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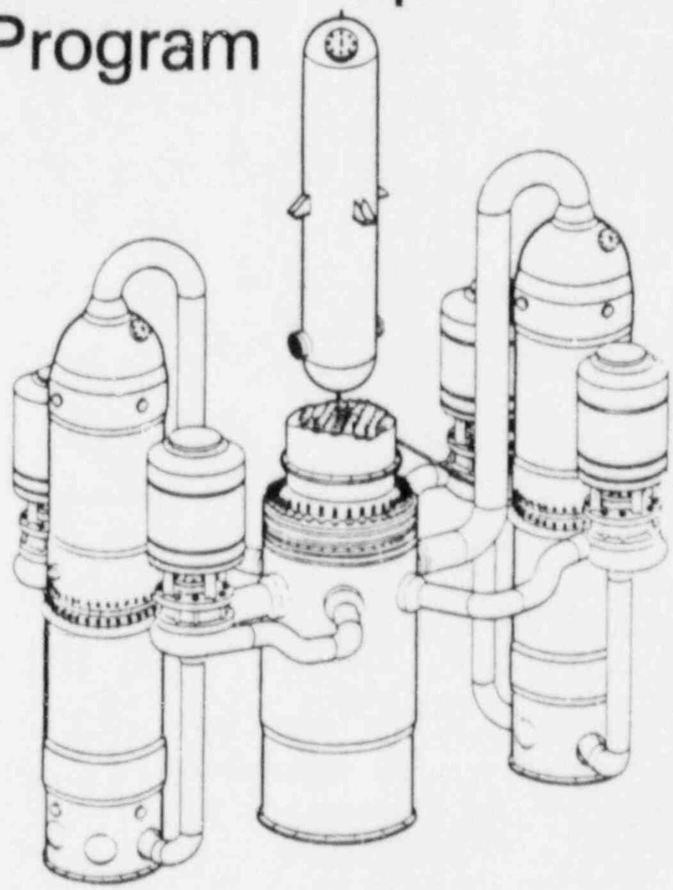
Supplement No. 1

Safety Evaluation Report
related to
Babcock & Wilcox Owners Group
Plant Reassessment Program



U.S. Nuclear Regulatory
Commission

Office of Nuclear Reactor Regulation

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ABSTRACT

Supplement 1 to the "Safety Evaluation Report (SER) Related to the Babcock & Wilcox Owners Group (BWOG's) Plant Reassessment Program" has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission (NRC). This supplement contains the NRC staff's evaluation of the BWOG reassessment of the integrated control system/non-nuclear instrumentation system, the emergency feedwater initiation and control system, reactor trip initiating events, several additional open items identified in the SER, and the BWOG comments on the SER.

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APPENDICES

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1 INTRODUCTION AND BACKGROUND

The Nuclear Regulatory Commission staff (referred to as NRC staff or staff) issued a Safety Evaluation Report (SER) related to the Babcock & Wilcox Owners Group (BWOG) Plant Reassessment Program (NUREG-1231, November 1987). Some of the BWOG submittals to the NRC staff were not received in time for the staff to complete its review of the entire BWOG program before the SER was issued. This supplement to the SER provides the staff's evaluation of the BWOG reassessment of the integrated control system/non-nuclear instrumentation (ICS/NNI) system, the emergency feedwater initiation and control system, reactor trip initiating events, and several additional open items identified in the SER. The status of the staff's evaluation of the BWOG plant reassessment program is provided in Table 1.1. As indicated by the table, all the staff's reviews of the BWOG Safety and Performance Improvement Program (SPIP) projects have been completed.

Each section and appendix of this supplement is numbered and titled so that it corresponds to the section or appendix of the SER that is relevant to the NRC staff's additional evaluation. Except where specifically noted, the material in this supplement does not replace the material in the corresponding SER section or appendix. Appendix A is a continuation of the bibliography, Appendix B is a continuation of the list of abbreviations, and Appendix C is a continuation of the list of staff contributors and consultants, as they relate to this supplement. Appendix E is a list of some of the more significant previously identified concerns involving the ICS/NNI systems, and Appendix F references NRC staff letters of interest to the ICS/NNI reassessment. Appendix G contains errata for the SER.

Table 1.1 Status of BWOG SPIP and NRC staff reviews

SPIP project	SER, SSER section	Status
Sensitivity study	SER 5	The BWOG study was submitted in April 1987. The NRC contractor and NRC staff review is complete.
Operating experience review	SER 4.2, 4.3, 4.5, 7.3	The BWOG evaluation of the Transient Assessment Program data is complete. The NRC contractor and NRC staff review is complete.
Reactor trip initiating events review	SSER 6.1, 10	The BWOG report was submitted October 1987. The NRC staff review is complete.
Risk assessment	SER 9 SSER 9.6	The BWOG evaluation is complete. The NRC staff utilized a contractor to perform an independent risk evaluation. The NRC contractor and NRC staff review is complete.
Integrated control system/ non-nuclear instrumentation system review	SSER 6.1	The BWOG evaluation is complete and was submitted in Revision 5 to BAW-1919 dated July 1987. The NRC contractor and NRC staff review is complete.
Main feedwater system review	SER 6.2	The BWOG evaluation is complete. The NRC staff review is complete.
Emergency feedwater/ auxiliary feedwater system review	SER 6.3 SSER 6.3.3	The NRC staff review was completed in the SER, except for the emergency feedwater initiation and control system review, Section 6.3 3. The NRC contractor and NRC staff review of Section 6 3.3 is complete.
Instrument air system review	SER 6.5	The BWOG evaluation is complete. The NRC staff review is complete.
Main steam pressure control review	SER 6.4	The BWOG evaluation is complete. The NRC staff review is complete.
Operations/maintenance personnel interviews	SER 4.4, 7.2	The BWOG evaluation is complete and was submitted in March 1987. The NRC contractor and the NRC staff review is complete.

Table 1.1 (continued)

SPIP project	SER, SSER section	Status
Emergency procedures review	SER 7.4	The BWOG evaluation has not been submitted and will not be completed as part of the SPIP. The BWOG estimates this effort will take until the summer of 1988. The NRC staff will review these procedures separate from the SPIP.
Operator burden evaluation	SER 7.5	The BWOG evaluation is complete and was submitted in May 1987. The NRC contractor and the NRC staff review is complete.

2 OVERVIEW OF B&W PLANT REASSESSMENT PROGRAM

2.2 Description of the NRC Staff Activities

As noted in the SER, an NRC staff member identified nine concerns related to the BWOG reassessment program. Four of these concerns were addressed in the SER and the remaining five were deferred to this supplement. These concerns are listed below as they appeared in the SER.

(2) There is a potential lack of independence in Science Applications International Corporation performing similar work as a subcontractor for the NRC and for B&W plant owners.

(3) A premature finding by the BWOG of the adequacy of the ICS/NNI.

(4) The BWOG or utilities have not analyzed effectively the proposed SPIP recommendations to determine the effects on the other parts of the plant.

(5) There is a potential for catastrophic reactor pressure vessel failure resulting from an overcooling transient assuming one control rod stuck out and a return to criticality.

(7) The B&W plants violate offsite dose limits under the conditions of a steam generator tube rupture design-basis accident.

Concern 2 is related to a potential lack of independence of the work performed by Science Applications International Corporation (SAIC) because they were negotiating a contract with a utility on a similar program while working as a subcontractor for Oak Ridge National Laboratory (ORNL) on an NRC contract with the Office of Nuclear Regulatory Research. The NRC Office of General Counsel has reviewed this matter and is of the opinion that, assuming the work being performed for the NRC is essentially completed, there is no logical basis for attempting to restrict contractors from performing similar work for others. With regard to any technical concerns related to the specific ICS/NNI system's failure modes and effects analyses, which are contained in Appendix R of BAW-1919, Revision 5, the NRC staff has performed a detailed independent review of this material and its evaluation is contained in Section 6.1.5 of this supplement. The staff, therefore, concludes that this concern does not affect the results of the BWOG SPIP or the staff's review.

The staff has addressed concerns 3, 4, and 5 in Sections 6.1.1, 11.5, and 5.5.3 of this supplement, respectively.

The NRC staff is evaluating the risk associated with steam generator tube events as part of unresolved safety issues A-3, A-4, and A-5 (NUREG-0844). As a result of these evaluations, the staff has identified several issues for further study that have been designated as generic issue 67 (GI-67), "Steam Generator Staff Actions." The staff member concern 7 is being addressed by GI 67.5.1, "Reassessment of Radiological Consequences," and GI 67.5.2, "Reevaluation of SGTR Design Basis," and thus will not be addressed in this report.

3 SUMMARY AND CONCLUSIONS

The summary and conclusions in the SER were primarily based on the results of the NRC staff's review of these areas within the BWOG SPIP that were addressed in the SER. As noted in the SER, there were three major areas remaining to be reviewed, the integrated control system/non-nuclear instrumentation system (ICS/NNI), the reactor trip initiating events, and the emergency feedwater initiation and control system for each B&W plant.

In addition to these areas, the staff has reviewed the results of the BWOG Safety and Performance Recommendation Integration Group (SPRIG), the BWOG plans for reviewing the implementation of the SPIP recommendations, and the BWOG comments (letter, December 21, 1987) on the initial SER. The staff also has evaluated the effect of the SPIP recommendations on specific safety issues and overall B&W plant risk.

As noted in the SER, the staff concluded that SPIP was broad-based and comprehensive. This conclusion was based, in part, on the continuous input provided by the staff during the development of the SPIP program. However, since the staff completed its review, it has become apparent that in several areas (such as human factors issues and the ICS/NNI review) the BWOG did not incorporate the staff's suggestions in the SPIP. The staff continues to believe that SPIP was a comprehensive program and will result in improved safety at B&W plants by reducing the number and complexity of plant trips and transients. Nevertheless, it would have been more comprehensive if the BWOG had addressed all of the staff's issues. The staff recognizes that even with implementation of all the recommendations proposed by the BWOG, future transients cannot be totally eliminated and will occur at B&W plants. Because the program has not fully resolved the staff's concerns, the staff believes it more probable that these future transients will result from, or be complicated by, the same systems studied in SPIP. The staff encourages the BWOG to continue to take actions to reduce reactor trips and complex transients at B&W plants and additionally suggests that the NRC staff concerns be considered as a part of these continuing efforts.

As noted in the SER, the risk studies performed by the BWOG and the staff were examined to identify areas that contribute significant fractions of core-damage frequency. From an examination of the SPIP recommendations, it also was concluded that the BWOG studies were responsive to these areas. The staff has further evaluated the SPIP recommendations versus the results of the risk assessment studies and concludes that, with proper implementation of the recommendations, the Category C events will be reduced to insignificant contributors to core-damage risk for the B&W plants.

In the SER the staff found that the BWOG efforts in the area of human factors concerns were deficient primarily because human factors engineering expertise were not applied in these efforts. As a result of this finding, the staff suggested that the BWOG re-examine the information gathered from its efforts to provide further assurance that all significant human factors issues have been properly addressed. The BWOG has concluded that another review of their efforts is unnecessary. The staff has reviewed the BWOG position; although no specific regulatory action to require the BWOG to re-review its previous efforts is required, the staff continues to believe that additional recommendations to improve operator performance could be developed from another review of the previous BWOG efforts.

As a result of the specific reviews described in this supplement, the staff finds that the additional recommendations developed as a result of the BWOG review of reactor trip initiating events will further enhance the BWOG efforts to reduce the frequency of reactor trips.

The staff also concludes that, while differences exist in the designs of the emergency feedwater initiating and control systems at the B&W plants, all these systems meet safety grade requirements and satisfy appropriate regulatory requirements. In addition, the staff concludes that all the emergency feedwater systems do not rely on the non-safety-grade ICS to fulfill their safety function.

With respect to the BWOG review of the ICS/NNI systems, the staff concludes that the overall approach was appropriate for achieving resolution of ICS/NNI concerns. The staff considers the development of upgraded/revised ICS/NNI systems requirements documents to be a significant step in the overall effort. If existing ICS/NNI designs are modified to achieve conformance with the systems requirements, the staff believes that many of the ICS/NNI problems previously identified by the staff from operating experience at B&W plants will be resolved.

As a result of its efforts, the BWOG has identified numerous recommendations to reduce the contribution of the ICS/NNI systems to reactor trips and complex post-trip response. In general, the staff agrees with the BWOG recommendations. However, the staff has identified some recommendations for which it either disagrees with the BWOG, has reservations concerning the implementation, or offers additional guidance to the BWOG. These recommendations are discussed in detail in Section 6.1 of this supplement.

Of special note from the BWOG ICS/NNI review is the BWOG recommendation that B&W plants should go to a known safe state on a loss of ICS/NNI power. Proper implementation of this recommendation is expected to eliminate transients similar to those that occurred at Rancho Seco (December 26, 1985) and Crystal River (February 26, 1980). The staff has provided further guidance on this recommendation, most notably aimed at limiting operator response to loss of ICS/NNI events.

In summary, the staff concludes that proper implementation of the BWOG ICS/NNI recommendations will result in enhanced safety margins at B&W plants by reducing the contribution of the ICS/NNI systems to reactor trips and complex post-trip response. The staff also concludes that the systems requirements do not fully resolve all previously identified concerns (PICs) with the ICS/NNI. However, the staff believes that substantial progress has been made toward the resolution of many of these PICs. As discussed in Section 6.1, the staff believes that with proper implementation of the recommendations, along with verification of proper implementation of plant modifications made in response to concerns identified in IE Bulletin 79-27, the B&W plant response to ICS/NNI failures will be significantly enhanced and will ensure that adequate safety margins exist for B&W plants for the long term.

The staff further notes that the BWOG has an ongoing project to evaluate replacement of the ICS/NNI with an advanced control system based on current technology. The staff encourages the BWOG to give high priority to this study. The staff further encourages the BWOG to ensure that all the staff's PICs with the ICS/NNI will be resolved by the advanced control system.

In the SER, the staff recommended that the BWOG continue to submit the yearly Operating Experience Report developed under the Transient Assessment Program (TAP) in order to allow the staff to maintain an overview of the effectiveness of the recommendations in improving the performance of the B&W plants. The BWOG has committed to provide this information. Additionally, the BWOG has committed to provide the staff with periodic updates to the recommendation tracking system (RTS) to keep the staff informed of progress made in implementing the recommendations. As noted in this supplement, the staff believes that enhancements to the RTS may be needed to make this information more useful and will initiate discussions with the BWOG to pursue this further.

The staff has reviewed the efforts of the Safety and Performance Recommendation Integration Group (SPRIG) to identify key recommendations and finds these efforts and the results to be acceptable. The staff has identified additional recommendations that it believes to be "high-priority."

Finally, the staff reiterates its position in the SER that to ensure improved performance of the B&W plants and acceptable safety margins in the long term, aggressive implementation of the SPIP recommendations is needed by the utilities. As discussed in Section 12.4 of this supplement, in the past 6 months little overall program progress by the utilities has been made in implementing and closing recommendations currently in the RTS. To ensure that the recommendations are aggressively pursued, the staff recommended in the SER that each B&W plant owner submit its plan and schedule for implementing the SPIP recommendations. The BWOG has noted that most of the recommendations call for further evaluation and thus it may not be possible to develop plans by that time. In recognition of the BWOG comments, the staff now recommends that each utility submits by June 1, 1988, its plans for those recommendations that have been evaluated, its schedule for evaluating the remaining recommendations, and a tentative implementation schedule for implementing the remaining recommendations. Additionally, the staff believes this information should be periodically updated as further progress is made in evaluating the recommendations. The staff will use this information to monitor the utilities' progress and to develop appropriate inspection schedules to ensure that the recommendations have been properly implemented.

4 INFORMATION GATHERING

4.5 Staff Activities

4.5.4 Status of Utility Compliance With NRC Actions

In Section 4.6 of the SER, the NRC staff stated that it would address the generic status of the B&W utilities' compliance with NRC actions developed as a result of previous B&W operating events. To obtain this information, the staff performed computer searches using the B&W docket numbers and key words from specific subjects addressed in the BWOG "Safety and Performance Improvement Program" (SPIP) report, BAW-1919, to identify the documents the staff believes should have been reviewed during the SPIP review. Large, stand-alone program efforts, such as the TMI action plan items, anticipated transients without scram (ATWS), fire protection, and equipment qualification, are not included because the staff believes they are being adequately covered as separate actions. However the status of NUREG-0737 action items, including supplemental items, for B&W plants were addressed in Section 4.5.4 of the SER. The integrated control system/non-nuclear instrumentation (ICS/NNI) system review included an independent search that resulted in a table specifically related to that aspect. This search was used as a basis for the staff's review presented in Section 6.1 of this supplement.

This search resulted in the identification of three additional generic documents that were developed as a result of previous B&W operating events. Two documents, Inspection and Enforcement Bulletin (IEB) 80-12 and an associated generic letter dated June 11, 1980, were issued after an event at Davis-Besse on April 19, 1980, and were related to decay heat removal system operability. All utilities have completed their actions with respect to this issue. The third document, Generic Letter 85-13, transmitted NUREG-1154, which is a report on the Davis-Besse loss of main and auxiliary feedwater incident of June 9, 1985. The issuance of NUREG-1154 prompted the BWOG to form the 1154 Task Force to review the Davis-Besse event of June 9, 1985. The results of the BWOG review, reported in Appendix H to BAW-1919, Revision 5, were evaluated by the staff and reported in the SER. On the basis of its review, the staff concludes that the B&W utilities have satisfactorily complied with the previous staff actions developed as a result of previous B&W operating events.

5 SENSITIVITY EVALUATION

5.5 Other B&W Plant Sensitivity Issues

5.5.3 Pressurized Thermal Shock

An NRC staff member raised concern, identified as Concern 5 in Section 2.2 of the SER, that for some overcooling transients, assuming a control rod cannot be inserted (stuck out), there is a potential for the reactor to return critical, which could possibly lead to a catastrophic failure of the reactor vessel. Specifically, this is a pressurized thermal shock scenario wherein the reactor coolant system (RCS) has overcooled, the pressurizer has completely filled with water as a result of the emergency core cooling systems (ECCS) activating, the RCS has pressurized, and the reactor has returned critical because insufficient shutdown margin existed as a result of the assumption of a stuck control rod. The subsequent pressurization resulting from the reactor returning critical raises the potential for catastrophic failure of the reactor pressure vessel.

To address this concern, the staff requested the BWOG to estimate the temperature at which the reactor will return critical, assuming the most reactive control rod could not be inserted in the core. The BWOG responded by letter dated October 16, 1987. It stated that by assuming end-of-life core conditions for a generic B&W plant with all rods inserted in the core, RCS temperature will need to be reduced to less than 35°F for criticality to occur. Assuming the most reactive control rod stuck out of the core, RCS temperature will need to be reduced to approximately 310°F before the reactor will return critical. The BWOG also stated that ECCS actuation, which results in boron addition to the RCS, will occur to compensate for the liquid contraction caused by the overcooling. Accounting for the negative reactivity of the boron, it was estimated that the RCS temperature again will have to be reduced to less than 35°F before the reactor will return critical.

The NRC staff finds the BWOG estimates of RCS temperatures required for the reactor to return critical to be reasonable. In light of these low RCS temperatures, the staff concludes that an overcooling event in a B&W plant will not result in the reactor returning critical, even if the most reactive control could not be inserted in the core.

6 SYSTEMS REVIEW

6.1 Integrated Control System/Non-Nuclear Instrumentation System Review

6.1.1 Introduction

The NRC staff stated in the SER that it had not yet completed its review of the BWOG integrated control system/non-nuclear instrumentation system (ICS/NNI) evaluation report (Appendix R, BAW-1919). This supplement provides a summary report of the staff's review and supersedes the information presented in the SER.

