintain pressurizer level on scale and pressure above HPI setpoint		X	X				II [illegible] of TMI-2 Action Plan
10.	Sensitivity studies of operational modifications		X	X				II.E.5 of TMI-2 Action Plan
11.	Modifications to eliminate immediate manual actions		X		X			None
12.	Qualified I&C technican on duty	X			X			NUREG-0654/FEMA-REP-1 (Section II.B)
13.	Operator Training on CR-3 event	X			X			Partially covered by Confirmatory Orders issued to B&W operating plants following CR-3 event
14.	Emergency procedures for loss of NNI/ICS	X			X			Same as number 13 above
15.	Mandatory simulator training for requalification		X					H.R. Denton to ALL POWER REACTOR APPLICANTS AND LICENSEES, dated 03/28/80 (Criteria D.4)

TABLE 8.1 (continued)

No.	Requirement	Priority 1	Priority 2	Action Group A	Action Group B	Action Group C	Action Group D	Similiar Requirements Which Should Be Considered
16.	Evaluation of RCP restart criteria		X	X				II.K.1 Table C.1 Item 27 of TMI-2 Action Plan
17.	Alternative solution to PORV unreliability/ safety system challenge rate concerns		X			X		None
18.	Completion of IREP Crystal River 3 study		X	X				II.C.1 of TMI-2 Action Plan
19.	Development of Performance Criteria for anticipated transients	X					X	None
20.	Continued evaluation of need to trip RCPs during small break LOCAs	X		X				II.K.3 Table C.3 Item 5 of TMI-2 Action Plan
21.	Reevaluate location of AFW injection into OTSG		X		X			None
22.	Staff study of personnel related LERs with respect to higher number for B&W plants		X	X				I.E.8 of TMI-2 Action Plan

9. REFERENCES

1. U.S. Nuclear Regulatory Commission, "NRR Status Report on Feedwater Transients in B&W Plants," April 25, 1979.

2. Confirmatory Shutdown Orders for B&W Operating Plants:

   a. Commission Order issued to Duke Power Company (Oconee 1, 2 and 3), May 7, 1979. Docket Nos. 50-269, 50-270, 50-287.

   b. Commission Order issued to Sacramento Municipal Utility District (Rancho Seco), May 7, 1979. Docket No. 50-312.

   c. Commission Order issued to Florida Power Corporation (Crystal River 3), May 16, 1979. Docket No. 50-302.

   d. Commission Order issued to Toledo Edison Company (Davis-Besse 1), May 16, 1979. Docket No. 50-346.

   e. Commission Order issued to Arkansas Power & Light Company (Arkansas Nuclear One, Unit 1), May 17, 1979. Docket No. 50-313.

Available in NRC PDR for inspection and copying for a fee.

*The indicated reports are available for purchase from the NRC/GPO Sales Program, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and the National Technical Information Service, Spingfield, Virginia 22161.

3. U.S. Nuclear Regulatory Commission, "Staff Report on the Generic Assessment of Feedwater Transients in Pressurized Water Reactors Designed by the Babcock & Wilcox Company," USNRC Report NUREG-0560, May 1979.*

4. U.S. Nuclear Regulatory Commission, "Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox Designed 177-FA Operating Plants," USNRC Report NUREG-0565, January 1980.*

5. U.S. Nuclear Regulatory Commission, "Report of the Bulletins and Orders Task Force," USNRC Report NUREG-0645, Vols. I and II, January 1980.*

6. Memorandum from H. R. Denton, NRC, to Chairman Hendrie, Commissioners Gilinsky, Kennedy, Bradford, and Ahearne, "Interim Report on Sensitivity Studies of the B&W Reactor Design," October 25, 1979. Available in NRC PDR for inspection and copying for a fee.

7. U.S. Nuclear Regulatory Commission, "NRC Action Plans Developed as a Result of the TMI-2 Accident," USNRC Draft Report NUREG-0660 (Draft 3), March 5, 1980.*

8. U.S. Nuclear Regulatory Commission, Office of Inspection and Enforcement Bulletin No. 79-27, "Loss of Non-Class 1E Instrumentation and Control Power System Bus During Operation," November 30, 1979. Available in NRC PDR for inspection and copying for a fee.

9. Letter from E. L. Zebroski, Nuclear Safety Analysis Center, and E. P. Wilkinson, Institute of Nuclear Power Operations, to H. R. Denton, NRC, Subject: Report on Crystal River Unit-3 Incident of February 26, 1980, by NSAC and INPO, dated March 11, 1980, Report NSAC-3/INPO-1. Docket No. 50-302. Available in NRC PDR for inspection and copying for a fee.

10. Letter from H. R. Denton, NRC, TO ALL OPERATING B&W REACTOR LICENSEES, Subject: Request for Additional Information, Under 10 CFR 50.54, Regarding the Crystal River 3 Event of February 26, 1980, dated March 6, 1980. Docket No. 50-302. Available in NRC PDR for inspection and copying for a fee.

11. Letter from J. A. Hancock, Florida Power Corporation, to H. R. Denton, NRC, Subject: Response to NRC Letter of March 6, 1980, dated March 12, 1980. Available in NRC PDR for inspection and copying for a fee.

12. U.S. Nuclear Regulatory Commission, "Generic Assessment of Delayed Reactor Coolant Pump Trip during Small Break Loss-of-Coolant Accidents in Pressurized Water Reactors," USNRC Report NUREG-0623, November 1979.*

13. U.S. Nuclear Regulatory Commission, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," USNRC Report NUREG-0578, July 1979.*

14. U.S. Nuclear Regulatory Commission, "TMI-2 Lessons Learned Task Force Final Report," USNRC Report NUREG-0585, October 1979.*

15. U.S. Nuclear Regulatory Commission, "Operating Units Status Report," USNRC Report NUREG-0020, Printed Monthly.*

16. Letter from J. G. Herbein, Metropolitan Edison Company, to H. R. Denton, NRC, Subject: Response to NRC Letter of March 6, 1980, dated March 13, 1980. Available in NRC PDR for inspection and copying for a fee.

17. Letter from R. P. Crouse, Toledo Edison, to R. W. Ried, NRC, Subject: Response to NRC Letter of March 6, 1980, dated March 12, 1980. Available in NRC PDR for inspection and copying for a fee.

18. Letter from C. L. Steel, Arkansas Power & Light Company, to H. R. Denton, NRC, Subject: Response to NRC Letter of March 6, 1980, dated March 12, 1980. Available in NRC PDR for inspection and copying for a fee.

19. Letter from J. G. Herbein, Metropolitan Edison Company, to H. R. Denton, NRC, Subject: Supplemental Response to NRC Letter of March 6, 1980, dated March 17, 1980. Available in NRC PDR for inspection and copying for a fee.

20. Letter from W. O. Parker, Jr., Duke Power Company, to H. R. Denton, NRC, Subject: Response to NRC Letter of March 6, 1980, dated March 12, 1980. Available in NRC PDR for inspection and copying for a fee.

21. Same as Reference number 11.

22. Letter from G. C. Moore, Florida Power Corporation, to R. W. Reid, NRC, Subject: PORV, Safety Valve Actuation Data and Reactor Trip Frequency, dated November 14, 1979. Available in NRC PDR for inspection and copying for a fee.

23. Letter from J. H. Taylor, Babcock & Wilcox, to D. G. Eisenhut, NRC, Subject: Response to Oral Request Made at a Meeting on March 4, 1980, Between the NRC, B&W and Operators of B&W Plants Regarding Crystal River 3 Event of February 26, 1980, dated March 12, 1980. Docket No. 50-302. Available in NRC PDR for inspection and copying for a fee.

24. Letter from J. H. Taylor, Babcock & Wilcox, to R. J. Mattson, NRC, Subject: Forwards B&W Report Entitled "Evaluation of Transient Behavior and Small Break Loss-of-Coolant System Breaks in the 177-FA Plant (Volumes I and II), dated May 7, 1980. Available in NRC PDR for inspection and copying for a fee.

25. Westinghouse Electric Corporation, "Report on Small Break Accidents for Westinghouse NSSS System," WCAP-9600 Vols. I to III, June 1979. Available in NRC PDR for inspection and copying for a fee.

26. Combustion Engineering Incorporated, "Review on Small Break Transients in Combustion Engineering Nuclear Steam Supply Systems," CEN-114, July 1979. Available in NRC PDR for inspection and copying for a fee.

27. U.S. Nuclear Regulatory Commission, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants," USNRC Report NUREG-0611, January 1980.*

28. Letter from M. S. Plesset, Advisory Committee on Reactor Safeguards, to J. F. Ahearne, NRC, Subject: Recommendations of the NRC Task Force on Bulletins and Orders, dated March 11, 1980. Available in NRC PDR for inspection and copying for a fee.

29. U.S. Nuclear Regulatory Commission, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accident in Combustion Engineering Designed Operating Plants," USNRC Report NUREG-0635, January 1980.*

30. "The Institute of Electrical and Electronics Engineers, Incoporated Standard: Criteria for Protection Systems for Nuclear Power Generating Stations," IEEE Standard 279-1971. Available from The Institute of Electrical and Electronics Engineers, Inc., 345 East 47 Street, New York, New York 10017.

31. U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants - LWR Edition," USNRC Report NUREG-75/087, Section 10.4.9, "Auxiliary Feedwater System (PWR)," Attached Branch Technical Position ASB 10-1, "Design Guidelines for Auxiliary Feedwater System Pump Drive and Power Supply Diversity for Pressurized Water Reactor Plants," Rev. 1.*

32. Letter from R. W. Reid, NRC, to J. J. Mattimoe, Sacramento Municipal Utility District, Subject: NRC Requirements for the Continued Upgrade of the Rancho Seco Auxiliary Feedwater (AFW) System, dated February 26, 1980. Docket No. 50-312. Available in NRC PDR for inspection and copying for a fee.

33. U.S. Nuclear Regulatory Commission, "Evaluation of Licensee's Compliance With the NRC Order Dated May 16, 1979-Toledo Edison Company and The Cleveland Electric Illuminating Company-Davis-Besse Nuclear Power Station, Unit No. 1-Docket No. 50-346," dated July 6, 1979. NOTE: subject document may be found as Enclosure 1 to Letter from H. R. Denton, NRC, to L. E. Roe, Toledo Edison Company, Subject: Authorization to Resume Power Operations, dated July 6, 1979. Available in NRC PDR for inspection and copying for a fee.

34. U.S. Nuclear Regulatory Commission," Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," USNRC Draft Regulatory Guide 1 97, Rev. 2, issued for public comments on December 4, 1979. Available from USNRC Division of Technical Information and Document Control, Washington, DC 20555.

35. Letter from J. H. Taylor, Babcock & Wilcox, to D. F. Ross, Jr., NRC, Subject: Forwards report entitled "Integrated Control Systems Reliability Analysis, BAW-1564," dated August 17, 1979. Available in NRC PDR for inspection and copying for a fee.

36. Letter from J. H. Taylor, Babcock & Wilcox, to D. F. Ross, Jr., NRC, Subject: Comparison of Reactor Trip History between B&W, CE, and Westinghouse Operating Plants, dated October 18, 1979. Available in NRC PDR for inspection and copying for a fee.

37. R. A. Capra, NRC, "Summary of Meeting Held on August 23, 1979, With the Babcock & Wilcox (B&W) Operating Plant Licensees to Discuss Recent (Post TMI-2) Feedwater Transients," dated September 13, 1979. Available in NRC PDR for inspection and copying for a fee.

38. Letter from R. W. Reid, NRC, TO ALL B&W OPERATING PLANT LICENSEES, Subject: Request for Additional Information Concerning B&W Recommendations Contained in ICS Reliability Analysis-BAW-1564, dated November 7, 1979. Available in NRC PDR for inspection and copying for a fee.