On December 26, 1985, an event occurred at the B&W-designed Rancho Seco nuclear plant that involved a loss of integrated control system (ICS) ±24-volt dc power. The ICS is not a safety-related system. The loss of ICS dc power caused the valves in the main feedwater (MFW) and main steam systems to automatically reposition and caused the loss of remote control of the affected valves from the control room. In addition, the MFW pump turbines slowed to minimum speed and the auxiliary feedwater (AFW) system pumps automatically started. The initial effect was an undercooling of the reactor coolant system (RCS) that led to a reactor trip on RCS high pressure. The reactor trip was followed by a safety features actuation system (SFAS) automatic initiation of the high-pressure injection (HPI) system and excessive overcooling of the RCS.

Investigation of this event at Rancho Seco revealed the same concerns related to equipment response to loss of ICS or non-nuclear instrumentation (NNI) power that had been previously identified from earlier events at B&W reactors. A number of the concerns were documented in IE Bulletin 79-27, "Loss of Non-Class 1E Instrumentation and Control Power System Bus During Operation," dated November 30, 1979, and in NUREG-0667, "Transient Response of Babcock & Wilcox-Designed Reactors," published in May 1980. The concerns include undesirable failure modes of plant equipment, mid-scale failures of control room indications that can potentially mislead operators, and the absence of a procedure for operator actions upon the loss of ICS power. The staff had intended that the concerns associated with loss of ICS/NNI power be resolved through licensee reviews and responses to IE Bulletin 79-27. However, as demonstrated by this event, the implementation of modifications by the B&W utilities to resolve the concerns identified in IE Bulletin 79-27 were not adequate.

The objective of the BWOG Safety Performance Improvement Program (SPIP) reassessment of the ICS/NNI design was to perform a comprehensive review to develop recommended improvements to limit the consequences of failures, and thus, reduce the ICS/NNI contribution to the frequency of reactor trips and the complexity of transients. The ICS/NNI reassessment and related activities of the SPIP were performed by the BWOG Instrumentation and Control (I&C) Committee. The BWOG reassessment of ICS/NNI designs is provided in Appendix R, "ICS/NNI Evaluation Final Report," to BAW-1919.

This section of the supplement provides the results of the staff's technical review and evaluation of the BWOG ICS/NNI reassessment. The staff's review focused on whether the recommendations developed by the BWOG from the reassessment would be sufficient to resolve previously identified staff concerns regarding the ICS/NNI, such as those identified in IE Bulletin 79-27 and NUREG-0667. A list of some of the more significant previously identified concerns is provided in Appendix E to this supplement. The staff believes that resolution of the concerns listed in Appendix E would significantly reduce the complexity of transients at B&W plants.

The staff met periodically with the BWOG I&C Committee during development and execution of the ICS/NNI reassessment program. On the basis of the information exchanged at the BWOG/NRC meetings and the review of information provided in earlier versions of BAW-1919, the staff, in a number of letters to the BWOG, documented its concerns and provided comments on program direction. In particular, these letters (1) documented specific ICS/NNI-related issues that the BWOG was requested to address and resolve as part of the reassessment, (2) provided requests for additional information (to be documented in BAW-1919) necessary for the staff to evaluate whether previously identified ICS/NNI concerns were being adequately addressed, and (3) provided general feedback on the scope and direction of the reassessment program. The letters of interest to the ICS/NNI reassessment are referenced in Appendix F to this supplement.

Information provided in BAW-1919 Appendix I, "Review of Category 'B' and Category 'C' Events at the BWOG Plants 1980-1985," shows that Category C and significant Category B transients at B&W-designed reactors have resulted from the failure to balance heat removal with heat production under post-trip conditions. Eight of the ten Category C events and twenty-six of the thirty-six significant Category B events have resulted from excessive primary-to-secondary heat transfer. One Category C event and three significant Category B events involved inadequate heat transfer. Therefore, the BWOG concluded that improved manual and automatic capability for controlling steam flow (steam generator pressure) and feedwater flow (steam generator inventory), and thus steam generator heat transfer, from the control room under post-trip conditions is the key to reducing the frequency and severity of Category C and Category B transients. The ICS/NNI systems were selected for review because of their role in controlling the MFW system and secondary side steam pressure relief (turbine bypass valves and atmospheric dump/vent valves) and because of their contribution to reactor trips and their involvement in the complexity of pre-trip and post-trip transient behavior.

Figure 6.1 is a block diagram that identifies the major elements of the BWOG I&C Committee's reassessment of the ICS/NNI. The BWOG I&C Committee first reviewed the operating experience (including information provided in NUREGs, IE bulletins, B&W reports, ongoing BWOG studies, licensee event reports (LERs), and other sources), published reports concerning the ICS/NNI (e.g., BAW-1564, "Integrated Control System Reliability Analysis"), and the original design requirements and plant operating philosophy for the ICS/NNI systems to develop a set of current, updated system design and functional requirements for the ICS/NNI. The existing as-built, plant-specific ICS/NNI designs were then compared to the current requirements for the ICS/NNI systems. From this comparison, a list of ICS/NNI problems and potential recommended solutions was developed. The I&C Committee then evaluated the recommended solutions against

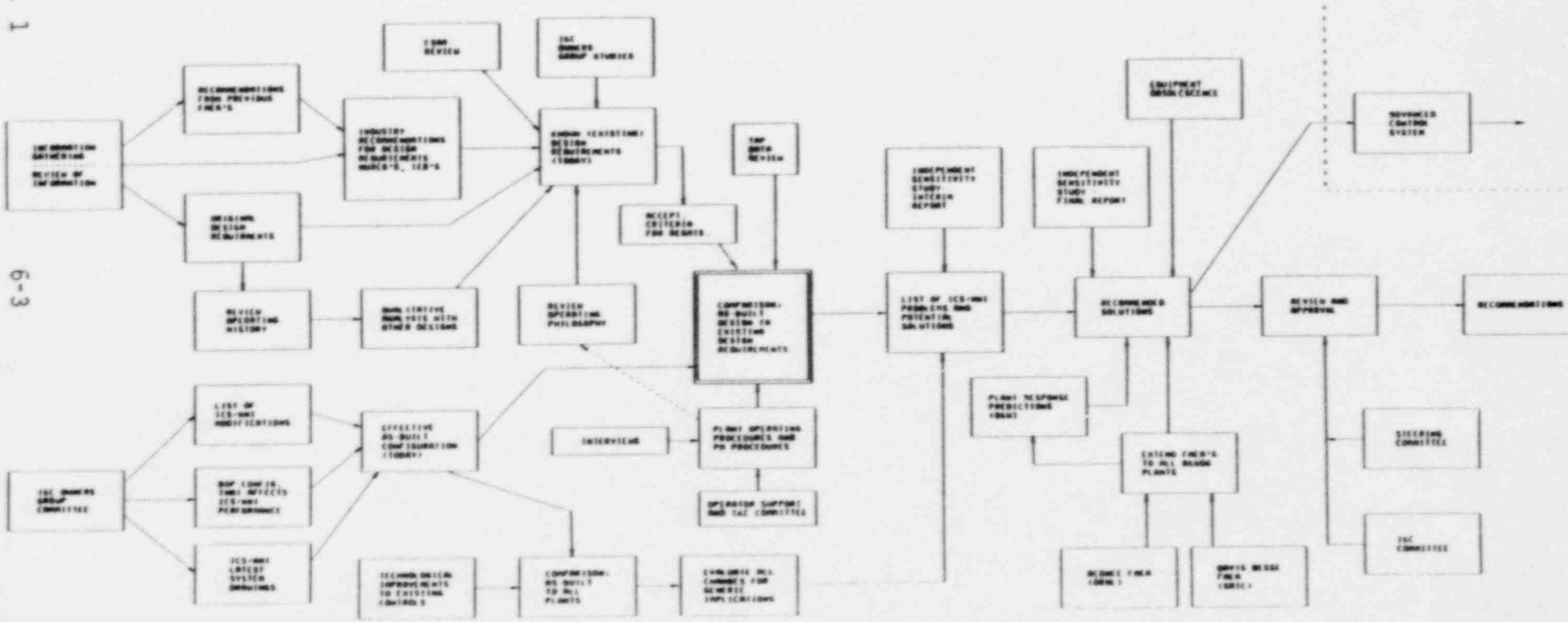

Figure 6.1 Major elements of the BWOG I&C Committee's reassessment of the ICS/NNI

the results of the Sensitivity Study performed by MPR Associates (MPR) (discussed in Section 5.1 of the SER), the failure modes and effects analysis (FMEA) performed on the ICS/NNI, the review of the BWOG Transient Assessment Program (TAP) data, and interviews with plant personnel concerning operating procedures and maintenance programs. On the basis of this evaluation the I&C Committee developed a final list of potential recommended solutions to forward to the SPIP Steering Committee for review. The recommendations approved by the Steering Committee were issued as final Level I recommendations (i.e., recommendations that will provide immediate improvements to the operation, availability, and reliability of the ICS/NNI) and were included in the SPIP recommendation tracking system (RTS) for implementation at BWOG plants.

In Section 2.2 of the SER, an NRC staff member raised concerns related to the reassessment program. Concern 3 stated that the BWOG had reached a premature finding with regard to the adequacy of the ICS/NNI. Specifically, the BWOG assumed the ICS/NNI systems to be adequate; thus the ICS/NNI reassessment did not consider replacement of the existing ICS/NNI with an advanced design. The staff believes that the existing ICS/NNI design can be modified to resolve the concerns listed in Appendix E to this supplement and, therefore, finds the BWOG approach acceptable. In addition, the staff notes that the BWOG Steering Committee directed the I&C Committee to evaluate replacement of the ICS/NNI with an advanced system. The BWOG took the approach to make immediate improvements in the existing ICS/NNI design while replacement of the ICS/NNI design was being studied under the equipment obsolescence program.

By letter dated October 7, 1987, the BWOG provided an updated RTS report that lists the most recent approved recommendations. There are 49 approved recommendations related to the ICS/NNI. Approved recommendations are identified by a "TR" prefix to the recommendation number. Revision 5 of BAW-1919 (dated July 1987) lists 31 pending Level I recommendations that will be entered in the RTS if approved by the Steering Committee. The pending Level I recommendations are listed in Section VII.B.2 of Appendix R to BAW-1919 and have been designated by the staff as recommendations B.2.1 through B.2.31 for reference purposes. Section VII.C of Appendix R lists 10 Level II recommendations; that is, recommendations potentially involving major modifications to existing equipment and requiring further in-depth evaluation before BWOG approval. These recommendations have been referenced by the staff as C.1 through C.10. Section VII.D of Appendix R lists six Level III recommendations; that is, recommendations involving replacement of the existing ICS/NNI system with a new system based on modern digital control technology. These recommendations have been referenced by the staff as D.1 through D.6. Section VII.E of Appendix R identified 26 additional potential recommendations undergoing additional review by the BWOG I&C Committee; these have been referenced by the staff as E.1 through E.26. The staff also has reviewed the ICS/NNI-related recommendations that were rejected by the BWOG Steering Committee. These rejected recommendations are identified by an "RR" prefix to the recommendation number. In addition, the staff reviewed ICS/NNI-related recommendations made by the BWOG Transient Assessment Committee in its report, "Review of Reactor Trip Initiating Events at the BWOG Plants 1980-1986"; these recommendations are identified by a "TAC" prefix to the recommendation number.

The sections below provide the staff's evaluation of all ICS/NNI-related recommendations. The staff discusses those areas of concern and additional considerations or actions that it believes are desirable before implementation of specific recommendations and those recommendations that it believes should not be implemented. The recommendations are evaluated on the basis of their potential contribution toward (1) reducing ICS/NNI involvement in reactor trip initiation and in transient complexity and (2) in achieving resolution of previously identified concerns that have been common to the more severe B&W plant transients involving ICS/NNI failures.

Because of the large number of ICS/NNI-related concerns and recommendations, the complexity of the ICS/NNI systems, and the large scope of the BWOG ICS/NNI reassessment (which involved failure modes and effects analyses, an independent sensitivity study, and numerous other subprograms), the evaluation provided below is a summary/overview of the staff's conclusions related to the BWOG ICS/NNI reassessment. A staff report discussing the more detailed technical aspects of the ICS/NNI reassessment program and recommendations is being prepared and will be issued separately.

6.1.2 Requirements for the ICS/NNI Systems

Requirements of the ICS/NNI systems (collectively referred to as system requirements) were developed as part of the ICS/NNI reassessment. The system requirements include functional requirements, hardware design requirements, and programmatic requirements (e.g., preventive maintenance and qualification and training of personnel). The BWOG I&C Committee developed recommendations for ICS/NNI improvement by comparing actual installed plant-specific designs with the ICS/NNI systems requirements. The staff reviewed the ICS/NNI systems requirements to determine whether they were sufficient to produce recommendations that, if implemented, would prevent or reduce the consequences of B&W plant transients such as occurred at Rancho Seco on December 26, 1985.

In general, the system requirements established the bases that the designers would have used in the original ICS/NNI systems had they had the benefit of the actual operating experience with regard to ICS/NNI failure modes and their effect on overall plant response. As mentioned earlier, the current requirements for the ICS/NNI systems were developed after a review of operating experience, the original design requirements for the ICS/NNI systems, and current plant operating philosophy. The contents of the system requirements documents for both the ICS and NNI are similar, each with sections addressing the following topics: environmental requirements, power supply design, loss and restoration of power, signal input reliability, pneumatic design, instrumentation accuracies, interfaces with other systems, general/functional requirements, instrumentation calibration, training/qualification, system performance, and requirements for maintaining up-to-date system documentation (e.g., schematic diagrams and instruction books).

From the staff's review of the ICS/NNI system requirements, it appears that the BWOG ICS/NNI design philosophy is (1) to improve system reliability by minimizing the effects of single instrument channel component failures (e.g., transmitter failures) and (2) to limit the consequences of ICS/NNI power losses by ensuring that equipment failure positions and plant response are such that the plant will assume a "known safe state." The BWOG has defined "known safe state" as maintaining the reactor coolant system pressure/temperature relationship within the bounds of the abnormal transient operational guidelines (ATOG) normal post-trip pressure-temperature window and allowing credit for normal operator actions from the control room upon loss of ICS/NNI power. Other system requirements that affect ICS/NNI hardware design are intended to improve the operator's capability to respond to loss of ICS/NNI power events, and to prevent plant transients when power is restored.

The staff's review of the ICS/NNI system requirements indicates that many of the problem areas identified from the investigations of B&W reactor transients involving the ICS/NNI would be resolved if existing plant ICS/NNI designs were modified to achieve conformance with the system requirements. However, the staff also believes that a number of areas need to be further addressed in the documents related to ICS/NNI system requirements. The staff views such documents as "living documents" that will be retained by the BWOG for reference with regard to the design and operation of the ICS/NNI systems. The staff recommends that the BWOG consider developing ICS/NNI design-basis documents that explain the basis for the individual requirements for the ICS/NNI systems and provide justification for values established by the system requirements. These documents should be retained as living documents to support after documents related to ICS/NNI system requirements. In the staff's opinion, the design-basis documents will significantly enhance any explanation regarding the basis for the existing ICS/NNI designs and will ensure proper development and implementation of any future ICS/NNI system modifications.

6.1.3 ICS Input Signal Failures

The results of the staff's technical review and evaluation of the BWOG recommendations related to improving ICS/NNI response to input failures are given below. Specifically included is the staff's review of BAW-1919, Appendix D, "Improvement of ICS Response to Input Failures." The following recommendations were reviewed:

Number	Recommendation
TR-001-ICS*	Replace RC flow signal input to ICS with RC pump status.
TR-003-ICS	Remove startup FW flow correction to main FW flow function from the ICS.
TR-005-ICS*	Remove neutron flux signal auctioneering circuitry from RPS and relocate in the ICS.
TR-006-ICS	Delete FW temperature correction to FW demand from ICS.
TR-104-ICS*	Incorporate automatic selection of valid inputs for ICS/NNI. (Note: This recommendation supersedes TR-002-ICS and TR-004-ICS.)

*The BWOG has identified these recommendations as key recommendations.

Number	Recommendation
TR-154-ICS*	Provide operator with unambiguous status of indicators and recorders in main control room on loss of ICS/NNI power or signal.
TR-189-ICS	Set selector switches to select maximum NNI-X dependence.
TR-196-ICS	Set pressurizer level signal select relays to automatic powered transmitters.
TR-198-ICS	Automatic powered reactor inlet and outlet temperature sensors should be selected, or logic changed to automatically select auto powered sensor on loss of hand power.
B.2.8	Spurious alarms should be suppressed.
B.2.10.4	All other signal-select hand stations should normally be set to select auto powered sensors.
B.2.12	"Pure" NNI-X and NNI-Y indicator and recorder channels should be developed.
B.2.13	Auctioneering circuit reliability and failure modes should be considered for improved input signal selection.
B.2.14	Maximize dependence of hand selectable ICS/NNI input signals on only one NNI ac power source.
C.8	Eliminate mid-scale failures that can affect indication or plant control.
E.15	Develop NNI-Y T_{cold} and T_{ave} signal.
TAC-3.d*	Expand TR-104-ICS to include all ICS input signals.

The staff agrees with and encourages implementation of the above recommendations with the exception of recommendations B.2.8, TR-189-ICS, and B.2.14; the staff has reservations about recommendations TR-001-ICS and TR-006-ICS.

The staff believes that recommendation B.2.8 (which involves changing alarm logic from deenergize-to-actuate to energize-to-actuate) is not appropriate in all cases; for example, failures of signal monitors and associated relays would no longer result in an alarm alerting the plant operators to the failure. Reducing nuisance alarms is a good objective; however, the advantages and disadvantages of this recommendation should be further evaluated for each alarm. Recommendation B.2.8 is discussed further in Section 6.1.6. The staff does not believe sufficient justification has been provided to support recommendations TR-189-ICS or B.2.14, which maximize dependence on NNI-X versus NNI-Y, thus potentially increasing the complexity of a loss of NNI-X power transient in order to make a loss of NNI-Y power transient more tolerable.

*The BWOG has identified these recommendations as key recommendations.

The staff has reservations concerning recommendations TR-001-ICS and TR-006-ICS. The staff agrees with the intent of TR-001-ICS, which is to reduce the number of reactor trips resulting from RC flow signal failures. However, the staff is concerned about the replacement of the flow signal with an "equivalent" signal based on RC pump breaker status. It is not clear that ICS performance with the equivalent signal will be acceptable for losses of RC flow that are not caused by RC pump trips. Another concern is that one of the proposed designs to provide an equivalent RC flow signal involves replacement of the equivalent signal with the "less reliable" actual flow signal during transient conditions. It also appears to the staff that the ICS unit load demand (ULD) runbacks on RC pump count and RC flow are not totally redundant; in which case the RC flow runback should not be deleted. The staff believes that hardening of the RC flow signal may be preferable to replacing it with an equivalent. BAW-1919 does not provide adequate justification for deleting the FW temperature correction circuit as recommended in TR-006-ICS.

The staff believes that when the startup feedwater flow signal is removed from the ICS (recommendation TR-003-ICS), the startup feedwater flow indication in the main control room should be retained for use by the operators. When the neutron flux signal auctioneering circuitry is relocated from the RPS to the ICS (recommendation TR-005-ICS), the BWOG should verify that the modified designs conform to the requirements--concerning control and protection system interaction--of Section 4.7 of Institute of Electrical and Electronics Engineers (IEEE) Standard 279, "Criteria for Protection Systems for Nuclear Power Generating Stations." The staff agrees with the concept of the smart analog signal selector (SASS) units (recommendation TR-104-ICS) and concludes that alarms should be provided to alert the operators to SASS operation. The information provided in BAW-1919 does not make it clear as to whether such alarms are provided.

The staff considers recommendations TR-154-ICS and C.8 to be key recommendations because of their importance with regard to resolving previously identified ICS/NNI concerns and strongly encourages their implementation.