39. "American National Standard Requirements for Selection and Training of Nuclear Power Plant Personnel," ANSI N18.1-1971. Available from the American Nuclear Society, 244 East Ogden Avenue, Hinsdale, Illinois 60521.

40. U.S. Nuclear Regulatory Commission, "Quality Assurance Program Requirements (Operation)," USNRC Regulatory Guide 1.33, Rev. 2, February 1978. Available from USNRC Division of Technical Information and Document Control, Washington, DC 20555.

41. R. W. Reid, NRC, "Summary of March 4, 1980 Meeting Regarding Crystal River 3 Incident of February 26, 1980," dated March 13, 1980. Docket No. 50-302. Available in NRC PDR for inspection and copying for a fee.

42. U.S. Nuclear Regulatory Commission, "Reactor Safety Study - An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," USNRC Report NUREG-75/014 (WASH-1400), October 1975. Available free upon written request from USNRC Division of Technical Information and Document Control, Washington, DC 20555.

43. "Summary of Meeting January 29, 1980 on Selected Responses on B&W Sensitivity, Midland Plants Units 1 and 2" dated February 14, 1980. Docket Nos. 50-329, 50-330. Available in NRC PDR for inspection and copying for a fee.

APPENDIX A

POST TMI-2 ACCIDENT REQUIREMENT

FOR B&W OPERATING PLANTS

APPENDIX A

POST TMI-2 ACCIDENT REQUIREMENTS FOR B&W OPERATING PLANTS

CATEGORY SOURCE: IE BULLETINS (TMI-2 RELATED)

Number	Requirement	Source	Implementation	Action Plan	Remarks
1.	Review TMI-2 Preliminary Notifications (PNs) and detailed chronology of TMI-2 accident.	79-05 (Item 1) 79-05A (Item 1)	COMPLETE	I.A.2.2 I.A.3.1 II.K.1	
2.	Review transients similar to TMI-2 that have occurred at B&W facilities and review the evaluation of the 11/29/77 transient at Davis-Besse 1.	79-05 (Item 2) 79-05A (Item 2)	COMPLETE	I.A.2.2 I.A.3.1 II.K.1	
3.	Review operating procedures for recognizing, preventing, and mitigating void formation during transients and accidents.	79-05 (Item 3) 79-05A (Item 3)	COMPLETE	I.C.1 II.K.1	
4.	Review operating procedures and training instructions to ensure that:				
	a. Operators do not override ESF actions unless continued operation will result in unsafe plant conditions.	79-05 (Item 4) 79-05A (Item 4a) 79-05B (Item 2)	COMPLETE	I.C.1 I.C.7 I.C.8 I.G.8 II.K.1	
	b. HPI system remains in operation (if actuated automatically) unless: (1) Both LPI pumps are operating at a flow rate greater than 1000 gpm and the situation has been stable for 20 minutes, or (2) HPI has been in operation for 20 mins. and the RCS is at least 50°F subcooled.	79-05A (Item 4b) 79-05B (Item 2)	COMPLETE	I.C.1 II.K.1	NOTE: There is no longer any requirements to maintain HPI in operation for 20 mins. HPI may be terminated upon reaching 50°F subcooling regardless of the time the system has been operating.

CATEGORY SOURCE: IE BULLETINS (TMI-2 RELATED) (continued)

Number	Requirement	Source	Implementation	Action Plan	Remarks
c.	Until automatic RCP trip is installed and operational: (1) Upon reactor trip and HPI initiation caused by low RCS pressure, trip all operating RCPs, and (2) Provide two operators in the CR at all times to accomplish RCP trip and other required items.	79-05A (Item 4c) 79-05C (Short-term Item 1a and 1b)	COMPLETE	I.C.1 I.A.1.3	79-05C superseded the requirements of 79-05A Item 4c) Evaluation is contained in NUREG-0623)
d.	Operators are provided with additional information and guidance not to rely on pressurizer level indication alone in evaluating plant conditions.	79-05A (Item 4d)	COMPLETE	I.C.1 I.A.3.1 II.F.2 II.K.1	
5.	Review all safety related valve positions and positioning requirements and positive controls and all related test and maintenance procedures to assure proper ESF functioning, if required. Verify all AFW valves are in the open position.	79-05 (Item 5) 79-05A (Item 5)	COMPLETE	I.C.2 I.C.6 II.K.1	
6.	Review operating modes and procedures for all systems designed to transfer potentially radioactive gases and liquids out of containment to assure that undesired pumping of radioactive gases or liquids will not occur inadvertently. Ensure that this does not happen on ESF reset. List all such systems and list: a. whether interlocks exist to prevent transfer on high radiation, and b. whether such systems are isolated by containment isolation signals.	79-05 (Item 6) 79-05A (Item 9)	03/31/80	II.E.4.2 II.K.1	

CATEGORY SOURCE: IE BULLETINS (TMI-2 RELATED) (continued)

Number	Requirement	Source	Implementation	Action Plan	Remarks
7.	Review containment isolation design and procedures and make necessary changes to assure that all lines whose isolation does not degrade core cooling capability will isolate upon initiation of safety injection (HPI).	79-05 (Item 7)	03/31/80	II.E.4.2 II.K.1	
8.	Review prompt reporting procedures to assure that the NRC is notified within one hour from the time the reactor is not in a controlled or expected condition of operation. Maintain continuous communication channel.	79-05 (Item 7) 79-05A (Item 12) 79-05B (Item 6)	COMPLETE	I.E.6 III.A.3.3 II.K.1	
9.	Implement positive position controls on manual valves and manually-operated, motor-driven valves that could compromise or defeat AFW flow.	79-05A (Item 7)	COMPLETE	II.E.1.1 II.K.1	
10.	Prepare and implement procedures which assure at least two 100% capacity AFW flow paths are available whenever the primary heat removal source is through the OTSG. Implement the following LCOs: If two paths are not available, shutdown within 72 hours and be cooled down within 12 hours. If at least one path is not available, be subcritical within one hour and be cooled down within 12 hours (or the maximum safe rate).	79-05A (Item 8)	03/31/80	II.E.1.1 II.K.1	
11.	Review and modify procedures for removing safety-related systems from service (and restoring to service) for maintenance and testing to assure operability status is verified and known.	79-05A (Item 10)	03/31/80	I.C.2 I.C.6 II.K.1	

CATEGORY SOURCE: IE BULLETINS (TMI-2 RELATED) (continued)

Number	Requirement	Source	Implementation	Action Plan	Remarks
12.	Assure all operating and maintenance personnel are aware of the seriousness and consequences of the simultaneous blocking of both AFW trains and the other actions taken during the early phases of the TMI-2 accident.	79-05A (Item 11)	COMPLETE	I.A.3.1 I.A.2.2 II.K.1	
13.	Develop procedures and train operators on methods of establishing natural circulation. Include: means of monitoring efficiency by available instrumentation; assure RCS is at least 50°F subcooled; precautions for pressurizer level indication, pressure control, P-T limits; and procedures in the event of LOFW while in natural circulation.	79-05B (Item 11)	COMPLETE	I.C.1 I.G.1 II.K.1	
14.	Modify design and procedures which reduce the likelihood of automatic PORV lifting during anticipated transients. Lower high pressure reactor trip setpoint.	79-05B (Item 3)	COMPLETE	II.E.5 II.K.1	PORV setpoint changed from 2255 to 2450 psig (DB-1 setpoint 2400) High pressure trip setpoint changed from 2355 to 2300 psig.
15.	Provide a manual trip (procedures and training) for the following high pressure transients: a. Loss of Main Feedwater, b. Turbine Trip, c. Main Steam Isolation Valve Closure, d. Loss of Offsite Power, e. Low Steam Generator Level, and f. Low Pressurizer Level	79-05B (Item 4)	COMPLETE	II.K.1	

CATEGORY SOURCE: IE BULLETINS (TMI-2 RELATED) (continued)

Number	Requirement	Source	Implementation	Action Plan	Remarks
16.	Provide design review and schedule for implementation of a safety-grade reactor trip upon: a. Loss of Feedwater, b. Turbine Trip, c. Significant Reduction in Steam Generator Level	79-05B (Item 5)	COMPLETE	II.K.1	A preliminary design approval was given to B&W licensees on December 20, 1979 for LOFW and TT. Low OTSG level was not included.
17.	Propose changes to Technical Specifications which must be modified as a result of implementing IE Bulletin items.	79-05B (Item 7)	03/31/80	II.K.1	
18.	Perform and submit a report of LOCA analysis for a range of small break sizes and a range of time lapses between reactor trip and RCP trip. Determine PCT and identify any area where PCT is greater than 2200°F.	79-05C (Short-Term Item 3)	COMPLETE	I.C.1	Evaluation of the LOCA analysis is documented in NUREG-0623.
19.	Based upon the analyses done in requirement 18 above, develop new guidelines for operator action for both LOCA and non-LOCA events that take into account the effect of RCP trip.	79-05C (Short-Term Item 3)	COMPLETE	I.C.1	Guidelines developed for this requirement subsequently modified under req. 21.
20.	Based upon guidelines developed in requirement 19 above, revise emergency procedures and train all operators.	79-05C (Short-Term Item 4)	COMPLETE	I.C.1 I.A.3.1 I.G.1	
21.	Provide analyses and develop guidelines and procedures for inadequate core cooling. Define RCP restart criteria.	79-05C (Short-Term Item 5)	COMPLETE	I.C.1 II.F.2	Latest version of B&W SBLOCA Guidelines dated 11/79.

CATEGORY SOURCE: IE BULLETINS (TMI-2 RELATED) (continued)

Number	Requirement	Source	Implementation	Action Plan	Remarks
22.	Propose and submit a design whic assure automatic tripping of the h under all circumstances in which this action may be required.	79-05C (Long-Term Item 1)	01/01/81	Table C.3 Item 5	Implementation date means installed and operational. Preliminary design approval given to B&W licensees in letters dated 12/17&18/79.

NOTES ON IE BULLETINS (TMI-2 RELATED):

1. IE Bulletins Applicable to B&W Operating Plants:

Bulletin	Issued	Comments
79-05	04/01/79	Superseded by 79-05A
79-05A	04/05/79	Supersedes 79-05
79-05B	04/21/79	Modifies some parts of 79-05A
79-05C	07/26/79	Supersedes Item 4c of 79-05A (RCP operation during a LOCA)

2. Present HPI termination criteria for B&W plants (see requirements #4b):

"If the HPI system has been actuated because of low pressure conditions, it must remain in operation until one of the following criteria is satisfied:

a. The LPI system is in operation and flowing at a rate in excess of 1000 GPM in each line and the situation has been stable for 20 minutes.

OR

b. All hot and cold leg temperatures are at least 50°F below the saturation temperature for the existing RCS pressure and the action is necessary to prevent the indicated pressurizer level from going off-scale high.

3. One of the requirements of Item 5 of IE Bulletin 79-05B required design submission for a safety-grade reactor trip upon significant reduction in OTSG level (see requirement #16). B&W licensees do not agree that low S/G level is anticipatory and have not included this requirement in their safety-grade design.

4. ng-term Item 1 of IE Bulletin 79-05C (see requirement #22) requests design of automatic RCP trip be submitted. Preliminary design approval was sent to the B&W licensees on 12/17-18/79 for use of coincident signals of low pressure ESFAS and low RCP current/power. Licensees now report that they may not be able to provide information requested by staff with regard to verifying correlation between RCP current/power and voiding in RCS. Licensees are investigating other signals which could be used.