6.1.4 ICS Design Features

The results of the staff's technical review and evaluation of the BWOG recommendations related to specific ICS design features are given below. Included is the staff's review of BAW-1919, Appendix C, "Final Report on Re-evaluation of ICS Design Features." The following recommendations were reviewed:

Number	Recommendation
TR-007-ICS*	Remove British thermal unit (Btu) limits from ICS.
TR-008-ICS*	Improvements to reactor runback capability.
TR-009-ICS*	Improvements in ICS tune control circuits.
TR-010-ICS	ICS control circuit modification.

*The BWOG has identified these recommendations as key recommendations.

Number	Recommendation
TR-011-ICS	Determine if the grid frequency error circuit has been detuned.
TR-106-ICS	Remove unused hardware from ICS/NNI cabinets.
TR-190-ICS	Develop backup controls for pressurizer level and pressure control.
TR-193-ICS	Review/test pressurizer heater low-low level interlock logic.
TR-194-ICS	Buffer hand-powered indicators and recorder inputs from automatic power signals.
TR-199-ICS	Failure of inputs to RC pump interlocks must not prevent pump restart.
TR-204-ICS	Eliminate or reduce automatic ICS runback rate on asymmetric rod conditions.
TR-209-ICS	Add signal limiters to prevent control integrals from going into saturation.
C.1	Provide independent controls for reactor power, steam demand, and feedwater such that no single failure would lead to simultaneous and spurious manipulation of control rods, turbine steam admission valves, and feedwater flow.
C.2	Separate control of feedwater pump speed from feedwater regulating valve (except CR & ANO-1).
C.3	Separate control of each feedwater pump speed control from each other.
C.4	Separate control of each feedwater regulating valve from each other.
C.5	Elimination of "Delta Tc" controller function in the ICS.
C.6	Control of feed pump speed based on pumps discharge pressure rather than feedwater valve delta P.
C.7	Provide separate subsystems for reactor coolant temperature and steam pressure control.
E.14	Remove automatic control functions for the atmospheric dump valves from the ICS/NNI system.
E.22	Provide redundant, 2 out of 2 logic, PORV control, separate from NNI. Evaluate leaving the PORV block valve normally closed.

The staff agrees with and encourages implementation of the above recommendations with the exception of recommendations TR-007-ICS and C.5. The staff has reservations about recommendations TR-204-ICS, E.22, TR-010-ICS, and C.4.

Recommendation TR-007-ICS involves removal of the Btu limits function from the ICS and the addition of a means to run back main feedwater flow following a reactor trip during situations where the ICS cross-limits function is rendered inactive. The Btu limits alarm function would remain active to alert the operators of the need for manual action in the event of an overfeed/overfill condition. Originally it was thought that removal of the Btu limits would preclude reactor trips caused by the failure of input signals to the Btu limits circuits and improve the overall reliability of the ICS by reducing the number of ICS modules, thus simplifying the ICS. However, because of the hardening of the T_{hot} input signal to the Btu limits that would occur with recommendation TR-104-ICS (discussed in Section 6.1.3), the staff believes that removal of the Btu limits would not significantly reduce reactor trip frequency. In addition, while B&W plants rely on the cross-limits function to prevent overcooling of the RCS resulting from excessive MFW flow following a reactor trip, the Btu limits serve as a backup to the cross-limits function when the cross-limits function is not active. The staff is not convinced that the addition of a rapid feedwater reduction circuit will adequately compensate for removal of the Btu limits function and believes that modifications to automatically switch the rod controls from manual back to automatic to restore the cross-limits function on a reactor trip could increase transient complexity and operator burden as well as possibly introduce potential undesirable failure modes. Thus, the staff believes that the BWOG Steering Committee should reconsider its acceptance of this recommendation.

Recommendation C.5 involves the elimination of the delta T_{cold} function from the ICS; this function is used to control temperature differences between the two reactor coolant piping system cold legs. Since insufficient justification was provided for removal of this function, the staff does not concur with removal of the delta T_{cold} controls from the ICS as proposed in recommendation C.5.

The staff has reservations concerning recommendations TR-204-ICS and E.22. The staff agrees with reducing the automatic ICS runback rate for asymmetric rod conditions to prevent unnecessary reactor trips; however, BAW-1919 does not provide sufficient justification to eliminate the asymmetric rod runback function from the ICS. E.22 recommends that plant operation with the PORV block valve normally closed be evaluated. It is not clear to the staff that plant operation with the block valve closed would be an enhancement to plant safety. This recommendation should be carefully evaluated from the standpoint of transients that could challenge pressurizer safety valves and availability of the PORV as an alternate means for depressurization and heat removal.

Recommendation TR-010-ICS proposes ICS modifications to eliminate problems experienced when controlling T_{ave} with the feedwater control system, including oscillations around the T_{ave} set point and upsets in feedwater flow when returning control of T_{ave} from the feedwater control portion of the ICS back to the reactor (control rod) control portion. The BWOG recommends a pilot program to install a new conceptual design to prevent these problems on an operating reactor. However, in BAW-1919 the BWOG indicates that because of the limitations of the simulators used, the conceptual control scheme was not evaluated to confirm its effectiveness. While the staff believes that improved control is desirable, the new controls should be carefully analyzed before installation at an operating plant. Recommendation C.4 is to separate control of each feedwater regulation valve from each other. It is not clear to the staff what is specifically intended by this recommendation.

The staff considers recommendation TR-190-ICS to be a key recommendation because of its importance with regard to resolving previously identified ICS/NNI concerns; the staff strongly encourages its implementation. The staff also strongly encourages implementation of recommendation TR-194-ICS and believes that recommendations C.1 and C.7 have significant potential benefit in developing a control system design that is more tolerant to failures; therefore, this too should be actively pursued by the BWOG.

6.1.5 Failure Modes and Effects Analyses

During the BWOG ICS/NNI reassessment effort, updated failure modes and effects analyses (FMEA) of the ICS/NNI systems were performed and the results of these studies were used for identifying and evaluating potential ICS/NNI improvements. The BWOG used the FMEA as a tool in the development and validation of recommendations for ICS/NNI improvements that, when implemented, will result in a reduction in transients either initiated or complicated by selected classes of failures in the ICS/NNI. The scope of these FMEA considered power supply failures, input signal failures, and output signal failures, since these classes of failures represent most of the failures that have occurred in ICS/NNI systems.

The recommendations developed from these FMEA can be correlated to recommendations in the RTS or recommendations pending approval by one or more BWOG committee. Since the corresponding BWOG recommendations are addressed in other sections of this supplement, the validity of the recommendations resulting from FMEA is not addressed in this section. Consequently, this section deals specifically with the scope of these FMEA and the models used to perform them.

Because the adequacy of the model significantly influences the adequacy of the analysis of simulated system performance, the staff evaluated the FMEA models and scope to ensure that the results of the FMEA were adequate for validating existing BWOG recommendations. Additionally, it needed to determine whether other classes of failures had been overlooked, thereby resulting in the omission of additional recommendations that could further reduce the occurrence of plant transients. If the models and scope of the FMEA proved to be adequate, the staff could conclude that the FMEA results were applicable to the analyses goals. If there are deficiencies in the models and scope, then the impact of those deficiencies must be evaluated with regard to the goals of the analyses.

Since some staff concerns regarding FMEA scope are based on the results of earlier analyses performed on the ICS/NNI systems, these FMEA will be discussed in the order of their development.

Following the TMI-2 accident in 1979, the NRC determined that certain aspects of the B&W plant design needed additional assessment. Consequently, in May 1979, the NRC issued a shutdown order to all B&W-designed plants. The order directed that the licensees analyze the role of control systems failures and their significance to safety.

B&W, in response to the NRC order and on behalf of the BWOG, performed a FMEA of the ICS (BAW-1564) to identify sources of transients initiated by the ICS and to define potential areas for improvement to reduce the frequency of transients. The FMEA also was used to determine whether an ICS failure could cause a failure mode whereby plant safety systems would not protect the reactor core. The emphasis was on analyzing ICS failures that could affect the feedwater system, emergency feedwater system, and pressurizer pressure and level controls and challenge the pilot-operated relief valves (PORVs), safety valves, and RPS/ESFAS.

The staff's review of the B&W FMEA led to the following conclusions:

(1) The significance of the ICS with regard to plant safety systems was not adequately addressed.

(2) A fault tree for loss of feedwater should have been developed on the basis of equipment diagrams rather than functional blocks.

(3) The FMEA should have included other systems with which the ICS interacts, such as NNI and power sources.

(4) Power supply failures should have been evaluated in detail, and specific recommendations concerning power supply reliability should have been developed.

(5) The simulation tools used in the FMEA were deficient in their dynamic range and component details; however, the deficiencies did not greatly affect the overall results, since a reactor trip was the terminating point for the simulations. If more detailed evaluations of ICS/NNI failure modes are performed, more sophisticated system simulation tools should be used.

(6) Improvement is needed in areas of ICS/NNI system arrangement, channeling, and selection of input signals.

(7) Since there is a tight coupling between the secondary system--which is controlled by the ICS--and the primary system, dynamic performance should be studied, including the effects of control limitations on plant stability.

These comments were forwarded to the BWOG, but further analyses were not performed to specifically address these comments.

Following a series of abnormal events involving non-Class-1E power supply failures at several B&W-designed plants between 1978 and 1980, the NRC issued IE Bulletin 79-27, which included requirements for all operating nuclear power facilities to review Class 1E and non-Class 1E buses supplying power to safety- and non-safety-related instrumentation that could affect the ability to achieve a cold shutdown using existing procedures. A FMEA of the power supplies to the Oconee Unit 1 ICS and NNI instrumentation systems was performed for the NRC by Science Applications International Corporation (SAIC). A detailed FMEA model was developed on the basis of equipment diagrams rather than functional modules. While the model included the NNI and power supplies, it did not include the plant safety systems.

Deficiencies in the scope of the Oconee-1 FMEA led the staff to conclude that additional recommendations regarding the Oconee-1 ICS/NNI would be forthcoming if the FMEA model were updated to reflect the staff concerns. These staff concerns are

(1) lack of dynamic range in the simulations
(2) missing power supply branches in the system models
(3) no consideration of the effects of individual module failures
(4) omission of safety system interactions
(5) no consideration of failures external to the ICS/NNI that could affect ICS/NNI responses

The staff further recognized that many of its comments described in its review of BAW-1564 were not applied to this FMEA. Therefore, the staff concludes that the scope of this FMEA was marginally adequate and the results must be carefully evaluated for applicability before being extended to other Model 721 ICS/NNI systems.

As part of the ICS/NNI reassessment, the BWOG performed FMEA of the ICS/NNI systems in use at Davis-Besse and Three Mile Island Unit 1 (TMI-1). The Davis-Besse FMEA was then used as the base FMEA for the analyses performed at Rancho Seco Unit 1, Crystal River Unit 3, and Arkansas Nuclear One Unit 1 (ANO-1). Additionally, the results of the Oconee-1 FMEA were incorporated into the TMI-1 FMEA. Significant failure modes discovered by these FMEA were then used by B&W as initial conditions to simulate plant responses.

The BWOG performed the FMEA assuming the plant was operating at full power. The staff believes the FMEA scope should have included other plant operating states. The 100-percent full-power operating state was selected because the BWOG concluded that transients caused by single ICS/NNI failures at this power level would bound all other classes of ICS/NNI failure modes. The staff, on the other hand, concludes that there are other operating states, such as single feedwater pump operation and operating with less than four reactor coolant pumps, that may lead to other significant plant transient responses that would not be revealed by analyzing only the full-power cases.

Operator responses were considered only from the perspective of initial indications in the control room. Incorrect operator responses were not simulated because the BWOG considered the problem of simulating the wide range of potential failures outside the scope of the FMEA program. Operator errors of omission or commission are difficult to analyze because of the wide variation in potential actions. Nevertheless, the FMEA scope should have considered operator errors of conservative magnitude, such as errors of failing to perform a prescribed function or errors caused by performing the most likely incorrect function for a given scenario. The fault tree methodology could have been employed to assist in the analysis of this class of transients.

Within the scope of the BWOG FMEA program and model limitations, the FMEA addressed loss of power to individual loads and individual buses (at all voltage levels) regardless of whether multiple power sources were provided through auctioneering or automatic bus transfer devices. The staff believes this is a significant improvement over the original FMEA (BAW-1564).

The scope of the FMEA was adequate for addressing staff concerns regarding overfill failure modes and effects, within the power level constraint imposed upon the FMEA simulation tools (100 percent power with all controls in their optimum positions). For certain failure scenarios, the Davis-Besse FMEA showed activation of the Btu limiting circuits to help mitigate the overfill condition. Both Crystal River 3 and ANO-1, which used the Davis-Besse FMEA as the base FMEA, do not have active Btu limiting circuits. While the BWOG noted this deficiency in the FMEA for these plants, no additional analyses were performed. The staff believes the BWOG should consider performing additional analyses for these plants to ensure that adequate overfill protection is provided.

These FMEA did not include safety system interactions (such as ECCS or ESFAS interactions). However, the staff believes that the safety systems are adequate to protect the plant from ICS/NNI failures. Thus, the fact that these FMEA did not model safety system interactions is not considered to be a significant deficiency.

The FMEA scope did not include ICS/NNI system responses to failures external to the ICS/NNI, such as in the instrument air system. Rather, the FMEA scope focused on how ICS/NNI failures affect plant response. Since there is a tight coupling between the secondary system and the primary system, dynamic system performance should have been included in the scope of these FMEA, including the effects of control limitations on plant stability. The BWOG appears to be assuming that the ICS/NNI will respond favorably in all cases where the initiating failure is not in the ICS/NNI hardware. This has not been verified by the FMEA.

A limited number of the FMEA recommendations could be used to validate some of the system requirements; however, the system requirements were general, while the recommendations were more succinct; consequently, this comparison effort could not be performed adequately. These FMEA identified problems other than loss of ICS/NNI power and indicated that more detailed systems requirements may be necessary.

The staff finds that these FMEA were specifically constructed to examine failures, including power supply failures, which were representative of the types of ICS/NNI failures that have been experienced. The staff also concludes that the results of these FMEA validate many of the SPIP recommendations. However, because simulation was limited to only 100-percent power operation, the completeness of the BWOG efforts cannot be determined.

In light of the staff's findings on the FMEA methodology, it is likely that there are additional ICS/NNI failure modes that have not been discovered or addressed by the SPIP recommendations. These failure modes may result in future B&W plant reactor trips and/or Category B and C transients. However, it is the staff's opinion that the B&W plant safety systems are capable of mitigating such events. Thus, the limitations of the FMEA are not believed to be safety significant. Nevertheless, to ensure that the frequency of complex plant transients is reduced at B&W plants, the staff encourages the B&WOG to consider the staff's comments on the FMEA program and to continue its investigation of the plant response to ICS/NNI failures.

6.1.6 Loss and Restoration of ICS/NNI Power

The results of the staff's technical review and evaluation of the BWOG recommendations related to the effects of partial and total loss of ICS/NNI ac and/or dc power on plant equipment and control room instrumentation and controls and the subsequent restoration of power are given below. The following recommendations were reviewed:

Number	Recommendation
TR-012-ICS*	Determine if operator has necessary information from procedures, indicators, etc. to detect loss of NNI power.
TR-032-ICS*	Evaluate restoration of ICS/NNI power.
TR-033-ICS*	Ensure that plant will go to a safe state on loss of ICS/NNI power.
TR-036-ICS	Evaluate turbine bypass valve position on loss of ICS.
TR-037-ICS*	Evaluate MFW pump speed control on loss of ICS power.
TR-062-OPS	Maintain a high SPDS availability by corrective and preventive maintenance.
TR-096-MSS*	Evaluate design of turbine bypass and atmospheric dump systems.
TR-097-EFW*	Evaluate design of EFW flow control valves.
TR-154-ICS*	Provide operator with unambiguous status of indicators and recorders in main control room on loss of ICS/NNI power or signal.
TR-158-OPS	Re-evaluate annunciator designs to ensure key alarms do not go unnoticed.
TR-159-OPS*	Evaluate secondary system controls to achieve remote manual control in the main control room of all post-trip steam flow paths, MFW and EFW.
TR-167-PES	Include in operating procedures guidance on restoration of power to electrical buses, especially if the ICS or ICS controlled equipment is affected.
TR-172-PRV	Evaluate PORV circuitry to determine if momentary loss of power or restoration of power can cause PORV to open.

*The BWOG has identified these recommendations as key recommendations.

Number	Recommendation
TR-178-ICS*	Ensure plant goes to a known safe state on loss of power to the ICS/NNI. (Note: this recommendation supersedes TR-033-ICS.)
TR-181-OPS	Verify adequacy of instrumentation and displays used to assess and control the ATOG stability parameters.
TR-185-ICS	Power feedwater flow recorders directly from NNI.
TR-190-ICS	Develop backup controls for pressurizer level and pressure control.
TR-191-ICS	Separate condensate flow from NNI power (for Crystal River only).
TR-195-ICS	Supply hand and automatic power circuits from separate panels.
TR-197-ICS	Provide automatic power transfer for the modulating pressurizer heater E/I converters.
TR-211-ICS	Develop modification to remove automatic ICS trip on NNI single power failure.
B.2.2	Power automatic (branch H) and hand (branch HY) power from different panels.
B.2.4	Indicate loss of ac power on load side of ICS automatic bus transfer (ABT) to control room operators.
B.2.7	Automatic trip of feedwater pumps.
B.2.8	Suppression of spurious alarms.
B.2.9	Power supply failure alarms.
B.2.10	Recommendations to "purify" automatic control response following loss of hand power (applicable to plants with Bailey Controls Model 721 equipment).
B.2.30	Review indicator, recorder, and signal select power supply tags to ensure complete information for signals that involve more than one hand station.
B.2.31	On input power momentary disruptions, the ICS and NNI shall not cause the PORVs, turbine bypass valves, and atmospheric dump valves (where applicable) to inadvertently operate, affecting the ability to attain a known safe state.
C.8	Eliminate mid-scale failures that can affect indication or plant control.

*The BWOG has identified these recommendations as key recommendations.

Number	Recommendation
E.9	Close main FW block valves on loss of ICS dc power.
E.10	Evaluate ICS response to NNI power failure for other input signal select switch positions.
E.11	Provide identification of automatic power bus transfer, which is discernable outside the system cabinets.
E.16	Develop independently powered hand or automatic controls for turbine bypass valves, main and startup feedwater valves, and main feed pump speed control.
RR-TIR-24	The BWOG I&C Committee should develop a post-trip or post-transient troubleshooting procedure to aid in diagnosing ICS module failures.

In general, the staff agrees with and encourages implementation of the above recommendations with the exceptions of B.2.8 and TR-211-ICS. Recommendation B.2.8 involves changing alarm logic from deenergize-to-actuate to energize-to-actuate to reduce the number of spurious signal monitoring alarms that occur upon ICS/NNI power supply failures. The staff agrees with the objective of eliminating irrelevant/nuisance alarms on power supply failures, but does not concur with recommendation B.2.8 in the absence of further direction to ensure that (1) the revised alarm configuration will not suppress useful information pertinent to timely identification of other ICS/NNI system failures and (2) the revised alarm configuration will not have the potential for misleading the operator in the absence of a legitimate alarmed condition. The alarm system should allow for timely identification of total or partial ICS/NNI power losses.

The staff is puzzled by the recent appearance (letter dated March 2, 1988) of recommendation TR-211-ICS. This recommendation was previously rejected by the BWOG because it was in conflict with recommendation TR-178-ICS regarding known safe state (KSS). As discussed later in this section, the staff considers the KSS concept to be the most significant recommendation of the ICS/NNI reassessment. Thus, the staff cannot agree with a recommendation that conflicts with the KSS recommendation. The staff strongly encourages the BWOG to reconsider its acceptance of TR-211-ICS.