CATEGORY SOURCE: COMMISSION ORDERS OF MAY 1979 (Short-Term Requirements)

Number	Requirement	Source	Implementation	Action Plan	Remarks
1.	Upgrade the timeliness and reliability of delivery of the AFW system.	All Orders	COMPLETE	II.E.1 II.K.2	Specific actions under this item varied from plant to plant. See pages A-10 and A-11 for a complete listing of actions.
2.	Procedures and training to initiate and control AFW independent of the integrated control system (ICS).	All Orders	COMPLETE	II.E.1.2 II.K.2	
3.	Install hard-wired, control-grade anticipatory reactor trip for loss of feedwater and turbine trip.	All Orders	COMPLETE	II.K.2	
4.	Perform small break LOCA analysis, procedures, and operator training.	All Orders	COMPLETED	I.A.3.1 I.C.1. II.K.2	Evaluation provided in NUREG-0565
5.	Complete TMI-2 simulator training for all operators and senior operators.	All Orders	COMPLETED	I.A.3.1	
6.	Reevaluate analysis submitted for dual-level setpoint control of OTSG in light of accident at TMI-2.	Davis-Besse 1 Order only	COMPLETED	NOT APPLICABLE	
7.	Reevaluate Davis-Besse 1 transient of September 24, 1977, in light of accident at TMI-2.	Davis-Besse 1 Order only	COMPLETED	NOT APPLICABLE	
8.	Submit a detailed thermal-mechanical report which shows the effect of long-term feed and bleed on reactor vessel integrity (SBLOCA with no AFW)	Letter from D. Ross to B&W Licensees dated 08/21/79	01/01/81	II.K.2 Table C.2 Item 13	Residual requirement identified during the review of the short-term requirements of the Orders.

CATEGORY SOURCE: COMMISSION ORDERS OF MAY 1979 (Short-Term Requirements) (continued)

Number	Requirement	Source	Implementation	Action Plan	Remarks
9.	Evaluation of PORV and safety valve lift frequency and increase in reactor trip frequency based on revised setpoints for PORV and high pressure reactor trip.	Letter from R. Reid to B&W Licensees dated 09/28/79	01/01/81	II.K.2 II.C.1 Table C.2 Item 14 Table C.3 Item 7	Same comment as requirement 8 above
10.	Analysis of effects of slug flow on OTSG tubes induced by transitioning between solid natural circulation and reflux boiling modes.	Letter from R. Reid to B&W Licensees dated 11/21/79	06/01/80	II.K.2 Table C.2 Item 15 Table C.3 Item 44	Same comment as Requirement 8 above
11.	Evaluation of the impact of RCP seal damage following a small break LOCA with loss of offsite power.	Letter from R. Reid to B&W Licensees dated 11/21/79	06/01/80	II.K.2 Table C.2 Item 16 Table C.3 Item 41	Same comment as Requirement 8 above
12.	Perform benchmark analysis of sequential AFW flow to OTSG using CRAFT-2 with 3 node OTSG representation.	Letter from D. Ross to B&W Licensees dated 08/21/79	01/01/81	II.K.2 Table C.2 Item 19	Same comment as Requirement 8 above
13.	Analysis of system response to a small break LOCA that causes the RCS to repressurize to the PORV setpoint (Two break LOCA)	Letter from D. Ross to B&W Licensees dated 08/21/79	01/01/81	II.K.2 Table C.2 Item 20	Same comment as Requirement 8 above
14.	LOFT L3-1 predictions	Letter from D. Ross to B&W Licensees dated 08/21/79	COMPLETED	II.K.2 Table C.2 Item 21	Same comment as Requirement 8 above. Results undergoing review by EG&G, Idaho.

CATEGORY SOURCE: COMMISSION ORDERS OF MAY 1979 (Short-Term Requirements) (continued)

Number	Requirement	Source	Implementation	Action Plan	Remarks
15.	Analyses of loss of feedwater and other anticipated transients and the development of guidelines and emergency procedures.	Letter from D. Ross to B&W Licensees dated 08/21/79	SB-COMPLETED ICC-COMPLETED TRANSIENTS/ ACCIDENTS 07/01/80	I.C.1 Table C.2 Item 18	Same comment as Requirement 8 above.
16.	Analysis of potential voiding in the RCS during anticipated transients.	Letter from R. Reid to B&W Licensees dated 01/09/80	01/01/81	I.C.1 Table C.2 Item 17	Concern identified by IE inspector. Connected with requirement 15 above.

SHORT-TERM AFW SYSTEM UPGRADE REQUIREMENTS SPECIFIED IN THE COMMISSION ORDERS OF MAY 1979 FOR B&W OPERATING PLANTS

ACTIONS REQUIRED	OCONEE 1-3	ANO-1	RANCHO SECO	CRYSTAL RIVER-3	DAVIS-BESSE 1
1. Provide for automatic start of all three EFW 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 (Manual capability)		X	X	X	
3. Station an operator at local valves during testing in communications with the control room.		X	X	X	
4. Develop procedures for control of AFW independent of ICS.		X	X	X	X
5. Verification of operability of AFW pumps.		X		X	
6. Provide for obtaining alternate sources of water for AFW.		X	X	X	
7. Provide for automatic start of motor-driven AFW pumps.		X		X	
8. Provide for timely operator notification of an automatic AFW intitiation.		X	X	X	
9. Provide for timely operator verification of AFW flow to OTSG upon automatic initiation of AFW.		X	X		
10. Verification that Tech. Spec. requirements for AFW are in accordance with the accident analysis.		X	X		
11. Provide AFW flow rate indication in the Control Room.		X	X		
12. Verify failure position of AFW flow control valves.			X	X	X

SHORT-TERM AFW SYSTEM UPGRADE REQUIREMENTS SPECIFIED IN THE COMMISSION ORDERS OF MAY 1979 FOR B&W OPERATING PLANTS

ACTIONS REQUIRED	OCONEE 1-3	ANO-1	RANCHO SECO	CRYSTAL RIVER-3	DAVIS-BESSE 1
13. Remove interlock which prevents the turbine-driven pump from injecting when the motor-driven pump is operating.				X	
14. Install dynamic breaking on turbine-driven pumps speed change motor.					X

NOTES ON THE COMMISSION ORDERS OF MAY 1979 (SHORT-TERM REQUIREMENTS)

*1. Applicable Orders

Facility:	Date of Order:	Letter Authorizing Restart (Enclosed staff evaluation of compliance with short-term requirements)
Oconee 1,2,3	05/07/79	05/18/79
Rancho Seco	05/07/79	06/27/79
Crystal River 3	05/16/79	07/06/79
Davis-Besse 1	05/16/79	07/06/79
ANO-1	05/17/79	05/31/79

* Order issued to MET-ED on 08/09/79 not covered under B&OTF work although same requirements generic to B&W 177-FA plants apply.

2. Of the 16 requirements identified on pages A-7 through A-9 only the first 7 (1 through 7) appear in the Orders. The remaining requirements (8 through 16) are residual items generated during the course of our review of the B&W licensees' compliance with the requirements of the Orders.

CATEGORY SOURCE: COMMISSION ORDERS OF MAY 1979 (LONG-TERM REQUIREMENTS)

Number	Requirement	Source	Implementation	Action Plan	Remarks
1.	Continued upgrade of AFW system reliability.	All Orders	01/01/81	II.E.1 II.K.2 Table C.2 Item 8	The Orders required the actions specified on page A-13; however, each B&W plant has performed an AFW reliability study from which additional requirements will be developed.
2.	Submit a Failure Mode and Effects Analysis of the ICS.	All Orders	01/01/81	II.K.2 Table C.2 Item 9	Report submitted on 08/17/79. ORNL con- for review. ORNL report received 01/21/80.
	(Licensees requested to report what actions they were taking based upon B&W recommendations)	Letter from R. Reid to B&W Licensees dated 11/07/79			Responses presently undergoing review.
3.	Upgrade the anticipatory reactor trip for loss of feedwater and turbine trip to safety-grade.	All Orders	01/01/81	II.K.2 Table C.2 Item 10	Licensees given preliminary design approval for safety-grade design 12/20/79. Instructed to install within 6 months.
4.	Continued operator training and drilling to assure a high state of preparedness	All Orders	01/01/81	I.A.3.1 I.A.2.2 I.A.2.5 I.G.1 II.K.2 Table C.2 Item 11	Licensees have complied with the intent of this long-term portion of the Order.
5.	Continued attention to transient analysis and procedures for management of small breaks.	Davis-Besse 1 Order only	01/01/81	I.C.1 II.K.2 Table C.2 Item 12	

LONG-TERM AFW SYSTEM UPGRADE REQUIREMENTS SPECIFIED IN THE COMMISSION ORDERS OF MAY 1979 FOR B&W OPERATING PLANTS

ACTIONS REQUIRED	OCONEE 1-3	ANO-1	RANCHO SECO	CRYSTAL RIVER 3	DAVIS-BESSE 1
1. Install two motor-driven EFW pumps per unit (in addition to present turbine-driven pump).	X				
2. Connect motor-driven AFW pump to a vital bus.		X			
3. Install AFW Automatic Control System (developed by B&W) which is completely separate from ICS.		X X			
4. Modify suction piping to improve separation.		X			
5. Provide control room annuncation for all AFW automatic start conditions.		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 rate verification in the Control Room.				X	
9. Continue to review performance of AFW system to assure reliability and performance.					X

NOTES ON COMMISSION ORDERS OF MAY 1979 (LONG-TERM REQUIREMENTS)

1. AFW system upgrade: Although the short and long-term requirements of the Orders do not appear to be consistent, the additional requirements generated during staff review of the AFW reliability study should bring each of the facilities' AFW system up to a fairly equal level of reliability. (see requirement #1).

2. FMEA: The report submitted by B&W on 08/17/79 (BAW-1564) contained an operating history section as well as an FMEA. One of the items identified by B&W in the report was a need to review and upgrade the ICS and NNI power supplies as well as the NNI inputs to ICS. (see requirement #2).

3. Safety-grade reactor trip: Although TMI-2 Action Plan has this item being implemented by 01/01/81, the licensees were informed to have the safety-grade reactor trip installed and operational in approximately six months from the date they received the staff's preliminary design approval. That approval letter was sent to the B&W licensees on 12/20/79 (see requirement #3).

CATEGORY SOURCE: NUREG-0565 (B&W SBLOCA EVALUATION)

Number	Requirement	Source	Implementation	Action Plan	Remarks
1.	Install automatic PORV isolation system and perform an operational test	2.1.2a	01/01/81 (I) First Refuel-ing (test)	II.K.3 Table C.3 Item 1	
2.	Evaluation of PORV opening probability during overpressure transients	2.1.2b	01/01/81	II.K.2 Table C.2 Item 14 Table C.3 Item 7	Same as requirement number 9 listed under Commission Order (Short-Term Requirements)
3.	Reporting of failures and challenges to the PORV	2.1.2c	04/01/80	II.K.3 Table C.3 Item 3	
4.	Evaluation of safety valve reliability	2.1.2d	01/01/81	II.K.3 Table C.3 Item 2	
5.	Reporting of failures and challenges to the safety valves	2.1.2e	04/01/80	Same as requirement 3 above	
6.	Revise and submit analysis methods for SBLOCA for approval under 10 CFR 50.46	2.2.2a	01/01/82	II.K.3 Table C.3 Item 31	
7.	Plant-specific calculations to show compliance with 10 CFR 50.46	2.2.2b	01/01/83	II.K.3 Table C.3 Item 32	
8.	Evaluation of effects of core flood tank (CFT) injection on SBLOCAs	2.2.2c	07/01/80	I.C.1 Table C.3 Item 36	

CATEGORY SOURCE: NUREG-0565 (B&W SBLOCA EVALUATION) (continued)