Implementation of recommendations TR-012-ICS, TR-158-ICS, B.2.4, and B.2.9 should help to improve ICS/NNI-related alarm/annunciator circuit designs. The staff also believes that implementation of recommendation TR-154-ICS to provide the operator with unambiguous status of control room indicators and recorders upon loss of ICS/NNI power or signals will result in substantial benefit to the operator in responding to such events, provided that the indication of failed status is obvious and that timely identification of the failed status does not rely on operator analysis.

A number of the proposed recommendations are directed toward reducing the contribution of ICS/NNI-controlled equipment failures to transient severity/

complexity by identifying and recommending preferred/desired equipment failure modes for ICS/NNI power losses and by retaining manual control capability for the equipment following power losses. These recommendations include TR-036-ICS, TR-037-ICS, TR-096-MSS, TR-097-ICS, TR-159-OPS, and TR-172-PRV. If properly implemented, these recommendations would help to resolve previously identified ICS/NNI concerns.

The staff is concerned that the scope of some recommendations is too limited, and that a number of the recommendations do not contain sufficient details to ensure that implementation will resolve the related concerns. It is not clear from the information provided in BAW-1919 why certain chosen equipment failure modes are considered optimum. It should not be assumed that a given failure mode is proper for all plant conditions. BAW-1919 does not contain sufficient information for the staff to conclude that implementation of the recommendations will resolve concerns regarding the loss of both automatic and remote manual control capability for plant equipment upon loss of ICS/NNI power, which can complicate plant recovery/stabilization. Furthermore, the staff cannot conclude that sufficient instrumentation remains available in the control room for the operator to assess plant status and to achieve a safe shutdown condition following a loss of ICS/NNI power, or that sufficient periodic surveillance is performed to ensure operability of the instrumentation.

The staff believes that recommendation TR-190-ICS to develop backup controls for pressurizer level and pressure control should be a high priority recommendation because it will result in a significant reduction in ICS/NNI transient complexity by providing redundant primary system level and pressure control capability. The staff encourages the BWOG (1) to provide further guidance for implementation of TR-190-ICS to ensure independence of the backup controls from the normal controls and (2) to address the problems of ICS/NNI failures concerning reactor makeup flow control, letdown flow control, and seal injection flow control.

The most significant recommendation resulting from the BWOG ICS/NNI reassessment effort is TR-178-ICS, which requires that the plant goes to a KSS in response to loss of ICS/NNI power transients and only relies on manual actions for which the operator is normally trained and that can be accomplished from the control room. However, based on the staff's review of Appendix R-g, "Known Safe State on the Loss of ICS/NNI Power," it appears that some losses of power can still cause failure of instrumentation and control channels for which backups do not exist and can potentially result in relatively complex transients requiring manual actions beyond those normally required of the operator. At present, the staff is not convinced that all B&W plants will be able to attain a KSS for all potential ICS/NNI power losses, in accordance with the BWOG definition of KSS. The staff believes that the combined benefit of implementation of approved Level I recommendations will result in substantial improvement in the ability of B&W plants to achieve a KSS following losses of ICS/NNI power.

The BWOG Steering Committee rejected recommendation RR-TIR-24 to have the BWOG I&C Committee develop a post-trip or post-transient troubleshooting procedure to aid in diagnosing ICS module failures. The basis for rejection was that the recommendation had no benefit to reduce the number of reactor trips or complex transients. The staff disagrees with the BWOG basis for rejection. It appears to the staff that careful inspection and determination of the root cause(s) for ICS module failures could result in corrective actions that could improve

ICS/NNI performance and reduce the ICS contribution to reactor trip frequency and transient complexity. The staff recommends that the BWOG reconsider this recommendation because it has the potential benefits.

The staff considers recommendations TR-154-ICS, TR-178-ICS, TR-190-ICS, and C.8 to be key recommendations because of their importance with regard to resolving previously identified ICS/NNI concerns; the staff strongly encourages their implementation.

6.1.7 ICS/NNI Power Reliability

The staff's technical review and evaluation of the BWOG recommendations related to reliability of the electrical power sources and distribution systems external and internal to the ICS/NNI are given below. The following recommendations were reviewed:

Number	Recommendation
TR-013-ICS*	Prevent loss of power to the ICS or NNI.
TR-039-ICS*	Wire the power supply monitor in the ICS/NNI directly to the output bus after the auctioneering diodes.
TR-053-SFI	Correct overheating problems that can lead to electronic power supply malfunctions.
TR-102-ICS*	Install redundant dc power supplies for NNI-Y (for AP&L only).
TR-103-ICS	Fuse external power leaving ICS/NNI cabinets (for FPC only).
TR-105-ICS*	Perform field verification of ICS/NNI drawings.
TR-113-PES	Review breaker control power distribution to determine effects of a loss of the battery bus.
TR-116-PES	Review dc charging system and ensure the charging voltage does not exceed plant equipment voltage rating.
TR-117-PES*	Modify inverter overcurrent protection to ensure the breaker/fuses open on overcurrent before inverters fail.
TR-118-PES	Evaluate loadings on ac and dc vital buses to ensure adequate margins exist without trip of equipment.
TR-119-PES*	Implement preventive maintenance for electrical buses.
TR-182-ICS	Evaluate installing automatic bus transfer switches of MFW pump controllers (for Davis-Besse only).
TR-183-ICS	Preventive maintenance and testing for ABT switches.

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*The BWOG has identified these recommendations as key recommendations.

Number	Recommendation
TR-184-ICS	Provide separate fuses for hand stations that use ac power.
TR-186-ICS	Minimize access to ICS/NNI cabinets during operation and train maintenance personnel on location of power distribution components.
TR-187-ICS	Install current and voltage meters for NNI power supplies (for Davis-Besse only).
TR-188-ICS	Maintain dc power supply current balance and perform a periodic full load test for each power supply.
TR-192-ICS	Remove/modify NNI-Z power supply and signal select logic (for Rancho Seco only).
TR-203-PES*	Establish preventive maintenance to increase reliability of inverters.
E.1	Increase dc current limit.

In general, the staff agrees with and encourages implementation of the above recommendations with the exception of TR-039-ICS, TR-192-ICS, and E.1. It is not clear to the staff that the advantages and disadvantages of all possible power supply monitor (PSM) locations have been thoroughly investigated. The staff believes that there are disadvantages associated with connecting the PSM sense/operate lines directly to the ±24-volt dc ICS/NNI buses as recommended in TR-039-ICS. The staff disagrees with removal of the NNI-Z power supplies at Rancho Seco as recommended in TR-192-ICS because it appears that their value (providing the capability to switch from failed to valid signals for input to the ICS and control room indications) outweighs any disadvantages concerning inadvertent PSM actuations. Raising the current limit of the ICS/NNI ±24-volt dc power supplies as recommended in E.1 would reduce design margins for the power supplies and may actually reduce system reliability under certain conditions; therefore, this is discouraged.

The staff believes that recommendation TR-116-PES, which requires verification that dc system charging voltage does not exceed the voltage rating of the associated plant equipment, should be expanded to also address potential problems that may occur because the equipment voltage ratings will be exceeded during the time the plant batteries are subject to an equalizing charge.

The staff believes that a number of the recommendations addressed in this section, if implemented, should significantly enhance ICS/NNI power distribution system reliability. The staff concludes that the likelihood of reoccurrence of specific past events will be reduced by implementation of the recommendations that focus on problem areas that have contributed to transient complexity. However, the staff believes that certain recommendations (e.g., TR-013-ICS, TR-053-SFI, and TR-113-PES) may be too general to ensure that implementation of the recommendation will satisfy the program objectives and systems requirements established by the BWOG.

*The BWOG has identified these recommendations as key recommendations.

Two major areas of concern regarding ICS/NNI power reliability, which the staff does not consider to be resolved by the ICS/NNI reassessment, are (1) resolution of the concerns addressed in IE Bulletin 79-27 and (2) the design and operation of the ICS/NNI PSM modules. These issues are discussed below.

On November 10, 1979 an event occurred at the B&W-designed Oconee nuclear plant that involved a loss of ICS and NNI power. As a result of this event, the NRC issued on November 30, 1979, IE Bulletin 79-27, "Loss of Non-Class 1E Instrumentation and Control Power System Bus During Operation." This bulletin required licensees (1) to review the effects of loss of power to each Class 1E and non-Class 1E bus supplying power to plant instrumentation and controls and (2) to determine the resulting effect on the capability to achieve a safe (cold) shutdown condition via procedures following the power loss. A further requirement of the bulletin was to develop procedures necessary to achieve cold shutdown following bus power failures. The intent of IE Bulletin 79-27 was to ensure that the loss of power to any bus could not result in control system actions that cause a plant upset/transient condition requiring operator action concurrent with the loss of control room information (e.g., indications and alarms) upon which the actions would be based.

On February 26, 1980, an event occurred at the B&W-designed Crystal River nuclear plant that involved a loss of NNI power. The event at Crystal River involved the types of concerns that were identified by IE Bulletin 79-27. By letter dated March 6, 1980, the staff required all licensees of B&W-designed reactors to expand the review of IE Bulletin 79-27 to include the implications of the Crystal River event. On March 7, 1980, the staff issued IE Information Notice 80-07, which described the Crystal River event and stated that IE Bulletin 79-27 was intended to cause licensees to investigate the loss of individual power supplies as well as the loss of inverters and vital buses. In addition, NUREG-0667, "Transient Response of Babcock & Wilcox - Designed Reactors," published in May 1980 recommended that prompt followup actions be taken on IE Bulletin 79-27 to improve ICS/NNI reliability.

On December 26, 1985, an event occurred at the B&W-designed Rancho Seco nuclear plant that involved a loss of ICS ±24-volt dc power. Upon loss of ICS dc power, equipment control modules lost power, thus providing zero dc voltage, and switching relays lost power, going to the de-energized state. This caused ICS-controlled plant equipment to change positions, initiating a plant transient, and caused the loss of remote manual control of key ICS-controlled plant equipment from the control room. Furthermore, the operators were misled by an MFW flow recorder that had failed to the mid-scale position upon the loss of power.

The incident at Rancho Seco on December 26, 1985, was significant because it again demonstrated that a single failure in the non-safety-related ICS/NNI could subject the plant to an undesirable transient and challenge the operator's capability to mitigate the transient without resulting in primary system undercooling or overcooling. The event demonstrated that the implementation of modifications to meet the concerns of IE Bulletin 79-27 were inadequate at B&W-designed reactors.

The staff's final review of utility responses to IE Bulletin 79-27 only focused on whether there was reasonable assurance that the concerns of the bulletin had been properly addressed. Additional background information regarding licensees' responses to IE Bulletin 79-27 and the staff's evaluation of these responses is provided in Section 7, "Precursors to the December 26, 1985 Incident at Rancho Seco and Related NRC and SMUD Actions," of NUREG-1195, "Loss of Integrated Control System Power and Overcooling Transient at Rancho Seco on December 26, 1985." The staff believes that if a more thorough and in-depth review of plant designs by the utilities in accordance with the bulletin had been performed the potential for this type of event would have been detected and resulted in hardware and/or procedural modifications to ensure that a safe shutdown condition could be achieved following an ICS power loss.

Because of the deficiencies identified in the IE Bulletin 79-27 review performed at one B&W-designed plant and the importance of proper resolution of the bulletin concerns, the staff requested the BWOG to address these concerns as part of the ICS/NNI reassessment effort. The BWOG has indicated that the IE Bulletin 79-27 issue is plant specific and should be addressed to the individual utilities and is not within the scope of BAW-1919. Therefore, the staff is unable to conclude from its review of the BWOG ICS/NNI evaluation that the concerns identified in IE Bulletin 79-27 have been satisfactorily implemented for B&W reactors. The staff considers resolution of the IE Bulletin 79-27 concerns at B&W reactors to be necessary to resolve concerns regarding the effects of non-safety-related (control) system power supply failures at B&W plants. Therefore, the staff recommends that, as a part of the plant-specific implementation audits discussed in Section 12.2, the plant specific actions taken in response to IE Bulletin 79-27 also be audited.

The BWOG has proposed (recommendation TR-039-ICS) to change the way in which the PSM is connected within the ICS and NNI ±24-volt dc distribution systems. The change involves connecting the PSM positive and negative 24-volt dc bus monitoring circuits to termination points on the buses themselves, thus sensing bus voltage directly. The PSM module has typically been included in a "daisy chain" circuit that provides power to additional ICS/NNI electronic modules. It is not clear to the staff that connecting the ICS/NNI PSMs directly to the buses is the preferred/optimum arrangement. One apparent advantage of locating the power supply module at the end of a daisy chain would be that the PSM would detect and provide protection for degraded voltage conditions within the distribution system (caused by individual module failures within the daisy chain) as well as for degraded voltage at the ±24-volt dc buses. The staff acknowledges that in the existing ICS/NNI power distribution configurations, which consist of multiple daisy chain end points, it is not possible to monitor the voltage to all system modules with a single PSM. However, it appears that with relatively minor changes it may be possible to modify the existing power distribution configuration such that a single PSM can be used to monitor the voltage supplied to the majority of the system modules. Connecting the PSM inputs directly to the buses does not provide a means to prevent control system operation with degraded voltage to individual cabinets or groups of modules because of the series resistances or open circuits in the distribution wiring; although it may reduce the probability of PSM actuation, it does not reduce the probability of this type of problem.

In addition, the staff has identified concerns regarding questionable design characteristics of the PSM module itself, including

- use of monitored voltage to operate the undervoltage detection circuits

- the lack of a seal-in feature upon sensing that the monitored voltage has exceeded alarm and trip set point values

- the small hysteresis value between the trip set point and reset point

The staff recommends that the BWOG consider performing additional investigations concerning the adequacy/desirability of the PSM design and operation. This review should include an independent design analysis of the PSM to determine the appropriateness of the design concept to serve its intended purpose and a detailed circuit analysis to determine the adequacy of the design to implement the design concept. The design analysis should include an assessment of the appropriateness of using the same voltage to operate the PSM as is being monitored by the PSM and an assessment of the need for "seal in" features. The analysis also should include a determination of the optimum PSM location within the ICS/NNI distribution systems and the benefit that could be gained by using redundant PSMs in conjunction with any PSM design changes considered to be appropriate. Furthermore, if it is determined that a design function of the PSM is not to provide protection against ICS/NNI module operation with voltages outside of the vendor-specified ±24-volt dc ±10 percent tolerance band, then methods should be developed for ensuring that module operating voltage remains within the specified tolerances.

6.1.8 ICS/NNI Maintenance and Surveillance Testing

The staff's technical review and evaluation of the BWOG recommendations related to ICS/NNI periodic maintenance and surveillance testing (including system tuning are given below). The following recommendations were reviewed:

Number	Recommendation
TR-009-ICS*	Improvements in ICS tune control circuits.
TR-010-ICS	ICS control circuit modification.
TR-011-ICS	Determine if the grid frequency error circuit has been detuned.
TR-038-ICS*	Develop and implement a preventive maintenance program for the ICS/NNI.
TR-068-MFW	Develop a post-maintenance testing program for the MFW pump turbines and governor controls.
TR-079-MFW	Put MFW regulating valves, main block valves, and startup control valves on a refueling frequency for an operational check.
TR-105-ICS*	Perform field verification of ICS/NNI drawings.
TR-106-ICS	Remove unused hardware from ICS/NNI cabinets.
TR-107-ICS*	Improved maintenance and tuning of ICS.

*The BWOG has identified these recommendations as key recommendations.

Number	Recommendation
TR-111-RPS	Review safety system surveillance procedures for checking which channel is available for testing prior to initiation of test.
TR-119-PES*	Implement preventive maintenance for electrical buses.
TR-163-EFW*	Review EFW surveillance and test procedures to ensure that components used in the EOPs are included in the test program.
TR-164-EFW*	Review EFW preventive maintenance program, including minimizing potential from common-cause failures arising from maintenance and testing procedures.
TR-165-EFW*	Review EFW maintenance and test procedures to eliminate conflicting and confusing instructions.
TR-166-EFW	Implement a program to improve and maintain the availability and performance of the EFW systems.
TR-168-MTS	Provide guidance in procedures when troubleshooting the EHC.
TR-183-ICS	Preventive maintenance and testing for ABT switches.
TR-186-ICS	Minimize access to ICS/NNI cabinets during operation and train maintenance personnel on location of power distribution components.
TR-188-ICS	Maintain dc power supply current balance and periodically perform full load test for each power supply.
TR-203-PES*	Establish preventive maintenance to increase reliability of inverters.
TR-208-ICS	Label ICS/NNI switches S1 and S2 to detect energized vs. tripped positions.
TR-210-ICS	Establish program to monitor control system.
B.2.15	Perform periodic surveillance of ICS and NNI cabinet DC voltage.
B.2.16	Periodically record ICS module input and output voltages.
B.2.17	Periodically test functions of ICS not normally demanded.
B.2.25	Review/revise ICS and NNI drawings as needed for legibility.
B.2.26	Improve documentation for ICS tuning.
E.26	Build a data bank for analyzing plant transients and assisting in predicting areas (hardware) that degrade overall system performance.

*The BWOG has identified these recommendations as key recommendations.

Number	Recommendation
TAC-2.e	The post-maintenance program recommended in TR-068-MFW should include proper checkout and tuning of newly installed systems or equipment prior to unit startup, and planned tuning at power should be incorporated into plant procedures.
TAC-3.a	Include the lessons learned from the Rancho Seco experience regarding high resistance relay contacts as a part of the recommended ICS/NNI preventive maintenance program (TR-038-ICS).
TAC-3.c	Modify the existing RTS recommendation TR-038-ICS to include internal and external power supplies to the ICS/NNI.
RR-TIR-26	Review and upgrade maintenance procedures as necessary to ensure that proper checkout of replacement modules and assemblies in the CRDS is made before their use. This recommendation should be extended to ICS and EHC components, circuit boards, etc.
RR-TIR-8	Review ICS tuning to ensure control settings are compatible with a turbine trip runback from less than 45 percent power.

The staff generally agrees with and encourages implementation of all of the above recommendations. As an important element necessary for improved ICS/NNI performance, the BWOG has identified the need to perform routine maintenance and surveillance on ICS and NNI components and actuated equipment to ensure and periodically verify their proper operation. Poor maintenance and surveillance practices for ICS/NNI components and actuated equipment at B&W plants have directly contributed to the frequency and severity of transients involving the ICS/NNI. Many reported reactor trips and operational transients have been attributed, in whole or in part, to inadequate maintenance and surveillance. On the basis of its review of the combination of approved and proposed recommendations developed by the BWOG concerning maintenance and surveillance of the ICS/NNI and other systems/equipment, the staff concludes that the BWOG is placing proper emphasis in this area. For example, the BWOG has emphasized the importance of correct system tuning (i.e., ensuring that system components/modules are adjusted such that controlled systems/equipment perform their design functions without unnecessary oscillations, perturbations, or delays that could challenge plant operators and plant equipment).

Although the BWOG reassessment effort has been very productive with regard to establishing ICS/NNI maintenance and surveillance programs, the staff is concerned that the guidance provided in recommendations TR-038-ICS (develop and implement a preventive maintenance program for ICS/NNI) and TR-107-ICS (improve maintenance and tuning of ICS) is too general to ensure resolution of certain specific concerns regarding instabilities and perturbations resulting from improper control system tuning and surveillance. The recommendations do not include or reference the guidance provided in BAW-1919, Appendix C or Appendix R-n, with regard to ICS tuning. These recommendations and the associated referenced source documents also do not include information concerning several additional issues, including the scope and intent of certain tests, recommended PSM module alarm and trip set point values, or specific ICS signals to be monitored during power escalation.

The surveillance frequency specified in the ICS/NNI systems requirements documents for instrument channel calibrations, actuated equipment tests, and system tuning is "at least every other refueling outage." It is not clear to the staff that this surveillance frequency is adequate to fulfill the intent of BWOG recommendation TR-038-ICS, which requires periodic ICS/NNI surveillance/preventive maintenance to improve ICS/NNI reliability. Loss of ICS/NNI power events at B&W plants have been attributed to the lack of preventive maintenance and/or surveillance. An interval of at least every other refueling outage could be 3 years or more. The staff considers such an interval to be excessive for calibration of electronic modules. Electrical/electronic equipment is typically checked annually for operability/drift, if not more frequently depending on its application and the consequences of failure. The BWOG has not provided information in BAW-1919 that supports the frequencies selected for periodic ICS/NNI surveillance and preventive maintenance.

The BWOG rejected recommendation RR-TIR-8 (TAC-1.h), which concerned tuning to ensure control settings are compatible with a turbine trip runback from less than 45 percent power, on the basis that ICS tuning is not a function of the anticipatory reactor trip system (ARTS) set point and that the intent of the recommendation is accomplished by TR-107-ICS. The staff agrees with this.

The BWOG also has rejected recommendation RR-TIR-26 (TAC-5.b), which called for the review and upgrade of maintenance procedures as necessary to ensure that proper checkout of replacement modules and assemblies in the CRDS is made before their use. It also was recommended that this proposal be extended to such items as ICS and EHC components and circuit boards. This recommendation was rejected on the basis that it is not a feasible or practical solution to the problem. The staff does not agree that checkout of replacement components is not feasible. On the contrary, the staff believes checkout of replacement components is not only feasible, but is a necessary and proper practice. The BWOG should reconsider this recommendation.

The staff concludes that implementation of these recommendations will contribute to improved maintenance and surveillance of the ICS/NNI. In addition to recommendations TR-038-ICS and TR-107-ICS, the staff considers recommendations TR-105-ICS, TR-111-RPS, and TR-163-EFW through TR-165-EFW to be key recommendations because of their importance to resolving previously identified ICS/NNI concerns; the staff strongly encourages their implementation. Recommendations TR-009-ICS through TR-011-ICS and recommendation TR-119-PES are further discussed in Sections 6.1.4 and 6.1.7, respectively.

6.1.9 Operator Burden, Procedures and Training

The results of the staff's technical review and evaluation of the BWOG recommendations intended to reduce the burden on control room operators and to improve procedures and operator training for events involving ICS/NNI failures are given below. The staff also reviewed non-ICS/NNI recommendations concerning operator burden, procedures, and training. The following recommendations were reviewed:

Number	Recommendation
TR-012-ICS	Determine if the operator has necessary information from procedures, indicators, etc. to detect loss of NNI power.

Number	Recommendation
TR-018-MFW	Provide training on MFW system components.
TR-034-ICS	Training for loss of ICS power.
TR-035-ICS	Familiarize operators with Rancho Seco event.
TR-059-OPS	Training for personnel who make emergency notifications.
TR-067-MFW*	Wherever possible, eliminate automatic MFW pump trip functions.
TR-069-MFW*	Eliminate automatic control of the MFW block valve except during a reactor trip.
TR-070-MFW*	Provide capability to override a close signal to the MFW block valve.
TR-091-MFW	Eliminate need for an auxiliary operator to open a deaerator feedwater tank drain line after reactor trips (for Davis-Besse only).
TR-098-MFW*	Overfill protection for MFW system.
TR-099-OPS*	Include guidance on excessive MFW, throttling AFW, and throttling HPI in plant procedures.
TR-121-IAS	Make appropriate personnel aware of importance of instrument air system prohibition of use for tools and need to report air system damage.
TR-128-IAS	Review training and loss of air response procedures for instrument air system.
TR-130-IAS	Expand procedure for the loss of instrument air (for ANO-1 only).
TR-144-IAS	Develop or upgrade a loss-of-instrument air procedure (for FPC, GPUN, SMUD and TED only).
TR-154-ICS*	Provide operator with unambiguous status of indicators and recorders in main control room on loss of ICS/NNI power or signal.
TR-155-EFW*	Limit maximum flowrate delivered by the EFW system.
TR-156-OPS	Provide a designated communicator to relay emergency plan messages.
TR-167-PES	Include in the operating procedures guidance on restoration of power to electrical buses, especially if the ICS or ICS controlled equipment is affected.
TR-171-OPS	Evaluate alarm set points to determine if adequate time is provided for operator response.

*The BWOG has identified these recommendations as key recommendations.

Number	Recommendation
TR-178-ICS*	Ensure plant goes to known safe I&C state on loss of power to the ICS/NNI.
TR-181-OPS	Verify adequacy of instrumentation and displays used to assess and control the ATOG stability parameters.
TR-207-OPS	Review operator training with regard to the manual control of MFW post-trip.
TR-212-ICS	Power source switches or breaker positions should be labeled to prevent their energized-vs.-tripped positions from being misinterpreted.
B.2.22	Train ICS/NNI cabinet monitoring personnel in location of power distribution components.
B.2.23	Train ICS/NNI maintenance personnel with respect to power supply distribution to indicators, recorders, and hand stations.
B.2.24	Periodically train operator and maintenance staff in the identification and definition of power supply failure modes and the location and use of alternate controls.
B.2.27	Improve/add NNI output functions and signal input ranges for enhanced readability.
E.6	Evaluate operator's ability to rapidly respond to a power failure to the ICS/NNI.
E.18	Review operating procedures with respect to present ICS trip circuits.
TAC-2.e	The post-maintenance program recommended in TR-068-MFW should include proper checkout and tuning of newly installed systems or equipment before unit startup, and planned tuning at power should be incorporated into plant procedures.
RR-TIR-24	The BWOG I&C Committee should develop a post-trip or post-transient troubleshooting procedure to and in diagnosing ICS module failures.
RR-TIR-26	Review and upgrade maintenance procedures as necessary to ensure that proper checkout of replacement modules and assemblies in the CRDS is made prior to their use. This recommendation also should be extended to ICS and EHC components, circuit boards, etc.

Several recommendations that relate to operator burden, procedures, and training are addressed in other sections of this report (not all of the recommendations

*The BWOG has identified these recommendations as key recommendations.

addressed in other sections are listed above). These include recommendations TR-104-ICS, TR-154-ICS and B.2.14 addressed in Section 6.1.3; recommendations TR-007-ICS, TR-009-ICS and TR-010-ICS addressed in Section 6.1.4; recommendations TR-012-ICS, TR-032-ICS, TR-096-MSS, TR-097-EFW, TR-154-ICS, TR-167-PES, TR-178-ICS, TR-181-OPS, and RR-TIR-24 addressed in Section 6.1.6; and recommendations TR-009-ICS, TR-010-ICS, TR-038-ICS, TR-068-MFW, TAC-2e, and RR-TIR-26 addressed in Section 6.1.8.

The BWOG Operator Support Committee defined operator burden as those factors that unnecessarily hinder the plant operator's ability to prevent a reactor trip or mitigate plant transients post-trip. Staff concerns related to operator burden include dependence on operator actions from outside the main control to mitigate the consequences of events involving the loss of ICS/NNI power and proper control system tuning for reactor operation at low-power levels to prevent undue operator burden in this region (e.g., having to manually control system parameters without tripping the reactor during star up).

Operator burden following loss of ICS/NNI power is closely associated with the concept of KSS. An early effort by the BWOG identified in the RTS (TR-033-ICS) was to make system changes to ensure that on any loss of ICS/NNI power the plant would go to a KSS without any operator action required. This action has since been superseded by recommendation TR-178-ICS, which states that the plant should go to a KSS on loss of power to the ICS/NNI; but with heat balance maintained by either automatic control and/or operator action for which the operator is normally trained and can be taken from the control room. The BWOG has stated that the operator action required for a loss of ICS/NNI power that results in a plant trip should be the same as for any other reactor trip. The staff agrees that it is reasonable to expect such action from the operator. However, the operator action required to achieve a KSS for some loss of ICS/NNI power transients has not been sufficiently addressed in BAW-1919, as discussed in Section 6.1.6 of this report. Appendix P, "A Comparative Study of the Sensitivity of B&W Reactor Plants," concludes that the operator burden following reactor trip (but with no malfunction of the automatic control system) in B&W plants is no greater than for other pressurized-water reactor (PWR) plants, but that the burden imposed on operators to oversee the automatic control system and to take corrective action when they fail is greater in B&W plants than in other PWRs. This is because the control of key variables are vested in a single system (the ICS).

The staff concludes that implementation of the recommendations listed above generally should help to reduce operator burden, hence the probability of operator error, when responding to losses of ICS/NNI power. A number of the recommendations proposed by the BWOG are designed to limit the consequences of losses of ICS/NNI power and to prevent the need for manual operator actions outside the control room following reactor trips and ICS/NNI failures. Improvements in procedures and training are also expected to reduce operator burden.

As part of the ICS/NNI reassessment, the BWOG was requested to evaluate procedures for loss and restoration of ICS/NNI power, maintenance and surveillance procedures for ICS/NNI components, and the adequacy of procedures used to achieve safe plant shutdown following the loss of power to any bus (as required by IE Bulletin 79-27, discussed in Section 6.1.7 of this report). The staff's review of BAW-1919, Appendix R-g, "Known Safe State on the Loss of ICS/NNI Power," indicates that not all plants can achieve a KSS without operator action. Appendix R-g does not provide a discussion of the procedures that would control the operator action, but does identify required operator action to achieve KSS and the timing of and the procedures governing the action as areas for which the BWOG expects to continue to study as a followup action to the SPIP.

BAW-1919 states that the EOPs are backed up by procedures that address abnormal plant configurations created by such conditions as station blackout, loss of ICS/NNI power or loss of instrument air. It also states that the connections and interactions between the EOPs and the specific procedures for abnormal plant configurations should be clear and concise with proper priorities maintained. As part of the ongoing emergency procedures review discussed in Section 7.4 of the SER, the BWOG is reviewing the procedural hierarchy for each plant to determine if the proper links and priorities have been maintained and to make recommendations on where improvements should be made.

BAW-1919 also states that training of personnel is a plant-specific item and that it shall be the owner's responsibility to designate and train system or performance personnel who will be responsible for proper operation of the ICS/NNI systems.

The staff concludes that the SPIP reassessment has proposed recommendations that if adequately implemented will reduce operator burden and improve procedures and training with regard to the ICS/NNI. However, the staff is concerned that insufficient references to source documents and insufficient detail concerning the basis for the recommendation exists for some recommendations. To ensure the resolution of concerns, as intended by the recommendations, the BWOG should ensure that all appropriate source documents are referenced.

6.1.10 Other ICS/NNI Considerations

The results of the staff's technical review and evaluation of the ICS/NNI reassessment with regard to resolution of concerns not covered in other sections of this report and the associated recommendations listed in BAW-1919. This section discusses the following topics:

- steam generator overfill protection
- FSAR assumptions
- NRC Standard Review Plan Section 7.7, "Control Systems"
- integrated vs. discrete controls
- advanced control systems

The following recommendations were reviewed:

Number	Recommendation
TR-098-MFW*	Overfill protection for MFW system.
TR-155-EFW*	Limit maximum flowrate delivered by the EFW system.

*The BWOG has identified these recommendations as key recommendations.

Number	Recommendation
C.1	Provide independent controls for reactor power, steam demand, and feedwater such that no single failure would lead to simultaneous and spurious manipulation of control rods, turbine steam admission valves, and feedwater flow.
C.7	Provide separate subsystems for reactor coolant temperature and steam pressure control.
D.1	Identify, document, and resolve problem areas in the ICS/NNI system that, based on past operating experiences and system evaluations, can be corrected by an advanced control system.
D.2	Develop an optimal system configuration for the advanced control systems to determine desirable interfacing to other plant systems and to minimize plant changes required to accommodate the new system.
D.3	Explore technological improvements/capabilities for incorporation into the advanced system to improve operability reliability and maintainability.
D.4	Review control loop architecture for possible upgrades to provide improved unit control and man-machine interfaces.
D.5	Ensure capabilities of the plant safety-related systems to achieve a safe shutdown condition are not dependent upon or compromised by action of the plant control system.
D.6	Provide advanced control system features to regain plant stability during abnormal plant conditions prior to challenging the plant protection systems.
E.14	Remove automatic control functions for the atmospheric dump valves from the ICS/NNI system.
E.23	Post-trip EFIC actuation and isolation of MFW on high steam generator (SG) level.
RR-6	Emphasize the need to establish communication with the control room before taking action on safety system equipment if an unplanned actuation occurs during maintenance, testing, etc.
RR-TIR-19	The plants that trip both MFW pumps on high SG level should evaluate sequential vs. simultaneous MFW pump trip.

The staff generally agrees with and encourages implementation of the above recommendations.

The staff has identified concerns regarding the potential for steam generator overfill (and potential overcooling) events from excessive MFW flow and/or excessive AFW flow. Overfill could occur as a result of the MFW startup valve failing mechanically or the loss of its ICS control signal upon loss of ICS power (this would cause the valve to go to the 50-percent open position, which could allow excessive post-trip MFW flow). In addition, although AFW system control has been made independent of the ICS, the AFW system designs typically include preferred failure modes for valves that ensure adequate AFW flow for decay heat removal; these same failure modes (e.g., flow control valves failing open on loss of power or instrument air) could lead to SG overfill.

The staff requested the BWOG to address the potential for SG overfill from both the MFW and AFW systems. The BWOG stated that the plant-specific FMEA program (addressed in Section 6.1.5 of this report) would address potential ICS failures that could cause SG overfill. Several of the recommendations proposed by the BWOG should help to reduce the probability of SG overfill events. The staff encourages implementation of recommendations TR-098-MFW and E.23 for mitigation of potential overfill events.

The staff agrees with the BWOG basis concerning rejection of recommendation RR-TIR-19 since the simultaneous tripping of both MFW pumps is consistent with the goal of preventing SG overfill and is consistent with recommendation E.23. However, the staff requests that the BWOG reconsider the basis for rejection of recommendation RR-6, which provides for emphasizing the need for communications between the control room and plant staff prior to working on safety-related equipment if an unplanned actuation occurs during maintenance or testing. It appears that the potential benefit of this recommendation would warrant its approval.

Following the loss of ICS power event at Rancho Seco on December 26, 1985, the staff identified concerns regarding the FSAR Chapter 15 transient and accident analyses, which appear to assume proper operation of the non-safety-related ICS/NNI. The ICS/NNI systems requirements state that:

> The ICS/NNI system is classified as non-safety. This classification derives from the fact that ICS/NNI is in no way used to protect the integrity of the primary reactor coolant pressure boundary, to assure the capability to shutdown (SIC) the facility and maintain it in a safe shutdown condition, or to assure the capability to prevent or mitigate the consequences of accidents which could result in unplanned offsite releases in excess of the criteria established in 10 CFR 100.

BAW-1919, Appendix R-t, "FSAR Assumptions Relating to ICS/NNI," provides a table that identifies ICS/NNI actions assumed in the FSAR analyses for 15 different accident scenarios at the six operating B&W reactor facilities (Crystal River, Davis-Besse, Oconee, TMI-1, Arkansas-1, and Rancho Seco). It appears that the ICS may be assumed to function properly to help mitigate the consequences of postulated accidents for some of the scenarios. The BWOG has stated that the information provided in Appendix R-t is preliminary and that the BWOG I&C Committee will continue its evaluation in this area as a post-SPIP follow-on activity. The staff encourages the BWOG to verify that the mitigation of postulated accidents is not dependent on proper operation of the ICS/NNI.

Section 7.7, "Control Systems," of the NRC Standard Review Plan (SRP), NUREG-0800, provides review guidance concerning non-safety-related control systems to be used to determine compliance with applicable regulatory requirements. Control system designs are considered to conform to the applicable general design criteria (GDC) if all of the following conditions listed in SRP Section 7.7 are satisfied:

(1) The review should confirm that the control systems satisfy the requirements of the acceptable criteria and the system design bases.

(2) The review should confirm that the plant accident analysis in Chapter 15 of the SAR does not rely on the operability control systems to ensure safety.

(3) The review should confirm that the safety analysis includes consideration of the effects of both control systems action and inaction in assessing the transient response of the plant for accidents and anticipated operational occurrences.

(4) The review should confirm that the consequential effects of anticipated operational occurrences and accidents do not lead to control systems failures that would result in consequences more severe than those bounded by the analysis in Chapter 15 of the SAR.

(5) The review should confirm that the failure of any control system component or any auxiliary supporting system for control systems do not cause plant conditions more severe than those bounded by the analysis of anticipated operational occurrences in Chapter 15 of the SAR. (The evaluation of multiple independent failures is not intended.)

The staff encourages the BWOG to pursue the implementation of recommendations that ensure B&W ICS/NNI designs conform to the guidance listed in SRP Section 7.7.

The BWOG has proposed recommendations (C.1 and C.7) on the basis of the MPR sensitivity study discussed in Section 5 of the SER. If implemented, these recommendations would tend to separate ICS control functions to reduce the ICS contribution to transient complexity upon single failures. The staff encourages implementation of these recommendations because they will cause the ICS to respond more like separate/discrete control systems to single failures. An advantage of discrete control systems is that when a single control system fails, the other control systems, being electrically separate and independent, are not affected and, therefore, tend to stabilize overall plant conditions. In contrast, when the integrated control system fails, the effects may be fed throughout the plant, causing the overall plant conditions to degrade rapidly.

As part of the BWOG ICS/NNI reassessment, consideration was given to the use of advanced control system designs to perform functions currently performed by the ICS. The design changes being considered in recommendations TAC-6.d and TAC-6.e and the changes being considered with regard to new advanced control system designs (D.1 through D.6) should result in substantial improvement in reducing the complexity of transients caused or aggravated by the ICS/NNI system.

6.1.11 Previously Identified ICS/NNI Concerns and Recommendations

The staff's conclusions regarding the effectiveness of the BWOG SPIP ICS/NNI reassessment in resolving concerns identified during the investigation of events at B&W reactors involving ICS/NNI system failures are given below. The staff reviewed a number of documents providing the results of investigations of B&W reactor transients involving the ICS/NNI to compile a list of previously identified concerns (PICs). The documents included

- BAW-1564, "Integrated Control System Reliability Analysis," dated August 1979.

- IE Bulletin 79-27, "Loss of Non-Class 1E Instrumentation and Control Power System Bus During Operation," dated November 30, 1979.

- NUREG-0667, "Transient Response of Babcock & Wilcox-Designed Reactors," dated May 1980.

- NUREG-1195, "Loss of Integrated Control System Power and Overcooling, Transient at Rancho Seco on December 26, 1985," dated February 1986.

The staff's review of these and other documents resulted in a list of PICs. A list of the more significant PICs, many of which appeared to be common to the more severe B&W plant transients involving the ICS/NNI, is provided in Appendix E to this report. While Appendix E includes guidance/recommendations for resolving the PICs, it should be noted that many of these recommendations have not been imposed as regulatory requirements. A complete list of PICs related to the ICS/NNI, including concerns of lesser significance and/or that may be applicable to only a small number of the B&W plants, will be provided in the staff's detailed technical report on the ICS/NNI reassessment.

Following the loss of ICS power at Rancho Seco on December 26, 1985, the staff questioned why complex plant transients involving the ICS/NNI were continuing to occur given that a number of improvements to ICS/NNI systems were thought to have been made. Investigation of this event and the March 19, 1984 loss of NNI power event (also at Rancho Seco) indicated that the PICs raised from earlier loss of ICS/NNI power events (e.g., loss of ICS/NNI power at Oconee Unit 3 on November 10, 1979, and loss of NNI power at Crystal River on February 26, 1980) continued to exist at some B&W plants. From the onset of the ICS/NNI reassessment effort, the staff specifically asked the BWOG (via the letters referenced in Appendix F) to address the PICs listed in Appendix E.