Number	Requirement	Source	Implementation	Action Plan	Remarks
9.	Install automatic trip of RCPs during LOCA	2.3.2a	01/01/81	II.K.3 Table C.3 Item 5	
10.	Review and upgrade reliability and redundancy of nonsafety-grade equipment upon which SBLOCA mitigation relies	2.3.2b	NONE	II.C.1 II.C.2 II.C.3 Table C.3 Item 4	
11.	Minimum simulator training requirements for SBLOCA training	2.3.2c	01/01/81	I.A.4.1 Table C.3 Item 55	
12.	Additional staff audit calculations of B&W SBLOCA analysis	2.4.2a	As required	I.C.1 Table C.3 Item 37	
13.	Consideration of diverse decay heat removal path for Davis-Besse 1 (low head HPI pumps)	2.5.2a	Not Scheduled	II.C.1 Table C.3 Item 8	Consideration based upon outcome of Davis-Besse IREP
14.	Experimental verification of two-phase natural circulation	2.6.2a	01/01/82	II.K.3 Table C.3 Item 33	
15.	Instrumentation to verify natural circulation	2.6.2b	01/01/81	I.C.1 II.F.2 II.F.3 Table C.3 Item 6	Additional instrumentation for accident monitoring - 01/01/81. Complete balance per R.G. 1.97 by 06/01/82

CATEGORY SOURCE: NUREG-0565 (B&W SBLOCA EVALUATION) (continued)

Number	Requirement	Source	Implementation	Action Plan	Remarks
16.	Analysis of plant response to an SBLOCA which is isolated, causing RCS to repressurize to the PORV setpoint	2.6.2.c	06/01/80	I.C.1 Table C.3 Item 38	
17.	Analysis of plant response to an SBLOCA in the pressurizer spray line with a stuck open spray line isolation valve	2.6.2d	05/01/80	I.C.1 Table C.3 Item 39	
18.	Evaluation of the effects of water slugs in the piping caused by HPI and CFT flows	2.6.2e	05/01/80	I.C.1 Table C.3 Item 40	
19.	Evaluation of RCP seal damage and leakage during an SBLOCA	2.6.2f	06/01/80	II.K.2 Table C.2 Item 16 Table C.3	Same as requirement 11 listed under Commission Orders of May 1979 (Short-Term Requirements)
20.	Submit predictions of LOFT Test L3-6 (SBLOCA with RCPs running	2.6.2g	PRETEST	I.C.1 Table C.3 Item 42	
21.	Submit information requested in NUREG-0565 on effects of noncondensible gases	2.6.2h	05/01/80	I.C.1 Table C.3 Item 43	
22.	Evaluation of mechanical effects of slug flow on OTSG tubes induced by transitioning from solid natural circulation to reflux boiling	2.6.2i	06/01/80	II.K.2 Table C.2 Item 15 Table C.3 Item 44	Same as requirement 10 listed under Commission Orders of May 1979 (Short-Term Requirements)

CATEGORY SOURCE: NUREG-0578 (SHORT-TERM LESSONS LEARNED)

Number	Requirement	Source	Implementation	Action Plan	Remarks
1.	Emergency power supplies for: pressurizer heaters level indication, PORV and block valve	2.1.1	01/01/80	II.G	II.G does not cover pressurizer heaters
2.	PORV and safety valve testing	2.1.2			
	Program and schedule		01/01/80	II.D.1	
	Complete testing		07/01/81	II.D.2	
3.	PORV and safety valve direct position indication	2.1.3a	01/01/80	II.D.3 II.F	
4.	Instrumentation for inadequate core cooling:	2.1.3b			
	Analysis and procedures		01/01/80	I.C.1	
	Design of new instruments, Install sub-cooling meter		01/01/80	II.F.2	
	Install new instrumentation		01/01/80		
5.	Diverse containment isolation signal	2.1.4	01/01/80	II.E.4.2	
6.	Dedicated Hydrogen penetrations	2.1.5a			
	Design		01/01/80	II.E.4.1	
	Installation		01/01/81		
8.	Systems integrity for high radioactivity	2.1.6a		III.D.1.1	
	Leak reduction program		01/01/80		
	Preventive maintenance program		01/01/80		
9.	Plant shielding review	2.1.6b		II.B.2	
	Design review		01/01/80		
	Plant modifications		01/01/81		

CATEGORY SOURCE: NUREG-0578 (SHORT-TERM LESSONS LEARNED)

Number	Requirement	Source	Implementation	Action Plan	Remarks
10.	Automatic initiation of AFW	2.1.7a		II.E.1.2	
	Control-Grade		06/01/80		All B&W plants have auto initiation
	Safety-Grade		01/01/81		
11.	AFW flowrate indication	2.1.7b		II.E.1.2	
	Control-Grade		06/01/80		All B&W plants meet control-grade
	Safety-Grade		01/01/81		
12.	Post-accident sampling	2.1.8a		II.B.3	
	Design and procedures		01/01/80		
	Plant modifications		01/01/81		
13.	High range radiation monitors	2.1.8b			
	In containment		01/01/81	II.B.3	
	Effluent monitoring		12/01/81	III.D.2.1	
14.	Improved Iodine Instrumentation	2.1.8c	01/01/80	III.A.1 III.D.3.3	
15.	Transient and Accident Analysis	2.1.9		I.C.1	
	a. SBLOCA analysis and guidelines		09/30/79		
	b. SBLOCA procedures and training		01/01/80		
	c. Analysis of ICC and guidelines		10/31/79		
	d. ICC procedures and training		01/01/80		
	e. Accidents & Transients analysis and guidelines		04/01/80		
	f. Accidents & Transients procedures and training		07/01/80		

CATEGORY SOURCE: NUREG-0578 (SHORT-TERM LESSONS LEARNED)

Number	Requirement	Source	Implementation	Action Plan	Remarks
16.	Additional Accident Monitoring Instrumentation	Letter from		II.F.1	
	a. Containment pressure monitor	H. Denton	01/01/81	II.B.1	
	b. Containment water level monitor	to All	01/01/81		
	c. Containment Hydrogen monitor	Operating	01/01/81		
	d. RCS vents	Rx. dated	01/01/81		
	Design	10/30/79	01/01/80		
	Installation		01/01/81		
17.	Shift Supervisor Responsibilities	2.2.1a	01/01/80	I.C.3	
18.	Shift Technical Advisor	2.2.1b		I.A.1.1	
	On Duty		01/01/80		
	Fully Trained		01/01/81		
19.	Shift Turnover Procedures	2.2.1c	01/01/80	I.C.2	
20.	Control Room Access	2.2.2a	01/01/80	I.C.4	
21.	On-site Technical Support Center	2.2.2b		III.A.1.2	
	Establish Center		01/01/80		
	Upgrade to meet all requirements		01/01/81		
22.	On-site Operation Support Center	2.2.2c	01/01/80	III.A.1.2	

CATEGORY SOURCE: NUREG-0585 (FINAL REPORT OF LESSONS LEARNED TF)

Number	Requirement	Source	Implementation	Action Plan	Remarks
1.	Personnel qualification and training:				
	1.1 Utility Management Involvement	2.3.1	06/01/82	I.B.1.1	
	1.2 Training Programs	2.3.3	Various	I.A.2	
	1.3 In-Plant Drills	2.3.3/8	01/01/81	I.A.2	
	1.4 Operator Licensing	2.3.1/2	Various	I.A.1/.3/B.1	
	1.5 NRC Staff Coordination	2.3.1/2/3	N/A	I.A.2	
	1.6 Licensed Operator Qualification	2.3.1/2	Various	I.A.2/.3	
	1.7 Licensee Technical and Management Support	2.3.1/2	06/01/82	I.B.1	
	1.8 Licensing of additional operating personnel	2.3.1/2	N/A	I.A.3	
2.	Staffing of Control Rooms	2.3.5	07/01/81	I.A.1	
3.	Working Hours	2.3.5	07/01/80	I.A.1	
4.	Emergency Procedures	2.3.4	07/01/80	I.C	
5.	Verification of correct performance of operating activities	2.3.6	01/01/81	I.B.1 I.C.6	
6.	Evaluation of Operating Experience	2.3.7	N/A	I.E	
	6.1 Nationwide network				
	6.2 Providing information to operators				
7.	Man-Machine Interface				
	7.1 Control room reviews	2.3.5/8	03/01/81/82	I.D.1	Short/Long-Term
	7.2 Plant safety status displays	2.3.5/8	06/01/81	I.D.2	
	7.3 Disturbance analysis systems	2.3.5/8	N/A	I.D	
	7.4 Manual versus automatic operation	2.1/2.3.5	N/A	I.A.4	
	7.5 Standard Control Room Design	2.3.5/2.3/8	N/A	I.D	
8.	Reliability assessment of final design	3.2	07/00/90	II.c	Crystal River
9.	Review of safety classification and qualifications	3.2		II.C	
10.	Design features for core-melt and core-damage	3.3	Various	II.B	

APPENDIX B

TABULATION OF SELECTED B&W OPERATING HISTORY

TABLE B.1

HISTORICAL SUMMARY OF NNI/ICS POWER FAILURES

	Date	Plant	Problem	Cause	Power Failure	Feedwater Transient	Reactor Trip	PORV Opened	ESF Actuation	Initial Plant Status	Source* of Info.
A.	NNI/ICS Power Failures Which Resulted in Reactor Trip										
1.	06 DEC 74	Arkansas 1	Lost NNI-Y power supply & condenser vacuum pumps	Water leakage into B Main Chiller load controller	NNI	Yes	Hi Press	Yes		80%	4, 10
2.	8 JUL 76	Arkansas 1	Loss of NNI Power, trip on RC HI Pressure	Maintenance Personnel	NNI	Yes	Hi Press	Yes		94%	1, 4, 10
3.	02 MAR 77	Crystal River 3	Loss of ICS Power (RPS - did trip)	Inverter Diode Failure	ICS	Stm dump open	CRD Power failure		No; Hi Cooldown Rate (CDR)	40%	1, 7, 8, 10
4.	26 FEB 80	Crystal River 3	Loss of NNI X 24 PORV opened (RPS - did trip)	+24 VDC NNI-X bus shorted causing the power supply to open S1 and S2 disconnecting +24 VDC from VAC feed	NNI	Yes	Hi Press	yes stuck open	ES-HPI	99%	1, 7
5.	12 JAN 79	Davis-Besse 1	Failure of Inverter to vital power source (reactor trip and turbine trip) SFRCS trip	Accidental grounding of containment hydrogen analyzer causing inverter fuse failure.	NNI	Yes	Hi Press	yes		100%	1, 8, 9, 10

*Sources of information: Listed on page B-8.