In general, based on the review of information provided in BAW-1919, the staff cannot conclude that the PICs listed in Appendix E will be fully resolved through implementation of the approved recommendations in the RTS. It does appear, given implementation of the SPIP-approved recommendations, that PIC number 6, regarding the provision of backup power supplies and automatic transfer capability for power supplies to increase ICS/NNI power reliability, will be resolved. This seems to be an area of significant improvement to ICS/NNI systems over the years.

For a number of the PICs, the BWOG recommendations (approved and proposed), if implemented, would provide significant progress towards resolution. This appears to be especially true for PICs 2, 8, 12, and 19 as a result of the recommendations to dictate the failure modes of ICS/NNI-controlled equipment, thus reducing the consequences/complexity of events involving ICS/NNI power failures. The staff also believes that significant improvements concerning resolution of PICs 10, 11, 14, 17, 18, 22, and 24 will be achieved if the BWOG I&C

Committee recommendations (approved and proposed) are properly implemented. Conversely, based on the information provided in BAW-1919, it is not apparent that PICs 1, 4, 5, 13, 16, 20, and 23 have been/will be resolved upon implementation of the BWOG recommendations. The staff's review concludes that insufficient information is provided in BAW-1919 to make a determination concerning resolution of PICs 3, 7, 9, 15, 21, and 25.

The overall staff conclusion regarding resolution of the PICs listed in Appendix E through the BWOG SPIP is that the combination of approved and proposed recommendations, if implemented, would help to resolve a number of these concerns. Therefore, the BWOG SPIP has been successful to a significant degree in achieving its goal of a reduction in the frequency of reactor trips and in the complexity of transients involving ICS/NNI-related failures. However, the staff remains uncertain regarding the resolution of a number of PICs. In addition, some of the PICs appear not to have been resolved by the BWOG SPIP.

6.1.12 Instrumentation and Control System Issues Not Directly Related to ICS/NNI

The staff's review and evaluation of instrumentation and control (I&C) systems addressed by the BWOG that are not directly related to the ICS/NNI are given below. BWOG recommendations in three areas were reviewed: (1) systems designed to isolate MFW and/or AFW flow to a depressurized steam generator (i.e., rupture control systems), (2) main feedwater system related I&C issues, and (3) the main turbine system electrohydraulic control (EHC) system. The following recommendations were reviewed:

Number	Recommendation
TR-014-MFW*	Install monitoring system on MFW pumps to document causes of pump trips.
TR-016-MFW*	Investigate oil system pressure in MFW pump.
TR-017-MFW*	Evaluate MFW pump control systems.
TR-019-MFW	Ensure there are sufficient annunciator and trip signals for MFW supply system.
TR-021-ICS*	Identify causes for MFW pump control problems.
TR-025-MTS*	Review EHC system for loss of input power.
TR-052-SFI*	Filter steam generator level signals in steam feedwater rupture control system.
TR-066-MFW*	Ensure that a single electrical failure will not cause a loss of both feedwater trains.
TR-067-MFW*	Wherever possible eliminate automatic MFW pump trip functions.

*The BWOG has identified these recommendations as key recommendations.

Number	Recommendation
TR-069-MFW*	Eliminate automatic control of the MFW block valve except during a reactor trip.
TR-070-MFW*	Provide capability of override a "close" signal to the MFW block valve.
TR-071-MFW	Install valve position indication for the startup and MFW regulating valves.
TR-072-MFW	Eliminate the transfer from the startup to the MFW flowmeter when the MFW block valve opens.
TR-085-MFW	Modify main FW pump recirculation valve for automatic control during startup and shutdown.
TR-088-MFW	Eliminate automatic plant runback on low MFW pump discharge pressure or establish set point to achieve a successful runback.
TR-091-MFW	Eliminate need for an auxiliary operator to open a deaerator feedwater tank drain line after reactor trips (for Davis-Besse only).
TR-100-MTS*	Review MSR drain tank level control and drain line configuration.
TR-169-MTS	Evaluate possibility for defeating the high vibration trip during main turbine valve testing (for GE turbines only).
TR-170-MFW	Evaluate placing orifice snubbers in the MFW pump control oil system.
TR-173-MFW	Ensure in procedures that MFW pump status to ARTS/RPS is reset after each MFW pump is operational.
TR-180-MTS	Provide a monitoring capability for the EHC system for purpose of root cause determination.
TR-200-MTS	Install a time delay relay or an orifice between the EHC oil system and the ARTS sensing line to prevent oil pressure perturbations.
TR-201-MTS	Review EHC overspeed and fast control and intercept valve circuits.
TAC-1.f	The BWOG I&C Committee should evaluate a design modification to automatically reset the ARTS bistables on turbine reset.
TAC-2.d	Expand RTS recommendation TR-067-MFW to include the condensate pumps and condensate (or feedwater) booster pump. Also, modify the wording of TR-067-MFW to harden those trip signals that are required for pump/plant protection.

*The BWOG has identified these recommendations as key recommendations.

Number	Recommendation
TAC-2.i	The BWOG I&C Committee should evaluate implementing a design modification to automatically reset the ARTS bistables on MFW pump reset.
RR-2.1.7	Although each utility has taken action to reduce the propensity for turbine overspeed trips, further actions are recommended.
RR-11	Provide indication of the status of the MFW pump and turbine trip bistables into ARTS/RTS in the main control room.
RR-TIR-19	The plants that trip both MFW pumps on high SG level should evaluate sequential vs. simultaneous MFW pump trip.

The BWOG has identified the MFW system as one that has contributed significantly to the frequency of reactor trips and that has been frequently involved in post-trip complications, especially with regard to primary system overcooling. Reactor trips and post-trip complications have been caused by problems/failures involving the MFW pumps and MFW system valves and flow control. Problem areas identified include instrumentation and control circuit malfunctions, spurious pump trips, and maintenance- and procedure-related concerns. EHC system failures also have contributed to the frequency of reactor trips and post-trip transient complexity. EHC system problems have included pump/oil system failures, control circuit malfunctions, and power supply reliability.

The staff generally agrees with and encourages implementation of the above recommendations, which have the potential to significantly reduce the frequency of reactor trips and the complexity of post-trip plant behavior. It is noted that recommendations TR-067-MFW and RR-TIR-19 are discussed in greater detail in Sections 6.1.9 and 6.1.10, respectively.

The BWOG investigation of MFW pump trips found that the root cause was listed as "not specified" or "unknown" in 40 percent of the cases. This was the basis for recommendation TR-014-MFW, which calls for installation of a system to monitor the causes of MFW pump trips. It is expected that such a system could provide the exact causes for MFW pump trips and eventually lead to plant modifications designed to reduce MFW pump trips and hence lower their contribution to the frequency of reactor trips. Therefore, the staff agrees with the BWOG designation of TR-014-MFW as a key recommendation and encourages implementation of recommendation TR-180-MTS concerning similar monitoring for the EHC system.

Steam generator rupture control systems and AFW/EFW initiation systems at B&W plants have been found to be susceptible to spurious actuations caused by perturbations/noise (e.g., main steam line pressure oscillations from turbine stop valve fast closure) that affects the associated steam generator level sensing instrumentation. The BWOG has recommended the use of time delay circuits or mechanical damping to reduce the impact of short duration pressure oscillations internal to the instrument sensing lines. Recommendation TR-052-SFI addresses steam generator level sensing instrumentation, and recommendations TR-170-MFW and TR-200-MTS similarly address MFW and EHC system oil pressure sensing instrumentation. The staff has previously approved similar techniques and considers them acceptable and beneficial to preventing spurious trips/actuations.

Item II.K.2.10, "Safety-Grade Anticipatory Reactor Trip," of NUREG-0737, "Clarification of TMI Action Plan Requirements," requires installation of a safety grade anticipatory reactor trip system (ARTS) to initiate reactor trip upon a loss of MFW and upon trip of the main turbine. Recommendations TR-173-MFW, TAC-1.f, TAC-2.i, and RR-11 are related to the ARTS. Recommendations TAC-1.f and TAC-2.i are for the BWOG I&C Committee to evaluate design modifications that would automatically reset the ARTS bistables following a reactor trip upon reset of the MFW pumps or main turbine. In general, the concept of automatic reset of protective actions/functions is inconsistent with the intent of Section 4.16, "Completion of Protective Action Once It Is Initiated," of IEEE Standard 279 "Criteria for Protection Systems for Nuclear Power Generating Stations." The design of a circuit modification to provide automatic reset of the ARTS bistables would have to be such that its failure would not prevent reactor trip on MFW pump trip or turbine trip when required. The only approved recommendation in the RTS regarding the ARTS is TR-173-MFW, which calls for a procedure to be used to ensure the MFW pump trip ARTS bistables are properly reset when the MFW pumps are returned to service. Using procedures to reset the ARTS bistables would seem to be adequate since there have been relatively few reactor trips caused by the failure to reset. The staff discourages the pursuit of design modifications for automatic reset of the ARTS.

The BWOG has rejected recommendations RR-2.1.7 and RR-11 related to the reduction of the frequency of reactor trips. The staff believes that implementation of these recommendations would provide some benefit; however, not implementing these recommendations does not appear to significantly increase the potential for reactor trips. Since it is not clear that the cost-benefit of implementation would be worthwhile, the staff does not disagree with the BWOG's basis for rejection. The staff strongly encourages implementation of recommendations TR-067-MFW and TR-071-MFW and agrees that recommendations TR-067-MFW and TAC-2.d are appropriate and should be implemented.

6.1.13 Conclusions

The staff concludes that the BWOG I&C Committee established a sound methodology for review and evaluation of ICS/NNI designs at operating B&W reactors and that the methodology is appropriate for achieving resolution of ICS/NNI concerns. The BWOG developed ICS/NNI system requirements (largely based on lessons learned from ICS/NNI operating history), compared the existing as-installed ICS/NNI designs to the system requirements, and proposed recommendations considered necessary to achieve conformance of the installed designs to the systems requirements. Implementation of the recommendations is intended to achieve the desired reduction in reactor trip frequency and post-trip transient complexity that result from ICS/NNI-related failures at B&W plants.

The staff considers the development of upgraded/revised ICS/NNI systems requirements to be a significant step of the ICS/NNI reassessment. The staff was encouraged that some of the systems requirements take the form of acceptance criteria. The staff concludes that, if existing ICS/NNI designs are modified to achieve conformance with the systems requirements, many of the problem areas identified from the investigation of B&W reactor transients involving ICS/NNI-related failures would be resolved. However, the staff believes that improvements in the ICS/NNI systems requirements are needed in several areas and that development of ICS/NNI system design-basis documents by the BWOG would be useful to support the systems requirements documents. Some of the recommendations resulting from the comparison effort are too general and may not refer to sufficient source documents (that provide the basis/background for the recommendation) to ensure that implementation will satisfy the intended ICS/NNI reassessment program objectives, or achieve conformance with the systems requirements. Where appropriate, the BWOG should upgrade the references for the recommendations.

From the onset of the BWOG SPIP, through a series of meetings with the BWOG, and through a number of letters to the BWOG (the letters are listed in Appendix F), the staff has emphasized that an objective of the SPIP reassessment should be to resolve the concerns identified from previous B&W reactor transients involving ICS/NNI-related failures. The more significant concerns are listed in Appendix E. The staff concentrated its review on resolution of the PICs because the loss of ICS power event at Rancho Seco on December 26, 1985, demonstrated that many of the PICs continued to exist despite past efforts to resolve them. Many of the concerns have been found to be common to the more severe transients at B&W reactors involving ICS/NNI power losses. The staff concludes that insufficient information was provided in BAW-1919 for the staff to make a positive determination concerning resolution of the majority of PICs. The BWOG provided responses to several of the staff letters concerning resolution of PICs in Appendix R-m, "Responses to NRC Questions Involving the ICS/NNI Evaluation," but many of the responses do not directly address the specific concerns and/or do not provide sufficient information for the staff to conclude that certain PICs have been resolved. The staff impression was that the BWOG was focusing its efforts toward resolution of the PICs. However, recent conversations with the BWOG indicate that it was never the intent of the SPIP to investigate the PICs because they were thought to have been resolved and to reinvestigate them was not deemed necessary.

The BWOG SPIP has made significant progress toward resolution of many of the PICs (see Section 6.1.11 of this report); therefore, the ICS/NNI reassessment program should prove successful in accomplishing its goal of achieving a reduction in reactor trip frequency and transient complexity. The degree of reduction is difficult to assess. It must be realized that the reassessment effort has resulted in an overall improvement in ICS/NNI designs and that the design and adequacy of B&W plant safety-related systems to mitigate the consequences of accidents and transients has not been questioned (with the exception of concerns regarding AFWS initiation and control, which are now considered to be resolved).

An important concept evaluated during the SPIP reassessment effort that has significant potential with regard to limiting the consequences of events involving ICS/NNI power losses, is that of "known safe state" (KSS). The BWOG has identified recommendation TR-178-ICS, which recommends that each B&W plant ensure that a KSS is attained on the loss of ICS/NNI power, as a key recommendation approved for implementation. Although it is not clear to the staff, based on a review of BAW-1919, that a KSS can be achieved following ICS/NNI power losses without reliance on substantial operator actions in excess of those normally required for other reactor trips, the staff concludes that implementation of the approved and proposed ICS/NNI-related recommendations will make it easier to attain a KSS following power losses. The staff encourages the BWOG to continue to pursue actions to achieve the KSS as defined in BAW-1919. Implementation of the recommendations to dictate equipment failure modes on loss of ICS/NNI power to reduce the consequences/complexity of such events and to retain remote manual control of ICS/NNI-controlled critical components from the control room following power losses are strongly encouraged by the staff. The staff concurs with the approach to implement recommendations to reduce the consequences of ICS/NNI power losses as well as trying to prevent the losses by increasing power supply reliability. ICS/NNI power losses have continued to occur despite past efforts to increase power supply reliability.

The Abstract to NUREG-1195 includes the following statement:

> The fundamental causes for this transient were design weaknesses and vulnerabilities in the ICS and in the equipment controlled by that system. These weaknesses and vulnerabilities were not adequately compensated by other design features, plant procedures or operator training. These weaknesses and vulnerabilities were largely known to SMUD and the NRC staff by virtue of a number of precursor events and through related analyses and studies. Yet, adequate plant modifications were not made so that this event would be improbable, or so that its course or consequences would be altered significantly. The information was available and known which could have prevented this overcooling transient; but in the absence of adequate plant modifications, the incident should have been expected.

Based on the review of information provided in BAW-1919, the staff concludes that in the absence of documented resolution of the PICs listed in Appendix E, some weaknesses and vulnerabilities in ICS/NNI designs probably continue to exist and losses of ICS/NNI power should still be expected to occur. However, in light of the apparent progress made in resolving some of the PICs, the staff concludes that the consequences of ICS/NNI power losses should not be as severe if the BWOG SPIP recommendations are implemented. Areas of concern that the staff considers to remain unresolved upon completion of the SPIP include

- implementation of modifications that adequately address the concerns of IE Bulletin 79-27
- ICS/NNI power supply monitor (PSM) design and application
- adequacy of backup instruments and controls following ICS/NNI power losses
- mid-scale failures of control room instrumentation
- operator actions required to achieve KSS following ICS/NNI power losses
- FSAR transient and accident analysis assumptions concerning the ICS/NNI

The basis for the staff's conclusion that certain PICs remain unresolved and that a conclusion could not be reached concerning resolution of other PICs, is that the concerns were not addressed in sufficient detail in BAW-1919. The BWOG considered several of the PICs to be plant specific (e.g., review of concerns identified in IE Bulletin 79-27); thus they were not addressed by the SPIP. The staff concludes that plant-specific reviews and site audits should be performed to obtain the information necessary to address resolution of the PICs as part of the implementation audits discussed in Section 12.2. The staff believes that with proper implementation of the BWOG recommendations, along with resolution of the concerns identified in IE Bulletin 79-27, the B&W plant response to ICS/NNI failures will be significantly enhanced and will ensure that adequate safety margins exist for B&W plants.

6.3 Auxiliary Feedwater/Emergency Feedwater (AFW/EFW) System Review

6.3.3 Acceptability of Emergency Feedwater Initiation and Control System for Each B&W Plant

In the SER, the NRC staff stated that it would address the acceptability of the AFW initiation and control system for each B&W plant in a supplement. The staff also will address details such as non-automatic response to steam and feedwater line breaks.

The NRC staff has completed its review of each B&W plant's AFW initiation and control system. The review included the following plants: Arkansas Nuclear One (ANO-1), Crystal River, Rancho Seco, Three Mile Island Unit 1 (TMI-1), Davis-Besse, and Oconee Units 1, 2, and 3. A general description of each different type of AFW initiation and control system and its acceptability are discussed below under the applicable B&W plant(s).

6.3.3.1 ANO-1, Crystal River, and Rancho Seco

The emergency feedwater initiation and control (EFIC) system is the initiation and control system for ANO-1, Crystal River, and Rancho Seco. The AFW/EFW system is designed to provide secondary coolant to the once-through steam generators (OTSGs) if the main feedwater (MFW) system becomes unable to perform this function or if AFW is needed to promote natural circulation in the reactor coolant system. The EFIC system is designed to automatically initiate AFW and to control the level in the OTSGs. When monitored plant parameters indicate the need for it, the system may be initiated manually at the discretion of the operator.

The EFIC system is designated as Class 1E equipment and is contained in cabinets that are physically and electrically isolated from one another. Each channel receives analog inputs from the OTSG level and steamline pressure transmitters associated with each OTSG. The level signals are compensated, as necessary, to reflect true water level. The EFIC system also receives initiation signals from the reactor protection system (RPS) and the safety features actuation system (SFAS). During plant operation, the EFIC system constantly monitors these input signals. An AFW initiation signal is generated when a process parameter exceeds its set point value, but an actual actuation will take place only if at least two channels issue initiating commands.

The EFIC system is designed to initiate AFW (1) on low water level in either OTSG, (2) on low pressure in either OTSG steam line, (3) when all four reactor coolant pumps trip (received from the RPS), (4) on loss of both MFW pumps at greater than 20 percent power (received as an anticipatory trip on loss of MFW pumps from the RPS), (5) on high reactor building pressure, or (6) on low reactor coolant system pressure. The EFIC system receives high reactor building pressure and low reactor coolant system pressure initiation signals from the safety features actuation system (SFAS). High building pressure is used to initiate AFW because it indicates a possible main feed line break, a main steam line break, or a loss-of-coolant accident inside containment.

The EFIC system actuation logic is arranged in a 1-out-of-2, taken twice logic. All four EFIC channels provide AFW initiation commands to the AFW trip logic modules. These trip logic modules are physically located in the A and B EFIC channel cabinets.

The EFIC system includes logic used to isolate AFW flow to a ruptured or depressurized OTSG. This logic is referred to as the feed only good generator (FOGG) or "vector" logic. Upon actuation, the vector logic precludes the continued addition of AFW to a depressurized OTSG; thus minimizing the overcooling effects of a steam leak. The vector logic may isolate AFW to one OTSG only, never to both.

The EFIC system also will isolate MFW flow to an OTSG when either a pressure of less than 600 psig or a high water level is detected in that OTSG. Four redundant instrument channels, A, B, C, and D, are provided to monitor each of these parameters for each OTSG. The EFIC system MFW isolation logic is arranged in a 1-out-of-2 taken twice logic identical to the AFW system actuation logic.

In the event of an AFW initiation, the operator may gain control of the system pumps and isolation valves by pressing the Manual Permissive button for the respective AFW train (A or B). This allows the operator to take manual control of the actuated equipment to optimize response to the existing condition, that is, shut down pumps and control valves, as required to handle the situation.

If the condition clears that caused the initiation while operating in the manual mode after an AFW initiation, the system will revert back to the automatic mode of operation and will re-initiate AFW if a condition occurs that requires it.

The AFW trains can be manually actuated by the operator pressing both AFW Initiate buttons on the control panel. The system will respond in exactly the same way as if it were automatically actuated.