TABLE B.1 (Continued)

	Date	Plant	Problem	Cause	Power Failure	Feedwater Transient	Reactor Trip	PORV Opened	ESF Actuation	Initial Plant Status	Source of Info.
A.	<u>NNI/ICS Power Failures Which Resulted in Reactor Trip</u>										
6.	14 JUL 76	Oconee 1	ICS power was lost (reactor tripped)	Power receptacle incorrectly wired into ICS Power Source. When High Voltage power supply was plugged in, line circuit breaker tripped.	ICS		Hi Press	yes		100%	1, 5, 10
7.	14 DEC 78	Oconee 1	Lost power to Tave recorder	Shorted while trouble-shooting alarm indication	NNI	Yes	Press-Temp		ES-HPI	98%	5, 8
8.	25 DEC78	Oconee 1	Loss of all Power (ac) (RPS - did trip)	Inverter Fuse Failure	ICS	Yes	Hi Press	yes		10%	1, 5, 10
9.	11 JUL 74	Oconee II	Lost ICS autopower	Maintenace Personnel	ICS		Low Press	yes stuck open	ES-HPI	80%	5
10.	23 SEP 74	Oconee II	Loss of Power to ICS Power Panel Board 2K1. (Reactor did trip)	Inverter Failiure	ICS	Yes	Hi Press	yes		95%	1, 5, 10
11.	10 NOV 79	Oconee III	Loss of power to ICS. Manual transfer to backup power required. After a loss of feedwater (RPS - did trip)	Loss of power to 120 VAC non-1E panel (inverter fuse failure)	ICS	Yes	Hi Press	No	No; Hi CDR	99%	1, 5, 8

TABLE B.1 (Continued)

	Date	Plant	Problem	Cause	Power Failure	Feedwater Transient	Reactor Trip	PORV Opened	ESF Actuation	Initial Plant Status	Source of Info.
A.	NNI/ICS Power Failures Which Resulted in Reactor Trip										
12.	22 NOV 74	Rancho Seco	Temporary loss of power to NNI "Y" and "Z" busses (Reactor did trip)	Due to Loss of "J" inverter (fuses blew)	NNI	Yes	Hi Press	Yes		32%	1, 6, 10
13.	26 DEC 74	Rancho Seco	Vital power bus Inverter "C" Tripped. Lost Power to ICS	Equipment (SCR) failure.	ICS	Yes	Hi Press	yes		40%	1, 6, 10
14.	31 DEC 74	Rancho Seco	Lost temporarily power to "J" inverter, and ICS/NNI power (Reactor tripped)	Inverter Fuse blew.	ICS		Press-temp	yes		40%	1, 6, 10
15.	28 DEC 74	Rancho Seco	Loss Power to ICS bus "X" (Reactor tripped).	Unknown.	ICS	Yes	Yes	Yes ?		40%	1, 6, 10
16.	16 APR 75	Rancho Seco	Lost of power to Inverter "B" Vital Power Bus.	Unknown.	ICS input		Yes	?		35%	1, 6
17.	20 MAR 78	Rancho Seco	Lost power to NNI "Y" power supply (Reactor did trip).	Dropped Light Bulb	NNI	Yes	Hi Press	yes & safety lifted	Hi CDR	72%	1, 6, 8

TABLE B.1 (Continued)

	Date	Plant	Problem	Cause	Power Failure	Feedwater Transient	Reactor Trip	PORV Opened	ESF Actuation	Initial Plant Status	Source of Info.
A.	<u>NNI/ICS Power Failures Which Resulted in Reactor Trip</u>										
18.	2 JAN 79	Rancho Seco	Loss of NNI Power (RPS - did trip)	Technician inadvertently shorted NNI power supply.	ICS input	Yes	Hi Press	yes		100%	1, 6, 10
19.	05 JAN 79	Rancho Seco	Loss of one ICS Power Supply	Electrical short by Technician	ICS	Yes	Hi Press	yes	Hi CDR	100%	1, 6, 8, 10
20.	22 APR 79	Rancho Seco	Loss of RC flow Signal to ICS	A Inverter trip	ICS input	Yes	Hi Press	No		100%	1, 6, 8, 10
B.	<u>NNI/ICS Power Failures Which Did Not Result in a Reactor Trip</u>										
1.	18 NOV 77	Arkansas 1	Loss of ICS Power Supply by Shorting of ICS Power Supply. RPS did not trip - Reactor Level reduced to 60% and then returned to normal level.	Inadvertent shorting of ICS power supply by Technician.	ICS		No		No		1, 4
2.	21 APR 77	Crystal River	X-Power Supply to ICS was lost, SG level went to 50% Turbine bypass and Atmos. dump valves opened, lost MFW.	Blew Fuse in vital Bus "B".	ICS	Yes	No		No	46%	1, 7, 10
3.	27 NOV 79	Oconee III	ICS inverter failure	Component failure (defective inverter Logic cards in 3K1)	ICS	No	No	No	No	99%	1, 5

TABLE B.1 (Continued)

	Date	Plant	Problem	Cause	Power Failure	Feedwater Transient	Reactor Trip	PORV Opened	ESF Actuation	Initial Plant Status	Source of Info.
C.	NNI/ICS Power Failures Which Occurred During Testing or Refueling Outages										
1.	29 OCT 75	Davis Besse 1	Failure of Power Supply Monitor Module BCCO #6625070A1	Defective Component							1
2.	11 MAR 76	Davis Besse 1	Failure of NNI Power Supply Monitor Module BCCO #6625070A1	Defective Component							1
3.	24 MAY 76	Davis Besse 1	Failure of NNI Power Supply Monitor Module BBCO #6625070A1	Defective Component							1
4.	11 FEB 76	Davis Besse 1	Failure of ICS Power Supply Monitor Module BCCO #6625070A1	Defective Component							1
5.	11 NOV 76	Davis Besse 1	Summary of Power Supply Monitor Module Failures	Summary of Failure							1
6.	3 MAY 79	Oconee III	ICS hand power breaker trip, due to defective recorder cord and plug assembly	Component failure (pressurizer WR level recorder)	ICS	NA	N/A	N/A	No	Cold shutdown	1, 5

TABLE B.1 (Continued)

	Date	Plant	Problem	Cause	Power Failure	Feedwater Transient	Reactor Trip	PORV Opened	ESF Actuation	Initial Plant Status	Source of Info.
C.	<u>NNI/ICS Power Failures Which Occurred During Testing or Refueling Outages</u>										
7.	29 MAR 78	TMI-2	Loss of vital power to NNI X bus, PORV opened (RPS - did trip)	Inverter fuse failure	NNI		Yes	Yes stuck open	Yes	0	1, 8

NOTE: For sources of information see next page.

Table B.1 (Continued)

Sources of Information

1. Letter from J. H. Taylor (B&W) to D. G. Eisenhut (NRC), Subject: Responses to Questions of March 4, 1980 concerning incident at CR-3, dated March 12, 1980.

2. Letter from J. G. Herbein (Met-Ed) to H. R. Denton (NRC), Subject: Responses to NRC letter of March 6, 1980 concerning incident at CR-3, dated March 13, 1980.

3. Letter from R. P. Crouse (TECO) to R. W. Reid (NRC), Subject: Responses to NRC letter of March 6, 1980 concerning incident at CR-3, dated March 13, 1980.

4. Letter form C. L. Steel (AP&L) to H. R Denton (NRC), Subject: Responses to NRC letter of March 6, 1980 concerning incident at CR-3, dated March 12, 1980.

5. Letter from W. O. Parker, Jr. (Duke) to H. R. Denton (NRC), Subject: Response to NRC letter of March 6, 1980 concerning incident at CR-3, dated March 12, 1980.

6. Letter from W. C. Walbridge (SMUD) to H. R. Denton (NRC), Subject: Response to NRC letter of March 6, 1980 concerning incident at CR-3, dated March 12, 1980.

7. Letter from J. A. Hancock (FPC) to H. R. Denton (NRC), Subject Response to NRC letter of March 6, 1980 concerning incident at CR-3, dated March 12, 1980.

8. Licensee Event Report File

9. U. S. Nuclear Regulatory Commission, "Operating Units Status Report," NUREG-0020 (Printed Monthly - commonly referred to as the "Gray Book").

10. Letter from G. C.Moore (FPC) to R. W. Reid (NRC), Subject: Information concerning the lift frequency of the PORV and safety valves, dated November 15, 1979.

TABLE B.2

REACTOR TRIP/PORV ACTUATION DATA FOR B&W PLANTS

Date	Transient Classification	Trip Signal	Cause of Transient	Initial Power Level	Pzr. Safety Valves Lifted	If Present Setpoints Had Been Used: PORV Actuation	If Present Setpoints Had Been Used: Lift Safety Valves
Reactor Trips with a PORV Actuation (pre-TMI-2 accident) - ANO-1							
10-15-74	Loss of Feedwater	Hi RCP*	"A" FWP Tripped on High Vibration	45	No	No	No
12-6-74	Loss of Feedwater	Pressure/ Temp	Loss of Vacuum Due to "B" Main Chiller Getting Wet and Shorting	80	No	No	No
- - - - - - Commercial Operation - - - - - -							
1-6-75	Load Rejection	Hi RCP	Generator Tripped on Differential Current Due to Loss of Bus Cooling	98.5	No	No	No
5-15-75	?	PWR/ Imbalance Flow	Flow Oscillations Occurred During Maneuvering	100	No	No	No
6-6-75	Instrument	PWR/ Imbalance	Loose Connection on Loop "B" T_c Signal	99	No	No	No
7-3-75	Instrument Failure	PWR Imbalance Flow	Technician Grounded T_H Signal to ICS	95	No	No	No

*"Hi RCP" indicates high reactor coolant pressure.

TABLE B.2 (Continued)

Date	Transient Classification	Trip Signal	Cause of Transient	Initial Power Level	Pzr. Safety Valves Lifted	If Present Setpoints Had Been Used: PORV Actuation	If Present Setpoints Had Been Used: Lift Safety Valves
Reactor Trips with a PORV Actuation (pre-TMI-2 accident) - ANO-1 (continued)							
7-23-75	Loss of Feedwater	Pressure/ Temp	Operator Lost Htr. Drain Pump Which Tripped FWP	50	No	No	No
7-8-76	Loss of Feedwater/ Power Supply Failure	Hi RCP	Inst. Techs Shorted NNI Power Supply	94	No	No	No
9-23-76	Turbine Trip	Hi RCP	Turbine Tripped When Vibration Trip Module was Reinserted by Technician	99	No	No	No
12-20-76	Rod Drop/Power Supply Failure	Hi RCP	Rod 8 in Group 4 Dropped Coupled with Loss of Y-11 Inverter	64	No	No	No
6-19-78	Turbine Trip	Hi RCP	Technician or Operator Error in Opening Wrong Feeder Breaker	?	No	No	No
9-16-78		Hi Flux	Burned Out Control Air Solenoid on MSIV	?	No	No	No
10-13-78	Instrument Failure	Pressure/ Temp	RPS Channel "B" RC Flow Signal Failed	?	No	No	No
12-20-78	Instrument Failure	Pressure/ Temp	Low Steam Pressure Caused by LVDT Linkage Breaking	99	No	No	No

Note: Based on Gray Book data, between 2/22/75 and 11/1/76, 8 automatic reactor trips not listed above occurred.

TABLE B.2 (Continued)

Date	Transient Classification	Trip Signal	Cause of Transient	Initial Power Level	Pzr. Safety Valves Lifted	If Present Setpoints Had Been Used	
						PORV Actuation	Lift Safety Valves
Reactor Trips with a PORV Actuation (pre-TMI-2 accident) - Crystal River 3							
1-30-77	Instrument Failure/ Turbine Trip	Lo RCP	Manual TT Due to Low Turbine Header Pressure Coupled with Inoperable MW Meter	15	No	No	No
3-2-77	Power Supply Failure	Loss of CRD Pwr	120 VAC-B Vital Bus was Lost Due to Failure of Diode in "B" Inventer?	40	No	No	No
3-9-77	----	Manual	Reactor - Turbine Trip Test TP-800-14	40	No	No	No
- - - - - - - - - - Commercial Operation - - - - - - - - - -							
4-21-77	Power Supply Failure (No Reactor Trip	None	"X" Power Supply to ICS Lost Due to Blown Fuse	46	No	No	No
4-23-77	- - - - -	Manual	Part of Test (Outside Control Room)	20	No	No	No
10-26-77	Power Supply Failure/Turbine Trip/Loss of Feedwater	Hi RCP	Inverter "A" Tripped Causing a Loss of Power to Vital Bus "A"	100	No	No	No
1-6-79	Turbine Trip	Manual	TT Followed by FW Block Valve FWV-30 Sticking Open or Partially Open	71	No	No	No

TABLE B.2 (Continued)

Date	Transient Classification	Trip Signal	Cause of Transient	Initial Power Level	Pzr. Safety Valves Lifted	If Present Setpoints Had Been Used: PORV Actuation	If Present Setpoints Had Been Used: Lift Safety Valves
Reactor Trips with a PORV Actuation (pre-TMI-2 accident) - Crystal River 3 (continued)							
1-17-79	Loss of Feedwater	Manual	Solenoid Failure on Inlet Seawater Block Valve to Secondary Services Heat Exchanger "A".	100	No	No	No
1-30-79	Loss of Feedwater	Lo RCP	Reason for Decrease in FW Not Stated	100	No	No	No
7-17-77	Rod Drop	Manual	Grp. 1 Dropped During Surveillance Test	90	No	No	No
11-13-77	Loss of Feedwater	Pressure/ Temp	FW Upset While Passing Block Valve Point. Cause is Operator Control and Poor Control System Operation/Performance	57	No	No	No

Note: Based on Gray Book data, between 3-18-77 and 2-28-79, 12 a[illegible]omatic reactor trips not listed above occurred.

TABLE B.2 (Continued)

Date	Transient Classification	Trip Signal	Cause of Transient	Initial Power Level	Pzr. Safety Valves Lifted	If Present Setpoints Had Been Used: PORV Actuation	If Present Setpoints Had Been Used: Lift Safety Valves
Reactor Trips with a PORV Actuation (pre-TMI-2 accident) - Davis-Besse 1							
9-2-77	Turbine Trip	Lo RCP	OTSG Overfed by Operator	~7.5	No	No	No
9-24-77	Loss of Feedwater	Manual	"Half-Trip" of SFRCS Isolated OTSGs	9	No	No	No
10-23-77	Loss of Feedwater	Lo RCP	SFRCS Caused Isolation of 1 OTSG, Later Both	16	No	No	No
- - - - - - Commercial Operation - - - - - -							
12-16-77	ICS in Manual	Lo RCP	Overfed "B" OTSG. Operator had MFW Pump in Hand	11	No	No	No
12-30-77	Loss of Feedwater	Lo RCP	FWP Tripped on High Exhaust Casing Water Level	72	No	No	No
1-21-78	Loss of Feedwater	Manual	Malfunction in Turbine Speed Control System Led to SFRCS Actuation	70	No	No	No
1-31-78	Loss of Feedwater	Hi RCP	Spurious SFRCS Trip after Performing SFRCS Monthly Test	67	No	No	No

TABLE B.2 (Continued)

Date	Transient Classification	Trip Signal	Cause of Transient	Initial Power Level	Pzr. Safety Valves Lifted	If Present Setpoints Had Been Used	
						PORV Actuation	Lift Safety Valves
Reactor Trips with a PORV Actuation (pre-TMI-2 accident) - Davis-Besse 1 (continued)							
3-1-78	Loss of Feedwater	Hi RCP	SFRCS Actuated on FW/STM Pressure ΔP; Deaerator Level Cont. Valve Failed Shut	49	No	No	No
4-2-78	Turbine Trip	Lo RCP	TT Test - During Runback, Rx Tripped, Overfed OTSG's	75	No	No	No
8-2-78	Overcooling	Lo RCP	FW Oscillations	40	No	No	No
9-10-78	Turbine Trip	Lo RCP	Tripped Turbine for Test TP-800-14	~75	No	No	No
9-28-78	Turbine Trip	Lo RCP	Loop 2 RCS Flow XMTR Failed Low, Runback @ 20%/Min Initiated. Operator Lost Control	90	No	No	No
10-3-78	Turbine Trip	Lo RCP	TT Caused by Starting 2nd EHC Pump. ICS Oversupplied FW, OTSG's Overfed	68	No	No	No
10-29-78	Loss of Feedwater	Lo RCP	EM Relief Cycled and Stuck Open Too Long	4	No	No	No
11-13-78	Power Supply Failure No PORV Actuation	Power to Pumps	Fuse for RC Pump Control Circuitry Blew	99	No	No	No

TABLE B.2 (Continued)

Date	Transient Classification	Trip Signal	Cause of Transient	Initial Power Level	Pzr. Safety Valves Lifted	If Present Setpoints Had Been Used: PORV Actuation	If Present Setpoints Had Been Used: Lift Safety Valves
Reactor Trips with a PORV Actuation (pre-TMI-2 accident) - Davis Besse 1 (continued)							
1-12-79	Loss of Feedwater	Hi RCP	Technician Shorted Inverter Causing Loss of Vital Bus Y2; SFRCS Trip	100	No	No	No
2-22-79	Loss of Feedwater	Manual	Malfunction in Turbine Speed Control System Led to SFRCS Actuation	87	No	No	No

Note: Based on Gray Book data, between 11-29-77 and 2-13-79, 12 automatic reactor trips not listed above occurred.

TABLE B.2 (Continued)

Date	Transient Classification	Trip Signal	Cause of Transient	Initial Power Level	Pzr. Safety Valves Lifted	If Present Setpoints Had Been Used: PORV Actuation	If Present Setpoints Had Been Used: Lift Safety Valves
Reactor Trips with a PORV Actuation (pre-TMI-2 accident) - Oconee 1							
5-5-73	Loss of Feedwater	Manual	Operations Error Tripped Main FW Pump	18	No	No	No
5-16-73	Loss of Feedwater	Hi RCP	Operator Error	15	No	No	No
5-23-73	Loss of Feedwater	Hi RCP	Operator and/or Procedure Error	25	No	No	No
5-26-73	Loss of Feedwater	Hi RCP	Operator/Procedure Error	35	No	No	No
5-27-73	Loss of Feedwater	Hi RCP	Began at CRDM Fault	40	No	No	No
5-28-73	Loss of Feedwater	Hi RCP	Attempt to Transfer from B to A FWP	40	No	No	No
5-30-73	Loss of Feedwater	Hi RCP	Cleaning Hotwell Pump Strainer	40	No	No	No
6-9-73	Loss of Feedwater	Hi RCP	Switching Powdex Units	40	No	No	No
6-13-73	Turbine Trip	Manual	CRDM Fault Following Turbine Trip Test and ICS Runback Signal	52	No	No	No
6-14-73	Loss of Feedwater	Hi RCP (?)	Maintenance Work on Hotwell Strainer	40	No	No	No

TABLE B.2 (Continued)

Date	Transient Classification	Trip Signal	Cause of Transient	Initial Power Level	Pzr. Safety Valves Lifted	If Present Setpoints Had Been Used	
						PORV Actuation	Lift Safety Valves
Reactor Trips with a PORV Actuation (pre-TMI-2 accident) - Oconee 1 (continued)							
6-21-73	Loss of Feedwater	RCP/Temp Ratio	Tripped Hotwell Pump Initiated FWP Trip	19	No	No	No
Commercial Operation							
7-15-73	Loss of Feedwater	Hi RCP	Faulty Speed Controller on a Main FW Pump	75	No	No	No
8-11-73	Turbine Trip	Loss PWR to Pumps Ind.	Inadvertent Closure of Turbine Intercept Valves	85	No	No	No
9-16-73	Turbine Trip	Hi Temp Pressure Ratio	Manually Initiated Turbine Trip Decreased Closing Setpoint of Bypass Valves	40	No	No	No
10-12-73	Turbine Trip	Hi RCP	Main Steam Bypass Valves did not open: Operator Put Rods in Manual	20	No	No	No
10-26-73	Turbine Trip	Hi RCP	Loss of Condenser Vacuum	75	No	No	No
12-11-73	Turbine Trip	Hi RCP	Spurious MWe Signal Detected by EHC System Led to Turbine Trip	90	No	No	No

TABLE B.2 (Continued)

Date	Transient Classification	Trip Signal	Cause of Transient	Initial Power Level	Pzr. Safety Valves Lifted	If Present Setpoints Had Been Used: PORV Actuation	If Present Setpoints Had Been Used: Lift Safety Valves
Reactor Trips with a PORV Actuation (pre-TMI-2 accident) - Oconee 1 (continued)							
8-23-74	Load Rejection	Hi RCP	Unit Loss of Electrical Load Acceptance Test	95	No	No	No
3-12-75	Loss of Feedwater	Hi RCP	Shorted Transistor in ΔT_C Controller	25	No	No	No
4-22-75	Turbine Trip	Hi RCP	Loss of EHC Control Power	100	No	No	No
4-23-75	Loss of Feedwater	Hi RCP	Rapid Feedwater Oscillations	46	No	No	No
6-8-75	Turbine Trip	Hi RCP	Low EHC Hydraulic Pressure	100	No	No	No
6-9-75	Loss of Feedwater	Hi RCP	FW Flow Oscillation	30	No	No	No
8-2-75	Instrument Failure	Hi RCP	Failure of Temperature Switch on Stator Coolant System	75	No	No	No
8-8-75	Turbine Trip	Flux/ Flow	Positive Voltage Spike in Turbine Speed Error Circuit	92	No	No	No
1-22-76	Turbine Trip	Hi RCP	Loss of Excitation on Generator	100	No	No	No
5-31-76	Loss of Feedwater	Hi RCP	FWP Turbine Speed Momentarily Decreased when Switching from Auxiliary to Main Steam	~15	No	No	No

TABLE B.2 (Continued)

Date	Transient Classification	Trip Signal	Cause of Transient	Initial Power Level	Pzr. Safety Valves Lifted	If Present Setpoints Had Been Used: PORV Actuation	If Present Setpoints Had Been Used: Lift Safety Valves
Reactor Trips with a PORV Actuation (pre-TMI-2 accident) - Oconee 1 (continued)							
6-27-76	Instrument Failure	Hi RCP	Short in Signal Amp for RPS Flow Indication: Secondarily Flow Runback	100	No	No	No
7-7-76		Hi RCP	Personnel Error	99	No	No	No
7-14-76	Loss of Feedwater and Power Supply Failure	Hi RCP	ICS Hand Power Circuit Breaker Tripped When Circuit was Overloaded with Calibration Equipment	100	No	No	No
8-14-76	Rod Drop	Hi RCP	Heat and Moisture Affected Electrical Components in CRD System Cabinets	60	No	No	No
4-3-77	Instrument Failure	Hi RCP	Failure of ICS Component	~15	No	No	No
4-24-77	Turbine Trip	Hi RCP	Misaligned Linkage Caused High Moisture Separator Reheater Drain Tank Level	68	No	No	No
5-24-77	Turbine Trip	Hi RCP	Loss of Condenser Vacuum	70	No	No	No
6-6-77	Turbine Trip	Hi RCP	Personnel Error	99	No	No	No
10-18-77	Loss of Feedwater	Hi RCP	Standby Condensate Pumps Off	15	No	No	No

TABLE B.2 (Continued)

Date	Transient Classification	Trip Signal	Cause of Transient	Initial Power Level	Pzr. Safety Valves Lifted	If Present Setpoints Had Been Used	
						PORV Actuation	Lift Safety Valves
Reactor Trips with a PORV Actuation (pre-TMI-2 accident) - Oconee 1 (continued)							
12-30-77	Loss of Feedwater	Hi RCP	Personnel Error - Inadvertent Closure of MFW Block Valve	100	No	No	No
6-1-78	Turbine Trip	Hi RCP	High Level in Moisture Separator Drain Tank Caused by Failure of MSDT Dump Valve and MSDT Level Control Valve	95	No	No	No
8-2-78	Turbine Trip and Power Supply Failure	Hi RCP	EHC-DC Power Lost	100	No	No	No
12-25-78	Power Supply Failure	Hi RCP	Blown Fuses Led to Loss of Feedwater	10	No	No	No
3-23-79	Instrument Failure	Hi RCP	Startup FW Summer Module Failed	100	No	No	No

Note: Based on Gray Book data, between 3-13-75 and 12-26-78, 11 automatic reactor trips not listed above occurred.