A more detailed description of the AFW EFIC system, as well as plant-specific information (e.g., pump and valve nomenclature and set points) is contained in the respective NRC staff's plant-specific SERs.

The staff has reviewed and evaluated the AFW EFIC system for ANO-1, Crystal River, and Rancho Seco, and has concluded that each is acceptable. Plant-specific SERs were issued to each licensee in letters dated July 13, 1982, September 17, 1982, and November 10, 1987, respectively.

6.3.3.2 TMI-1

The AFW initiation system at TMI-1 is designed to (1) initiate the start of the AFW system's two motor-driven pumps and one turbine-driven pump, which provide the in-flow of water to the OTSGs; (2) initiate automatic control to maintain the required water level in each OTSG; and (3) provide a manual control capability. The system is capable of withstanding a design-basis event and a single active failure while supplying a heat removal path to allow safe shutdown of the reactor. The design modification also ensures that a single active failure will never inadvertently initiate the AFW system nor isolate the MFW system. The licensee performed these modifications before startup following the Cycle 6 refueling outage, which occurred in November 1986.

The initiation and control of the AFW system is performed by the heat sink protection system (HSPS). The HSPS is designed to initiate the AFW system on low water level in either OTSG; on high containment pressure, when both main feedwater pumps trip; or when all four reactor coolant pumps trip. Indication of high containment pressure provides an alert of a possible main feed line break, a main steam line break, or a loss-of-coolant accident inside containment. This indication, along with an indication of low water level in the OTSGs, would indicate a possible failure of the secondary heat sink.

The initiating signals for the main steam line rupture detection circuitry portion of the AFW system originate in the HSPS. The HSPS is designed to initiate MFW and AFW isolation on low pressure in either OTSG main steam line supply to the turbine generator while supplying water to the "good" generator.

The sensing portion of the HSPS for steam generator level and containment pressure consists of four independent channels, with a two-out-of-four actuation logic. This arrangement is designed to withstand a single failure in any single channel simultaneously with another channel bypassed for maintenance, test, or repair. The actuation portion of the HSPS electronics consists of two independent trains. The trains are designed such that a single failure of either train will not prevent at least one train of the AFWS from operating.

The automatic initiation signals from the sensors to the control circuitry and the control circuitry for the AFW system at TMI-1 comply with the general functional requirements of IEEE Standard 279-1971 with regard to the requirement to automatically initiate appropriate protective action whenever a condition monitored by the system reaches a preset level. The main steam line rupture detection circuitry associated with the HSPS was reviewed to the extent necessary to ensure proper separation and isolation from the AFW system initiating circuitry portion of the HSPS.

The automatic initiation circuitry from the sensors to the control circuitry and the control circuitry of the AFW system was reviewed to determine compliance with the single-failure criterion of IEEE Standard 279-1971. The criteria of paragraph 4.17 of IEEE Standard 279-1971 specify that the protection system shall include means for manual initiation of each protective action at the system level. The only system level protective action for the AFW system is AFW injection, which requires starting the AFW system's pumps and opening the control valves. This action can be performed by overriding the automatically selected set points at each of the AFW system's controller display stations in the control room. By switching the controllers to manual control, the operator can initiate the automatically initiated circuitry, which starts the AFW pumps and opens the control valves. These controllers and appropriate indicators are physically located next to each other in the control room so that the operator can take control quickly and efficiently with the minimum number of operations. The staff's review of the AFW injection design provided assurance that the present means of manual initiation meets the criteria of paragraph 4.17.

The system design at TMI-1 provides for four separate automatic AFW initiation signals. Three of these signals meet all the requirements of IEEE Standard 279-1971. TMI-1 requested a deviation from IEEE-279 with regard to the fourth signal, for which manual removal of a bypass of the signal was proposed.

For this fourth signal, the licensee believed it would be preferable to have the results of operating experience at low steam generator levels before establishing the set point level that would trigger automatic initiation of the AFW system. This would prevent numerous inadvertent initiations of the AFW system. Therefore, the licensee proposed a design in which automatic initiation of the AFW system on low level was bypassed when the plant was operating below 30 percent power, in which case it was proposed that the system would be actuated manually. The licensee also proposed that this bypass be removed manually when the plant exceeded 30 percent power.

In a letter dated February 18, 1987, the NRC staff told the licensee that manual removal of the bypass was not acceptable because it did not meet the guidelines of IEEE Standard 279-1971, which call for automatic removal of operating bypasses when permissive conditions are not met. In addition, the licensee had not provided adequate justification for the system proposed.

However, as an interim resolution to this problem, the NRC staff, by issuance of License Amendment No. 124, has permitted manual removal of the bypass until the end of Cycle 6 operation. At that time, the AFW system will be modified to delete manual removal of the bypass of the fourth initiation signal and will be in full compliance with IEEE Standard 279-1971, the licensee has provided adequate compensating features to ensure that bypass removal will function properly.

The licensee agreed to implement design and procedural controls to ensure that the bypass switch is in the appropriate position. First, when the bypass is actuated, an alarm indicates in the control room. Second, the control room operator must acknowledge, by signature, that the bypass status has been checked and that the bypass has been removed when the reactor power is increased above 30 percent power.

In approving this design in Amendment No. 124, the NRC staff imposed additional restrictions on the licensee. Use of this manually removed bypass is restricted to the controlled conditions of a normal reactor startup or shutdown during Cycle 6 operation only. Thus, this bypass will be used very infrequently and only during a very short period (hours) following a decision to either start up or shut down the unit. Operation at power levels above 30 percent is the normal and more frequent mode of operation, and the bypass is always removed during this mode of operation. For this limited period, the use of these special procedural controls is adequate to satisfy the basic purpose of NUREG-0737.

The NRC safety evaluations and inspections indicate that a safety-grade, highly reliable AFW system has been installed and is operational at TMI-1. A system that fully meets the guidelines of IEEE Standard 279-1971 will be provided by the end of Cycle 6. During Cycle 6, compensating features to satisfy the intent of NUREG-0737 will be provided.

On the basis of the NRC staff's review of the TMI-1 EFW automatic initiation system, the licensee's response to the requirements of Item II.E.1.2 is adequate to determine compliance to the criteria of IEEE Standard 279-1971 and, therefore, is acceptabl

6.3.3.3 Davis-Besse

The AFW system at Davis-Besse is designed to supply an independent source of feedwater to the steam generators when the normal feedwater system is not available to maintain the heat sink capabilities of the the steam generators. The primary sources of water for the AFWs are the two, non-safety-related condensate storage tanks (CSTs). The secondary source of water for the turbine-driven AFW pumps is the safety-grade service water system with an automatic switchover from the CST on low suction pressure at the pumps. The AFW system is an engineered safety feature system that is relied upon to aid in preventing core damage in the event of transients such as a loss of normal feedwater, a steam system pipe rupture, or a small-break loss-of-coolant accident.

The system consists of two redundant safety-related essential trains, each with its own steam-turbine-driven pump, associated valves, piping, controls, and instrumentation. A non-safety-related, motor-driven feedwater pump (MDFWP), associated valves, piping controls, and instrumentation also are able to provide flow equivalent to one AFW pump to either steam generator. Each of the AFW pumps is capable of supplying water to either or both steam generators, as is the MDFWP. Each AFW pump has a design flow of 1050 gpm (which includes 250 gpm minimum recirculation) at 1050 psig. One turbine-driven AFW system train is completely independent of ac power. Each of the AFW supply paths (including the MDFWP) to the steam generator contains two check valves and a motor-operated isolation valve. The flowpath from the MDFWP includes two check valves and a flow control valve. On initiation of the AFW system, the isolation valve will be automatically opened.

The turbine-driven AFW pumps are initiated on low steam generator level, low steam generator pressure, loss of the four reactor coolant pumps, high steam generator level, and high differential pressure between the main feedwater line and steam line. Manual initiation is accomplished by operator action in the control room.

The MDFWP is operated from the control room with non-safety-grade controls and instrumentation. The MDFWP and its associated equipment are normally aligned to receive power from one diesel generator. Operating this equipment with power from the other diesel generator requires some operator action from outside the control room. The licensee has committed to making the necessary modifications to permit aligning all necessary MDFWP-associated equipment to either diesel generator from within the control room before Cycle 6.

The steam and feedwater rupture control system (SFRCS) is designed as an engineered safety features system to monitor plant parameters such as steam generator water level and pressure, differential pressure between the steamline and main feedwater line for each steam generator, and the loss of all four reactor coolant pumps. Under plant conditions indicative of a main steamline break, main feedwater line break, or loss of heat sink, the SFRCS initiates appropriate actions to isolate a ruptured steam generator and initiate AFW system flow to the intact steam generator(s). Valves controlled by the SFRCS to isolate a ruptured steam generator include the main steam isolation valves (MSIVs), the MFW regulating startup valves, and the AFW system containment isolation valves. The SFRCS also controls the AFW system steam admission and pump discharge valves.

The NRC staff's preliminary review of the SFRCS design, following the event on June 9, 1985, concluded that the SFRCS was unacceptable because it was not capable of performing its required safety functions (providing AFW flow to the intact steam generator) following a design-basis event and a single active failure. Furthermore, the staff raised concerns regarding the SFRC's capability to cut off all sources of feedwater to both steam generators, requiring operator intervention and successful operation of several active components to re-establish core cooling.

The licensee performed a single-failure analysis of the SFRCS to ensure that for each analyzed event, given any credible active single failure, AFW would be available to the intact steam generator. On the basis of the results of the licensee's analysis and the short-term modifications to the SFRCS to resolve the single-failure concerns identified in NUREG-1154 with respect to reopening an AFWS containment isolation valve to feed an intact steam generator, the staff concluded that the design of the SFRCS was acceptable to allow plant restart. The staff also concluded that the short-term modifications to the SFRCS were sufficient to resolve staff concerns regarding SFRCS isolation of all sources of feedwater to both steam generators. The NRC staff's SER, NUREG-1177, dated June 1986, delineates the SFRCS and the staff's conclusion in more detail.

6.3.3.4 Oconee

The AFW system at Oconee consists of two motor-driven pumps and one turbine-driven pump. The motor-driven pumps and the turbine-driven pump start automatically on low main feedwater pump discharge pressure or low main feedwater pump control oil pressure. The AFW system is capable of feeding to either or both steam generators under automatic or manual initiation and control.

The automatic initiation signals and circuitry for AFW at Oconee comply with the general functional requirements of IEEE Standard 279-1971 with regard to the requirement to "automatically initiate appropriate protective actions whenever a condition monitored by the system reaches a preset level." The use of low main feedwater pump discharge pressure or low main feedwater pump control oil pressure signals in the AFW system initiation logic represents an anticipatory loss of feedwater. This design feature permits automatic initiation of emergency feedwater in a more timely manner to reduce the likelihood of steam generator dryout. The anticipatory loss of feedwater initiation is detected by pressure switches sensing diverse variables.

Unlike the other B&W plants, Oconee has no automatic features for isolation of a depressurized steam generator in the event of a main steam or feedwater line break. Following such events, operator action must be taken to isolate MFW or AFW to the affected generator and to ensure adequate AFW to the intact steam generator. These actions can be accomplished by manual action within the control room. Emergency Operating Procedure EP/1/A/1800/01 directs the control room operator in the necessary steps to isolate the affected steam generator.

In response to IE Bulletin 80-04, issued February 7, 1980, analyses have been performed for Oconee, which assume blowdown of both steam generators prior to operator action being taken and no ICS action to isolate the MFW. These analyses confirm that core cooling will be maintained and that the main steam and feedwater line break consequences, such as peak containment pressure and reactor vessel overcooling, are maintained within acceptable limits. Thus, the staff concluded that automatic isolation of the steam generator is not required. Further details and the bases for the staff's conclusions are provided in the staff's October 14, 1982 safety evaluation of Oconee (letter to H. B. Tucker, Duke Power, from J. Stolz, NRC) and in Chapter 15.3 of the Oconee FSAR.

The automatic initiation signals and circuitry used at Oconee comply with the single-failure criterion of IEEE Standard 279-1971. The scope of the single-failure analysis was limited to the emergency feedwater initiation circuitry, electrical power sources, and control systems. No single failure within the manual or automatic initiation systems prevents initiation of protective action by manual or automatic means.

As stated in the NRC letter dated June 3, 1981, the NRC staff concluded that NUREG-0737 Item II.E.1.2 had been adequately resolved for Oconee.

6.3.3.5 Conclusion

The staff has reviewed and evaluated each B&W plant for conformance to NUREG-0737, Item II.E.1.2, Parts 1 and 2, "Auxiliary Feedwater System Initiation and Flow Indication," and concludes that each is acceptable. In addition, the staff concludes that the emergency feedwater system does not rely on the non-safety-grade ICS to fulfill its safety function.

7 EVALUATION OF HUMAN FACTORS ISSUES

7.7 Response to BWOG Comments on SER

In its letter dated December 21, 1987, the BWOG provided comments to the NRC staff on the staff's SER. The NRC staff has evaluated these comments and determined that, with the exception of one comment pertaining to the further evaluation of the six identified concerns related to the human factors (contained in Section IV of Appendix K of BAW-1919), the BWOG has not provided any new or additional information that was not known or factored into the staff's review at the time of the SER was written.

Based on the BWOG letter, it appears that the staff and the BWOG are in mutual agreement that human factors are important. The BWOG has acknowledged this by its commitment to consider human factors in the future. However, whereas the staff believes that human factors are sufficiently important to warrant the participation of personnel trained and experienced in the discipline of human factors engineering in multidisciplinary teams of investigators, the BWOG takes the position that it is sufficient for engineers to consider human factors in the course of their traditional activities.

Although the BWOG has committed to the future consideration of human factors, adequate details were not provided for the staff to assess the acceptability the BWOG actions. Therefore, the BWOG's response to the SER has not changed the staff's evaluation of the BWOG efforts in the area of human factors as described in Section 7 of the SER.

9.6 Risk Recommendations/Relationship to SPIP

As part of the BWOG Safety and Performance Improvement Program (SPIP), the Owners Group has recommended a number of plant modifications intended to reduce the frequency and severity of transients at B&W plants. These recommendations are documented in Appendix J to the SPIP report (BAW-1919, Revision 4, April 1988). In the SER these recommendations were assessed to identify those that would help to reduce the frequency of core damage resulting from the Category C events. The NRC staff also has examined the potential risk benefit achievable from selected recommendations; this is discussed below. Since the details of the actual plant implementation of the recommendations are not known, the actual plant-specific benefit cannot be estimated. Rather, the maximum possible risk benefit was estimated assuming that the recommendation was properly developed and implemented. This helps provide a perspective on what may be the more effective areas for action and potential benefit.

Since the time of this evaluation, Appendix J has been updated to include additional recommendations approved by the BWOG and to identify recommendations that have been rejected or superseded. These recent updates to Appendix J have not been factored into this evaluation.

9.6.1 Areas Covered by the BWOG Recommendations

Each of the 154 BWOG recommendations identified in Revision 4 of the SPIP report was reviewed and separated into one of the following 12 categories identified as being meaningful to the evaluation.

(1) improvements affecting integrated control system/non-nuclear instrumentation
(2) improvements affecting main feedwater
(3) improvements affecting instrument air
(4) improvements in plant operations
(5) improvements affecting main steam
(6) improvements affecting plant electrical supply
(7) improvements affecting motor-operated valves
(8) improvements affecting plant administration
(9) improvements affecting the main turbine systems
(10) improvements affecting main steam/feedwater isolation
(11) improvements affecting emergency feedwater
(12) improvements affecting the reactor protection system

Table 9.1 of this supplement identifies the specific recommendations discussed in this evaluation by number and title, as assigned in the SPIP report, and category.

9.6.2 Category C Core Damage Reduction Potential of the BWOG Recommendations

In this section, the staff identifies the BWOG recommendations that could potentially reduce the contribution of Category C events to core damage frequency.

Based on the results presented in Section 9 of the SER, it is possible to identify the improvement areas that will most likely produce reasonable reductions. It is very important to note that only the effect of these improvements on the Category C core damage frequency is evaluated in this section. Other benefits that may be accrued to other parts of the risk profiles of these B&W plants are not within the scope of this study to consider, nor are they even possible to evaluate using the models or information compiled in the analysis conducted within the context of this study.

The first area to be considered is failures in the integrated control system (ICS). In Section 9 of the SER, the staff illustrated that these failures were the dominant contributor to core damage from Category C events in the B&W plants studied. Further, in looking at just what are the causes for these failures being so dominant, it was clear that reasonable core damage reductions could not be achieved without making improvements in the ICS area. To summarize the results for the four plants where Category C events can be considered to be significant contributors to core damage frequency, the contribution from loss of ICS versus total Category C contribution is as follows:

Crystal River 3:	1.3E-5 out of 2.5E-5, or about 50%
Davis Besse:	8.4E-6 out of 1.2E-5, or about 70%
ANO-1:	1.7E-5 out of 3.0E-5, or about 55%
Rancho Seco:	1.0E-5 out of 1.5E-5, or about 65%

For the two plants where Category C events are not considered to be significant contributors to core damage frequency, the results are as follows:

Oconee:	4.6E-6 out of 6.0E-6, or about 75%
Three Mile Island:	5.7E-7 out of 1.9E-6, or about 30%

The reason for these results is twofold: (1) ICS failure causes loss of main feedwater and also may induce other effects on the secondary systems that increase the severity of events, and (2) ICS failure causes substantial instrument upset that, even though backed up by safety-grade instrumentation, can result in a substantial amount of operation confusion, increasing the chance of operator error.

The BWOG recommendations include a large set of improvements in the ICS area; Table 9.1 lists those that, if properly implemented, could have a significant effect on reducing core damage from ICS upset events. In particular, if recommendations TR-001-ICS, TR-002-ICS, TR-004-ICS, TR-013-ICS, TR-038-ICS, TR-102-ICS, and TR-104-ICS were implemented, they will have the most significant effect on reducing the number of plant trips resulting from failures in the ICS (a significant reduction in initiating event frequency). Recommendations TR-033-ICS and TR-036-ICS will reduce the severity of the effect of the loss on plant systems (such as main feedwater). Recommendation TR-035-ADM, if effectively implemented, will largely reduce the operator confusion problem for the remaining occurrences of total system upset. In addition, TR-017-MFW will enhance feedwater availability during loss of ICS and TR-060-OPS will further enhance operator response during loss of ICS.

The combination of these effects could be such that the loss of ICS core damage contribution, presented above, would be made so small as to no longer be a dominant contributor to the Category C core damage frequency. The maximum amount of the reduction would, for all intents and purposes, be the total percentage contribution shown above for the present plant configurations, meaning that the expected Category C frequencies would be:

Crystal River 3:	1.2E-5
Davis Besse:	3.6E-6
ANO-1:	1.3E-5
Rancho Seco:	5.0E-6
Oconee:	1.4E-6
Three Mile Island:	1.3E-6

Thus, in addition to Oconee and Three Mile Island, two more plants (Davis Besse and Rancho Seco) would have the contribution of Category C events to core damage frequency reduced to the level of not being significant by effectively implementing those recommendations discussed above.

Following implementation of the ICS recommendations, the next logical area to consider for possible plant improvement (based on staff and contractor results) is in the reliability of main feedwater and/or bleed-and-feed capability. Failure of these functions also was illustrated in the SER to be a dominant contributor to core damage from Category C events. For the two plants where Category C events would still be considered to be significant contributors to core damage frequency, the contribution from total loss of feedwater and bleed-and-feed versus total Category C contribution, assuming the ICS recommendations are implemented in a manner to achieve minimum Category C frequency, is as follows:

Crystal River 3:	4.2E-6 out of 1.2E-5, or about 35%
ANO-1:	1.0E-5 out of 1.3E-5, or about 75%

For the four plants where Category C events now would not be considered to be significant contributors to core damage frequency, the results are as follows:

Davis Besse:	3.6E-6 out of 3.6E-6, or about 100%
Rancho Seco:	3.3E-6 out of 5.0E-6, or about 65%
Oconee:	4.9E-7 out of 1.4E-6, or about 35%
Three Mile Island:	5.6E-7 out of 1.3E-6, or about 45%

The reason for this contribution to core damage is threefold: (1) a high overall frequency of loss of main feedwater events plus (2) a relatively high human error probability for failure to properly respond, and (3) a relatively low probability of recovering feedwater once it has failed. Obviously, the contributions given above constitute the maximum possible reduction in core damage frequency if it were possible to eliminate these total loss of cooling sequences.