TABLE B.2 (Continued)

Date	Transient Classification	Trip Signal	Cause of Transient	Initial Power Level	Pzr. Safety Valves Lifted	If Present Setpoints Had Been Used	
						PORV Actuation	Lift Safety Valves
Reactor Trips with a PORV Actuation (pre-TMI-2 accident) - Oconee 2							
12-2-73	Loss of Feedwater	Hi RCP	Leak in 1" Line Around One FWP. Operator Manipulation of FW SU Valves Led to LOFW	15	No	No	No
12-3-73	Loss of Feedwater	Hi RCP	Control Rod Groups 6 and 7 Lost Proper Overlap	9	No	No	No
12-12-73	Loss of Feedwater	Hi RCP	Too Large a Pressure Loss in Powdex Units. Condensate Booster Pumps Tripped	30	No	No	No
1-4-74	Turbine Trip	Hi RCP	Erroneous Activation of Breaker Failure Relay System	75	No	No	No
5-30-74	Manual Rx Trip	Manual	Operator Mistakenly Injected HPI Water into RC System	75	No	No	No
6-13-74	Turbine Trip	Hi RCP	Turbine Intercept Valves Closed Due to Faulty Pot	?	No	No	No
7-11-74	Instrument Failure	Lo RCP	Loss of ICS Auto Power opened PORV	80	-	-	-
- - - - - - - - - - - - Commercial Operation - - - - - - - -							
9-17-74	Turbine Trip	Hi RCP	TT During Testing of Thrust Bearing Wear Detector	100	No	No	No

TABLE B.2 (Continued)

Date	Transient Classification	Trip Signal	Cause of Transient	Initial Power Level	Pzr. Safety Valves Lifted	If Present Setpoints Had Been Used: PORV Actuation	If Present Setpoints Had Been Used: Lift Safety Valves
Reactor Trips with a PORV Actuation (pre-TMI-2 accident) - Oconee 2 (continued)							
9-23-74	Power Supply Failure and Loss of Feedwater	Hi RCP	ICS Power Lost During Switching of Feeds to Inverter. Main Feed Pumps Tripped	95	No	No	No
3-27-75	Load Rejection	Hi RCP	Loss of Electrical Load Test During Startup	100	No	No	No
3-27-75	Instrument Failure	Manual	Loss of Condenser Vacuum Led to FWP Trip	15	No	No	No
4-1-75	Turbine Trip	Hi RCP	Manual Trip as Part of Turbine/Reactor Trip Test	100	No	No	No
8-5-79	Loss of Feedwater	Hi RCP	Blown Gasket on Emergency Governor Lockout Valve in Hydraulic Control System	62	No	No	No
8-23-75	Loss of Feedwater	Manual	Malfunction of Condenser Vacuum Switches Tripped FWPs. Reactor Manually Tripped	14	No	No	No
9-19-75	Loss of Feedwater	Manual	FWP Trip on Low Vacuum Manual Reactor Trip	10	No	No	No

TABLE B.2 (Continued)

Date	Transient Classification	Trip Signal	Cause of Transient	Initial Power Level	Pzr. Safety Valves Lifted	If Present Setpoints Had Been Used: PORV Actuation	Lift Safety Valves
Reactor Trips with a PORV Actuation (pre-TMI-2 accident) - Oconee-2 (continued)							
7-12-76	Turbine Trip	Hi RCP	FW Oscillation Occurred While Taking Main Turbine Off Line	23	No	No	No
7-27-76	Loss of Feedwater	Hi RCP	Steam Leak on Main Turbine Caused Load to Hold at 20%. ICS Caused FW Oscillations	20	No	No	No
9-7-76	Turbine Trip	Hi RCP	Back-up Speed Control System Failed and Intercept Valves Closed During TT Test	100	No	No	No
5-4-78	Turbine Trip	Hi RCP	Moisture Separator Level Control Failed to Function	100	No	No	No
10-17-78	Loss of Feedwater	Hi RCP	Air Line Blew Off Startup FW Valve	100	No	No	No
10-30-78	Loss of Feedwater	Hi RCP	Welding Crew Ignited Oil Around FW Pump with Sparks, Causing FWP to Trip	55	No	No	No
10-30-78	Loss of Feedwater	Hi RCP	FW Pump Leak. Switching of Pumps not Accomplished	12	No	No	No

Note: Based on Gray Book data, between 5-1-75 and 8-24-78, 7 automatic reactor trips not listed above occurred.

TABLE B.2 (Continued)

Date	Transient Classification	Trip Signal	Cause of Transient	Initial Power Level	Pzr. Safety Valves Lifted	If Present Setpoints Had Been Used: PORV Actuation	If Present Setpoints Had Been Used: Lift Safety Valves
Reactor Trips with a PORV Actuation (pre-TMI-2 accident) - Oconee 3							
10-13-74	Loss of Feedwater	Power to Pumps	Debris Obstructed Hotwell Pump Strainer	15	No	No	No
10-17-74	Loss of Feedwater	Hi RCP	Debris Obstructed Hotwell Pump Strainer	16	No	No	No
- Commercial Operation - - - - - - - -							
4-7-75	Loss of Feedwater	Hi RCP	Servicing Powdex Tripped Condensate Booster Pump	75	No	No	No
4-30-75	Load Rejection	Hi RCP	Loss of Electrical Load Test	100	Yes	No*	No
6-13-75	----	Low RCP	While Shutting Down, Turbine Switched to Manual at 19%, Bypassed Valves Opened, ULD Increased FW Demand → FWP and OTSG Level Oscillations	19	No	No	No
7-1-76	Turbine Trip	Power to Pumps	TT on Low Turbine Shaft Oil Pressure	98	No	No	No
4-6-77	Turbine Trip	Power to Pumps	TT on Momentary Loss of DC Power to EHC	100	No	No	No
8-21-77	Loss of Feedwater	Hi RCP	Manual Adjustment of FW by Operators	15	No	No	No

TABLE B.2 (Continued)

Date	Transient Classification	Trip Signal	Cause of Transient	Initial Power Level	Pzr. Safety Valves Lifted	If Present Setpoints Had Been Used: PORV Actuation	If Present Setpoints Had Been Used: Lift Safety Valves
Reactor Trips with a PORV Actuation (pre-TMI-2 accident) - Oconee 3 (continued)							
11-3-78	Loss of Feedwater	Hi RCP	TT Due to Low FWP Discharge Pressure	44	No	No	No
2-21-79	Instrument Failure	Hi RCP	Noise Spike on ICS Cable After CRDM Testing	99	No	No	No

Note: Based on Gray Book data, between 4-27-75 and 11-7-78, 18 automatic reactor trips not listed above occurred.

*Setpoint would have been reached, but block valve was closed.

TABLE B.2 (Continued)

Date	Transient Classification	Trip Signal	Cause of Transient	Initial Power Level	Pzr. Safety Valves Lifted	If Present Setpoints Had Been Used	
						PORV Actuation	Lift Safety Valves
Reactor Trips with a PORV Actuation (pre-TMI-2 accident) - Rancho Seco							
11-19-74	Loss of Feedwater	Manual	FW Oscillation During ICS Tuning	25	No	No	No
11-22-74	Loss of Feedwater Power Supply Failure	Hi RCP	Technician Misoperation. Power Lost to "Y" and "Z" NNI Busses	32	No	No	No
12-4-74	Loss of Feedwater	Hi RCP	Inadvertent Actuation of Reheater Intercept Valve	40	No	No	No
12-15-74	Rod Drop (GRP. 6&7)	Lo RCP	CRDM Motor Fault: Programmer Assembly	39.5	No	No	No
12-17-74	Rod Drop (GRP. 7)	Lo RCP	Same as Preceding Transient	41.3	No	No	No
12-26-74	Loss of Feedwater and Power Supply Failure	Hi RCP	Failure of 2 SCR's in "C" Inverter	39.5	No	No	No
12-31-74	Power Supply	Pressure/ Temp.	Operator Error in Paralleling Inverters	40	No	No	No
2-12-75	Turbine Trip	Hi RCP	Spurious Overspeed Trip Signal	92	No	No	No
2-18-75	Manual Load Rejection for Trip	Lo RCP	Poor ICS Tuning	75	No	No	No

TABLE B.2 (Continued)

Date	Transient Classification	Trip Signal	Cause of Transient	Initial Power Level	Pzr. Safety Valves Lifted	If Present Setpoints Had Been Used	
						PORV Actuation	Lift Safety Valves
Reactor Trips with a PORV Actuation (pre-TMI-2 accident) - Rancho Seco (continued)							
4-14-75	Loss of Feedwater	Hi RCP	Startup Valve in Auto (Closed), but "A" OTSG Blew Down	15	No	No	No
- - - - - - - - - - - - Commercial Operation - - - - - - - - - -							
6-15-75	Loss of Feedwater	Hi RCP	Transferring Steam Supply for FWP from Aux. to Main Steam	13	No	No	No
10-10-76	Loss of Feedwater	Hi RCP	FWP Speed Control Lost FWP Governor was Dirty	13.6	No	No	No
10-10-76	Loss of Feedwater	Hi RCP	Same as Preceding Transient	?	No	No	No
1-13-77	Loss of Feedwater	Hi RCP	Technician Shorted Out FWP Thrust Bearing Indicator	98	No	No	No
1-5-78	Unknown	Hi RCP (?)	Unknown	100	No	No	No
3-20-78	Loss of Feedwater	Hi RCP	Dropped Light Bulb Shorted NNI Cabinet	72	Yes	No	No
12-31-78	Loss of Feedwater	Hi RCP	Condensate Valve Failure	100	No	No	No

TABLE B.2 (Continued)

Date	Transient Classification	Trip Signal	Cause of Transient	Initial Power Level	Pzr. Safety Valves Lifted	If Present Setpoints Had Been Used: PORV Actuation	If Present Setpoints Had Been Used: Lift Safety Valves
Reactor Trips with a PORV Actuation (pre-TMI-2 accident) - Rancho Seco (continued)							
1-2-79	Loss of Feedwater	Hi RCP	Loss of Vital Bus 1A	100	No	No	No
1-5-79	Loss of Feedwater	Hi RCP	Technician Shorted Wires in ICS Cabinets	100	No	No	No

Note: Based on Gray Book data and Table B.1, between 12-28-79 and 9-3-78, 9 automatic reactor trips not listed above occurred.

TABLE B.2 (Continued)

Date	Transient Classification	Trip Signal	Cause of Transient	Initial Power Level	Pzr. Safety Valves Lifted	If Present Setpoints Had Been Used: PORV Actuation	If Present Setpoints Had Been Used: Lift Safety Valves
Reactor Trips with a PORV Actuation (pre-TMI-2 accident) - TMI-1							
6-18-74	Loss of Feedwater	Hi RCP	"A" Instrument air Compressor Tripped on Thermal Overload	7	No	No	No
7-13-74	Loss of Feedwater	Pressure/ Temp	LOFW Noticed Prior to 3-sec. Rod Withdrawal	15	No	No	No
7-14-74	Loss of Feedwater	Pressure/ Temp	Technician Grounded T_{AVE} Signal	76	No	No	No
8-13-74	Load Rejection	Hi RCP	Generator Trip Test	98	No	No	No
8-30-74	Turbine Trip	Hi RCP	Turbine Bearing Failure	75	No	No	No
- - - - - Commercial Operation - - - - -							
3-30-75	Turbine Trip	Hi RCP	Erroneous Signal From Faulty 701 Relay Indicated Loss of 125-V Supply to Turbine EHC Systems	100	No	No	No
5-9-75	Turbine Trip	Hi RCP	"B" Moisture Separator Drain Tank High Level Trip Device Shorted	100	No	No	No

TABLE B.2 (Continued)

Date	Transient Classification	Trip Signal	Cause of Transient	Initial Power Level	Pzr. Safety Valves Lifted	If Present Setpoints Had Been Used: PORV Actuation	If Present Setpoints Had Been Used: Lift Safety Valves
Reactor Trips with a PORV Actuation (pre-TMI-2) - TMI-1 (continued)							
6-18-75	Turbine Trip	Hi RCP	Voltage Spikes Transmitted Into Turbine EHC System	100	No	No	No
11-14-77	ICS Component Failure. No PORV Actuation Reported	Flux/Flow Imbalance	ICS Signal-Converter "L" Module Failed to Midrange	100	No	No	No

Note: Based on Gray Book data, between 5-22-75 and 5-27-76, 3 automatic reactor trips not listed above occurred.

TABLE B.2 (Continued)

Date	Transient Classification	Trip Signal	Cause of Transient	Initial Power Level	Pzr. Safety Valves Lifted	If Present Setpoints Had Been Used: PORV Actuation	If Present Setpoints Had Been Used: Lift Safety Valves
Reactor Trips with a PORV Actuation - TMI-2							
			No Pumps in Loop "A" Signal. Fuse Blew in 2-IV. RCP-2A Already Out				
4-19-78	Loss of Feedwater	Hi RCP	Operator Blew Down Condensate Strainers	15	No	No	No
9-20-78	Loss of Feedwater	Hi RCP	Valving Error Tripped Condensate Booster Pump	24	No	No	No
9-21-78	Low Feedwater	Hi RCP	Feed Pump and Feed Reg. Valve Problems	19	No	No	No
9-25-78	Load Rejection	Hi RCP	High Pressure Due to Reducing Load on Turbine. Incorrect Suction Pressure Switch or Logic Error on C.B. Pumps Caused FWP Trip	17	No	No	No
10-14-78	Loss of Feedwater	Lo RCP	FWP-1A Lost	26	No	No	No
11-7-78	Loss of Feedwater	Pressure/ Temp	TP-800-05 (Reactivity Coefficients) was being performed at T_{AVE} - 588F. Heater Drain Tank Low Level Alarm Tripped FWP 1B.	92	No	No	No

TABLE B.2 (Continued)

Date	Transient Classification	Trip Signal	Cause of Transient	Initial Power Level	Pzr. Safety Valves Lifted	If Present Setpoints Had Been Used	
						PORV Actuation	Lift Safety Valves
Reactor Trips with a PORV Actuation -TMI-2 (continued)							
11-3-78	Loss of Feedwater	Hi RCP	Loss of Consensate Booster Pumps	90	No	No	No
- - - - - - - - - - Commercial Operation - - - - - - -							
1-15-79	Instrument Failure	Lo RCP	Atmospheric Relief Bellows Failed	5	No	No	No
3-6-79	Turbine Trip	Power Imbalance	Turbine Trip PORV Actuation not Reported	?			
3-28-79	Loss of Feedwater	Hi RCP	TMI-2 Accident	97	No	?	?

TABLE B.3

REACTOR TRIPS SINCE TMI-2

Date	Transient Classification	Trip Signal	Cause of Transient	Initial Power Level	PORV Lifted	Pzr. Safety Valves Lifted	If Old Setpoints Had Been Used		
							PORV Actuation?	Trip on High Pressure?	Lift Safety Valves
ANO-1									
8-13-79	Turbine Trip	Hi RCP*	Switchyard Relay Failure	75	No	No	Yes	No?	No
7-8-79	Turbine Trip	Antici-patory	Governor Valve Control Failure	75	No	No	Yes	No?	No
4-7-80	Loss of Offsite Power	?	Tornado Downed 500 kv Lines	-	-	-	-	-	-
Crystal River-3									
8-2-79	Loss of Feedwater	Antici-patory	Dirt in MFP "B" Controller. Low Level both OTSGs.	10	-	-	-	-	-
8-16-79	Loss of Feedwater	Hi RCP	FW Upset After RCP Trip	73	No	No	Yes	No	No
8-16-79	Loss of Feedwater	Hi RCP	FW Valve Actuator Failure. OTSG underfed	47	No	No	Yes	No	No
8-17-79	Loss of Feedwater	Hi RCP	FW Going From One to Two-Pump Operations; "A" Pump Speed Lower and "B" Higher Than Required. OTSG underfed	48	No	No	Yes	No	No
8-17-79	Loss of Feedwater	Hi RCP	Operator Went From Manual to Auto With ICS With Off Normal Plant Conditions. OTSG underfed	24	No	No	Yes	No	No

*"Hi RCP" indicates high reactor coolant pressure.

TABLE B.3 (continued)

Date	Transient Classification	Trip Signal	Cause of Transient	Initial Power Level	PORV Lifted	Pzr. Safety Valves Lifted	If Old Setpoints Had Been Used: PORV Actuation?	Trip on High Pressure?	Lift Safety Valves
Crystal River-3 (continued)									
9-15-79	Steam Leak		Welded Cap Failed on 3/4 in. instrument connection on main steam chest						
9-18-79	Loss of Feedwater	Anticipatory Low Level Both OTSG's	FWP Regulator Failed	72	No	No	Yes	No	No
12-21-79	Instrument Failure	Hi RCP	Incorrent ICS input signal raised feedwater flow						
2-26-80	Power Supply Failure	Hi RCP	Lost NNI-X Power Supply	99	Yes	Yes	-	-	

TABLE B.3 (continued)

Date	Transient Classification	Trip Signal	Cause of Transient	Initial Power Level	PORV Lifted	Pzr. Safety Valves Lifted	If Old Setpoints Had Been Used		
							PORV Actuation?	Trip on High Pressure?	Lift Safety Valves
Davis Besse-1									
9-18-79	Turbine Trip	Antici-patory	Perturbation In EHC Fluid Pressure	99.8	No	No	Yes	No	No
9-26-79	Turbine Trip	Hi RCP	Failure of Power Supply For Turbine Throttle Pressure Limiter XMTR	100	No	No	Yes	No	No
10-15-79	Turbine Trip		ICS Pulser to EHC Failed						
10-25-79	Loss of RCP	PWR/ Flow	Blown Fuse in RCP 2-2 DC Power Supply For Starting						
3-27-80	Instrument Failure	Manual	Two groups of Safety Control rods drifted in. Manual rod control attempt unsuccessful	72	No				
4-7-80		Hi flux	Drop in Condenser Vacuum Changed Steam Plant Efficiency						

TABLE B.3 (continued)

Date	Transient Classification	Trip Signal	Cause of Transient	Initial Power Level	PORV Lifted	Pzr. Safety Valves Lifted	If Old Setpoints Had Been Used: PORV Actuation?	Trip on High Pressure?	Lift Safety Valves
Oconee 1									
6-11-79	Loss of Feedwater	Antici-patory	EHC Card Failure	99	No	No	No	No	No
6-11-79	Manual Reactor Trip	Manual	Low OTSG Level	1	No	No	No	No	No
6-17-79	Partial Loss of RCS Flow	PWR/Pumps	Two RC Pumps Tripped	97	No	No	No	No	No
8-6-79	Turbine Trip	Antici-patory	Valving-out of Pressure Switch	40	No	No	No	No	No
7-4-79			LPI Cooler Tube Leak						
7-20-79	Pressure Transient	Hi RCP	Performing RCS Leak Test						
10-8-79	CRD Power Supply		Personnel Error During CRD Power Supply Test						

TABLE B.3 (continued)

Date	Transient Classification	Trip Signal	Cause of Transient	Initial Power Level	PORV Lifted	Pzr. Safety Valves Lifted	If Old Setpoints Had Been Used: PORV Actuation?	If Old Setpoints Had Been Used: Trip on High Pressure?	If Old Setpoints Had Been Used: Lift Safety Valves
Oconee 2									
5-7-79	Loss of Feedwater	Hi RCP	Underfed OTSG 2A	15	No	No	Yes	Yes	No
6-3-79	Loss of Feedwater	Hi RC/	Malfunction of Main FW Block Valve	30	No	No	Yes	Yes	No
7-18-79	Turbine Trip?	Flux/ Flow	Lightning truck Substation Breaker	---	No	No	No	No	No
10-23-79	Turbine Trip	Antici- patory	MSRH Drain Tank Level inst failure						
1-30-80			Error in 230 kV Substation relay testing						

TABLE B.3 (continued)

Date	Transient Classification	Trip Signal	Cause of Transient	Initial Power Level	PORV Lifted	Pzr. Safety Valves Lifted	If Old Setpoints Had Been Used		
							PORV Actuation?	Trip on High Pressure?	Lift Safety Valves
<u>Oconee 3</u>									
10-31-79	Turbine Trip	Antici-patory	High Level in MSRH Drain Tank	?					
11-10-79	Loss of Feedwater	Hi RCP	Hotwell Pumps Tripped, Lost Power to ICS	99	No	No			
3-14-80	Turbine Trip	Antici-patory	Turbine Trip	?					

TABLE B.3 (continued)

Date	Transient Classification	Trip Signal	Cause of Transient	Initial Power Level	PORV Lifted	Pzr. Safety Valves Lifted	If Old Setpoints Had Been Used: PORV Actuation?	Trip on High Pressure?	Lift Safety Valves
Rancho Seco									
4-2[illegible]	Loss of Feedwater	Hi RCP	Loss of "A" Inverter	100	No	No	Yes	Yes	No
7-1-79	Loss of Feedwater	Hi RCP	Test of STP-070	13	No	No	Yes	No	No
7-12-79	Turbine Trip	Antici-patory	Spurious Activity in Overspeed Protection Circuit	100	No	No	No	No	No
9-12-79	Turbine Trip	Antici-patory	Spurious Activity in Overspeed Protection Circuit	100	No	No	No	No	No
9-13-79	Unspecified	PWR/Flow	Imbalance on Restart	30	No	No	No	No	No

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U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

1. REPORT NUMBER (Assigned by DDC)
NUREG-0667

2. (Leave blank)

3. RECIPIENT'S ACCESSION NO.

4. TITLE AND SUBTITLE (Add Volume No., if appropriate)
TRANSIENT RESPONSE OF BABCOCK & WILCOX-DESIGNED REACTORS

5. DATE REPORT COMPLETED
MONTH: April | YEAR: 1980

DATE REPORT ISSUED
MONTH: May | YEAR: 1980

6. (Leave blank)

7. AUTHOR(S)
B&W Reactor Transient Response Task Force
R.L. Tedesco, Chairman

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)
Same as 9 above

13. TYPE OF REPORT

PERIOD COVERED (Inclusive dates)

14. (Leave blank)

15. SUPPLEMENTARY NOTES

16. ABSTRACT (200 words or less)

On February 26, 1980, the Crystal River Unit No. 3 Nuclear Generating Plant, designed by the Babcock & Wilcox Company (B&W), experienced an incident involving a malfunction in an instrumentation and control system power supply. This resulted in a reactor and turbine trip; the opening of the pressurizer power-operated relief valve, spray valve and a Code safety valve; decreased feedwater flow; actuation of the engineered safety features systems; and a discharge of approximately 40,000 gallons of primary coolant into the containment building.

Faced with the Crystal River Unit 3 incident and the apparently high frequency of such near similar types of transients in other B&W-designed plants, a special Task Force (i.e., B&W Reactor Transient Response Task Force) was established within the Office of Nuclear Reactor Regulation to provide an assessment of the apparent sensitivity of the B&W-designed plants to such transients and the consequences of malfunctions and failures of the integrated control system and non-nuclear instrumentation. This report provides an assessment of these issues.

17. KEY WORDS AND DOCUMENT ANALYSIS

17a. DESCRIPTORS

17b. IDENTIFIERS/OPEN-ENDED TERMS

18. AVAILABILITY STATEMENT
Unlimited Availability

19. SECURITY CLASS (This report)
Unclassified

20. SECURITY CLASS (This page)
Unclassified

21. NO. OF PAGES

22. PRICE
$

NRC FORM 335 (7-77)

☆U.S. GOVERNMENT PRINTING OFFICE: 1980 620-269/150 1-3

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

OFFICIAL BUSINESS
PENALTY FOR PRIVATE USE, $300

POSTAGE AND FEES PAID
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