There are a number of BWOG recommendations (identified in Table 9.1) that would enhance performance in the three areas mentioned above, most of which pertain to the main feedwater system. The particular recommendations with the greatest potential for reducing core damage are TR-014-MFW, TR-015-MFW, TR-066-MFW, TR-074-MFW, and TR-052/053/054-SFI with respect to the high frequency of loss of main feedwater events. With respect to the low probability of feedwater recovery, recommendations TR-070-MFW and TR-071-MFW (for main feedwater recovery) and TR-055/056/057-ADM (for the emergency feedwater recovery) appear to be potentially the most effective. Recommendations TR-018-MFW and TR-067-MFW would be effective to both reduce the frequency of loss of main feedwater events and enhance main feedwater recoverability. In the area of operator response to total loss of feedwater, recommendations TR-060-OPS and TR-064-OPS would enhance operator response in recognizing the need to initiate bleed and feed and recognizing the need to recover EFW, respectively. In addition to these recommendations, the modification of the Davis-Besse bleed-and-feed capability, which was discussed in Section 5.3.8 of the SER, is deemed to be essential to enhance bleed-and-feed reliability at that plant.

The implementation of these recommendations could potentially result in a combined reduction of an order of magnitude in the frequency of core damage resulting from these total loss of cooling event sequences. After requantifying the model for the suggested modifications, the minimum core damage frequency of Category C events for each plant is calculated to be:

Crystal River 3:	8.6E-6
Davis Besse:	5.0E-7
ANO-1:	3.0E-6
Rancho Seco:	1.8E-6
Oconee:	9.9E-7
Three Mile Island:	8.0E-7

Thus, Category C events will become insignificant contributors for all B&W plants.

As noted above, these conclusions reflect the maximum risk effectiveness of the selected BWOG recommendations. Details of implementation will determine how much, if any, of this benefit will be realized at a specific plant. For example, the design features of Davis-Besse are such that the potential improvement cited above may not be fully achievable. At present, deficiencies in bleed and feed manifest themselves as beneficial to preventing transient-induced loss-of-coolant accidents (LOCAs). If bleed-and-feed capability is improved as part of the modifications, it is likely that some of this benefit will be lost to an increased core damage frequency resulting from a LOCA, although to what extent is not possible to quantify in the absence of more detailed information on the new Davis-Besse bleed-and-feed capabilities. The final result will probably fall more in the 2E-6 range rather than 5E-7. These results do, however, indicate that many of the recommendations have been well directed towards responding to the operational and design features of B&W plants that have generated concern. Thus, the staff concludes that proper implementation of the recommendations will increase safety of B&W plants by decreasing the overall plant risk.

Table 9.24 Recommendations that reduce the frequency of core damage from Category C events

Category/ Recommendation Number	Recommendation
ICS/NNI System Recommendations:	
TR-001-ICS	Replace reactor coolant (RC) flow signal input to integrated control system (ICS) with equivalent signals based on RC pump status.
TR-002-ICS	Eliminate plant transients and trips due to a single failure of a T_{hot} and T_{cold} signal. Implement a modification to automatically detect invalid RC temperature input to ICS.
TR-004-ICS	Implement a modification to automatically detect an invalid input to ICS of turbine header pressure.
TR-013-ICS	Install the necessary equipment to prevent loss of ± 24-volt power to the ICS or non-nuclear instrumentation (NNI) resulting from the loss of a single power source.
TR-033-ICS	Make appropriate changes to ensure that plant will go to a known, safe state without any operator action required on loss of ICS/NNI power.
TR-036-ICS	Evaluate turbine bypass valve position on loss of ICS.
TR-038-ICS	Develop and implement a recommended preventive maintenance program for ICS/NNI.
TR-102-ICS	Install redundant dc power supplies for NNI-Y Arkansas Power & Light Co. (AP&L) only.
TR-104-ICS	Incorporate automatic selection of valid input signals for ICS/NNI.
Main Feedwater System Recommendations:	
TR-014-MFW	Install a monitoring system in main feedwater pump (MFWP) trip circuitry to document the primary causes of MFWP trips.
TR-015-MFW	Determine if a low suction pressure trip is needed. Then decide what trip or response to low suction pressure should be implemented.
TR-017-MFW	Evaluate the MFWP control systems and their interaction with the ICS. Implement a program to identify improvements needed in both control systems.

Table 9.24 (Continued)

Category/ Recommendation Number	Recommendation
Main Feedwater System Recommendations (Continued)	
TR-018-MFW	Provide training to operators and maintenance personnel and ensure procedures are adequate for line-up, operation, and maintenance of main feedwater (MFW) system components.
TR-066-MFW	Check all MFW and condensate system protective circuits, interlocks, motors, etc., to ensure that a single electrical failure will not cause a loss of both feedwater trains.
TR-067-MFW	Evaluate the set point and functions of the automatic MFWP trip features. Wherever possible, eliminate these trip functions altogether.
TR-070-MFW	Provide capability to override a close signal to the MFW block valve to enable the control room operator to stop the block valve at any intermediate position during valve closure.
TR-071-MFW	Install valve position indication for the startup and MFW regulating valves (and low load control valves at applicable plants).
TR-074-MFW	Schedule instrumentation and control (I&C) calibration and inspection work so as to minimize the number of times the main FW pumps and turbines I&C equipment is disturbed during power operation.
Plant Operations Recommendations:	
TR-060-OPS	Stress in operator training that emergency operating procedures are to be followed explicitly even when such procedures are considered as drastic actions.
TR-064-OPS	Operator training to reset turbine-driven emergency feedwater (EFW) pumps after overspeed trips should be part of formal training programs and should include hands-on training.
Plant Administration Recommendations:	
TR-035-ADM	Familiarize operators with Rancho Seco event.
TR-055-ADM	Coordinate activities of plant operations, security and radcon (health physics) personnel to facilitate timely access to critical equipment.
TR-056-ADM	Move chain link fences as necessary, to provide better access to critical components.

Table 9.24 (Continued)

Category/ Recommendation Number	Recommendation
Plant Administration Recommendations (Continued)	
TR-057-ADM	Consider ways to improve access to critical components where problems have been identified with gaining access to critical components because of 10 CFR Appendix R fire barriers.
Main Steam/Feedwater Isolation System Recommendations:	
TR-052-SFI	AP&L, General Public Utility Corp. (GPUN) and Sacramento Municipal Utility District (SMUD) need to filter their steam generator level signals in the steam feedwater rupture control system.
TR-053-SFI	Correct overheating problems that can lead to electric power supply malfunctions and correct problems caused by degraded voltage power supplies (AP&L, GPUN, and SMUD).
TR-054-SFI	Redesign pneumatic hardware for main steam isolation valves to ensure this equipment is exercised during surveillance testing (AP&L only).

10 REACTOR TRIP INITIATING EVENTS REVIEW

In the SER, the NRC staff reported that its evaluation of the recommendations contained in the BWOG report, "Review of Reactor Trip Initiating Events at the BWOG Plants 1980-1986" (Report 47-1168891-00, September 1987), would be provided in a supplement to the SER. This supplement documents the staff's review and supersedes the original evaluation contained in the SER.

10.1 Description of Reactor Trip Initiating Events Review

As part of the SPIP, the BWOG Transient Assessment Committee reviewed reactor trip events that occurred at B&W plants during 1980-1985. The committee performed a quick sort of the BWOG Transient Assessment Program (TAP) data to identify major trip initiators and a detailed review of Category B and C events, which emphasized post-trip response. The results of these BWOG efforts are discussed in Section 4 of the SER.

To further the BWOG efforts to reduce the number of reactor trips, the BWOG Transient Assessment Committee reviewed the 235 reactor trip events, focusing on the initiating events that resulted in reactor trips at B&W plants from January 1, 1980, to July 31, 1986. The BWOG transmitted the "Review of Reactor Trip Initiating Events at the B&WOG Plants 1980-1986" to the staff in a letter dated October 1, 1987.

The BWOG reviewed each reactor trip to determine which event initiated the sequence of events that ultimately resulted in a reactor trip. These initiating events were grouped into the six categories listed below.

(1) turbine system
(2) feedwater systems
(3) integrated control systems (ICS)
(4) reactor coolant pumps (RCPs)
(5) control rod drive (CRD) system
(6) other

In addition, each reactor trip was examined to determine what trip function (e.g., high reactor coolant system (RCS) pressure) actually tripped the reactor and what inherent or design features should have prevented, but did not prevent, the reactor from tripping. This allowed each event to be reviewed from two perspectives: (1) prevention of the initiating event and (2) avoidance of the reactor trip by preventing or limiting the RCS response to the initiating event.

The B&W utilities have completed many corrective actions directed toward preventing trips. The BWOG review evaluated the effectiveness of these completed actions. This review also assessed the potential effectiveness of implementation of the recommendations developed from the SPIP. On the basis of this assessment, additional recommendations were developed to reduce the reactor trip frequency at B&W plants.

10.2 Conclusions From BWOG Review

On the basis of its review of the reactor trip events, the BWOG concluded that corrective actions taken by the B&W plants have been effective in reducing the reactor trip frequency by nearly 40 percent (1984-1986 average versus 1980-1983 average). Additionally, 41 recommendations in the RTS were identified as applicable to reducing the frequency of initiating events leading to reactor trips. Nevertheless, the BWOG concluded that additional efforts are necessary to further reduce the frequency of reactor trips and developed 36 additional recommendations. The specific recommendations developed and the NRC staff's assessment of these recommendations are provided in Section 10.3 of this supplement.

Other general conclusions from the study include:

(1) The turbine system was the leading trip initiator during 1980 through 1986, although corrective actions taken by the B&W plants have reduced the turbine trip frequency by two-thirds since 1983. Eight new recommendations have been proposed to further reduce turbine trip initiators.

(2) The feedwater system was the second leading trip initiator during 1980 through 1986. From 1984 through 1986, main feedwater (MFW) upsets, initiated by either the MFW system itself or in response to ICS failures, have accounted for over 50 percent of the reactor trips. Although numerous corrective actions have been implemented at the B&W plants, the frequency of MFW upsets has not declined. The BWOG expects that, with the implementation of the recommendations in the RTS, the frequency of MFW-initiated reactor trips will decline. However, an additional 11 recommendations were proposed for the MFW system.

(3) The ICS was the third leading trip initiator, primarily resulting from the loss of signal inputs. Although the RTS includes numerous recommendations related to the ICS, five new recommendations have been developed for this system.

(4) The control rod drive (CRD) system was the fourth leading trip initiator and corrective actions already taken by several of the B&W plants have been effective in reducing its contribution. Nonetheless, five new recommendations have been developed to further reduce CRD system initiated trips.

(5) The RCPs were the fifth leading trip initiator, primarily as a result of spurious actuations of the reactor coolant pump power monitors (RCPPMs). Corrective actions already taken at B&W plants have effectively eliminated these events and no additional recommendations were deemed necessary.

(6) The high RCS pressure trip was the leading RPS trip function accounting for 59 percent of the RPS trips. These trips were primarily the result of feedwater upsets, loss of ICS inputs, and MFW trips. While specific recommendations were developed to decrease these initiators, additional recommendations were made to limit the RCS thermal-hydraulic response to these initiators and thereby reduce the frequency of high-pressure RCS trips.

10.3 BWOG Recommendations and NRC Staff Findings

10.3.1 BWOG Recommendations Related to Reactor Trips Initiated by the Turbine System

As a result of its review, the BWOG developed the eight recommendations listed below relating to reducing reactor trips initiated by the turbine system.

(1) Plants with General Electric turbines should evaluate bypassing the high-vibration trip during stop-valve testing.

(2) Plants should install a time delay relay, or an orifice, between the electrohydraulic control (EHC) oil system and the anticipatory reactor trip system (ARTS) sensing line.

(3) The BWOG Instrumentation and Control (I&C) Committee should review the EHC overspeed and fast control and intercept valve circuits to determine why they are so frequently actuated and how they can be corrected to prevent recurrence.

(4) Protective covers should be placed over local level/trip switches, the inadvertent actuation of which can directly result in turbine or reactor trips.

(5) Procedures governing plant startup should be verified to ensure they have sufficient instructions for properly resetting anticipatory reactor trip system (ARTS) bistables/contact buffers.

(6) The BWOG I&C Committee should evaluate a design modification for automatically resetting the ARTS bistables on turbine reset.

(7) The turbine trip runback should be included in the operator training program.

(8) Integrated control system (ICS) tuning should be reviewed to ensure control settings are compatible with a turbine trip runback from less than 45 percent power.

Recommendations 1, 2, and 3 above were evaluated in Section 8.3 of the original SER and are already entered into the BWOG Recommendation Tracking System (RTS) as TR-169-MTS, TR-200-MTS, and TR-201-MTS, respectively. Recommendation 4 has been placed in the RTS as TR-213-ADM. Recommendations 5 and 6 are still being processed by the BWOG (see SER Section 12.1). The NRC staff notes that these recommendations will reduce unneeded reactor trips and thus are acceptable.

Recommendations 7 and 8 have been rejected by the BWOG Steering Committee. Recommendation 7 was rejected because it is standard industry practice to retrain operators after modifications are made, and recommendation 8 was rejected because ICS tuning is covered under recommendation TR-107-ICS. The staff notes that the specific training mentioned in recommendation 7 is involved with the implementation of the revised arming threshold for the ARTS on turbine trip. Thus, the standard practice to retrain operators after plant modifications will cover this recommendation. The staff also believes that recommendation 8 is adequately covered by the existing recommendation to improve tuning of the ICS. Therefore, the staff agrees with the rejection of these recommendations. Refer to Section 6.1 of this SSER for further evaluation regarding the electrical aspects of many of these recommendations.

10.3.2 BWOG Recommendations Related to Reactor Trips Initiated by the Main Feedwater System

As a result of its review, the BWOG identified the 11 recommendations listed below to reduce the frequency of trips initiated by the MFW system.

(1) Each plant should review, as a minimum, the MFW system design to identify and eliminate single failures that can cause the loss of multiple/redundant equipment necessary to support MFW pump operation.

(2) Each plant should review the feedwater startup and operations procedures to determine if switchover of steam supplies can be made at lower power levels and if the second MFW pump should be running when switchover is made.

(3) A time delay relay or an orifice between the MFW pump oil system and the ARTS sensing lines should be installed to prevent transient pressure oscillations from actuating the RPS circuitry.

(4) The BWOG should identify additional recommendations to improve the capability for successful plant runback on loss of one MFW pump.

(5) RTS item TR-067-MFW should be expanded to include the condensate and booster pumps, and the wording of TR-067-MFW should be modified to ensure that those trip signals that are required for pump/plant protection are hardened.

(6) The post-maintenance program recommended in TR-068-MFW should include proper checkout and tuning of newly installed systems or equipment before unit startup and planned tuning at power should be incorporated into plant procedures.

(7) The Transient Assessment Committee should develop a conceptual design for a multiparameter control room display that would provide the operator with a primary versus secondary heat balance.

(8) The Crystal River/Davis-Besse problems associated with deaerator tank level control and the Crystal River/Oconee problems associated with feedwater recirculation during startup need to be reviewed and necessary design changes implemented to prevent their recurrence.

(9) The BWOG I&C Committee should evaluate implementing a design change to automatically reset the ARTS bistables on MFW pump reset.

(10) Each plant should provide the status of the ARTS bistables on each MFW pump trip in the control room.

(11) The plants that trip both MFW pumps on high steam generator level should evaluate sequential versus simultaneous MFW pump trip.

Recommendation 11 above has been rejected by the BWOG Steering Committee because it determined that simultaneous MFW pump trip was the proper response to the steam generator high-level signal because of the severity of the consequences of steam generator overfill and potential overcooling concerns. The staff agrees with the disposition of this recommendation. Recommendations 1 and 2 were evaluated in Section 6.2.2 of the original SER and are both in the RTS. Recommendation 3 has already been entered into the RTS as TR-170-MFW, which was incorporated from the review of Transient Assessment Program (TAP) Report No. ANO-85-06. The remaining recommendations are still being processed by the Steering Committee and therefore are not in the RTS. Many of the remaining recommendations are closely related to existing recommendations in the tracking system so that further coordination is necessary for their disposition.

The staff concurs with these recommendations and encourages the BWOG to approve and enter them into the RTS to further aid in reducing the frequency of reactor trips initiated by the MFW system. Refer also to Section 6.1 of this supplement for a further evaluation concerning the electrical aspects of these recommendations.

10.3.3 BWOG Recommendations Related to Reactor Trips Initiated by the ICS

The BWOG identified five recommendations to reduce reactor trips initiated by the ICS.

(1) The lessons learned from the Rancho Seco experience regarding high-resistance relay contacts should be included as a part of the recommended ICS/NNI preventive maintenance program in TR-038-ICS.

(2) A periodic preventive maintenance program should be established to increase the reliability of inverters and vital buses.

(3) Existing RTS recommendation TR-038-ICS should be modified to include internal and external power supplies to the ICS/NNI.

(4) RTS recommendation TR-104-ICS should be expanded to include all ICS input signals.

(5) The BWOG I&C Committee should develop a post-trip or post-transient troubleshooting procedure to aid in diagnosing ICS module failures.

These recommendations are included in the NRC staff's evaluation of the BWOG ICS/NNI system review documented in Section 6.1 of this supplement.

10.3.4 BWOG Recommendations Related to Reactor Trips Initiated by the Control Rod Drive (CRD) System

Five recommendations related to reducing reactor trips initiated by the CRD system are listed below.

(1) Each B&W plant should evaluate eliminating or reducing the automatic ICS runback rate on assymetric rod condition from 30 percent to 3 percent full power per minute or some other reduced rate that is compatible with plant Technical Specifications.

(2) Plant operating procedures should provide instructions for de-energizing CRD controllers to stop uncommanded rod group insertions. The method used at Arkansas Nuclear One, Unit 1 (ANO-1), should be reviewed for applicability at each B&W plant.

(3) Each B&W plant should review its CRD cabinets for proper labeling and consistency with procedures.

(4) Maintenance procedures should be reviewed and upgraded as necessary to ensure that replacement modules and assemblies in the CRD system are properly checked out before use. This recommendation should be extended to include ICS and EHC components, circuit boards, etc.

(5) Plant procedures should include a requirement to trend the power/imbalance versus time relationship during xenon oscillations.

The first two recommendations have been entered into the RTS as TR-204-ICS and TR-214-CRD, respectively. The first recommendation is evaluated in Section 6.1.4 of this supplement as part of the staff's overall evaluation of the BWOG ICS/NNI review. The third and fourth recommendations were rejected by the BWOG Steering Committee. The third recommendation was rejected because this was deemed to be an industry issue being addressed on a broader plant-wide basis. The staff believes that the ongoing control room design reviews being performed by industry are sufficient to address this concern. Thus, the staff agrees with this rejection. The fourth recommendation was developed as the result of several reactor trips that occurred after failed components were replaced with components that also were defective. However, the BWOG Steering Committee rejected the recommendation to check out replacement components before installation because it is not a feasible or practical solution to the problem. The staff believes that the recommendation is directly responsive to the operating experience and that the Steering Committee should re-evaluate its rejection. The last recommendation is still being processed by BWOG.

The NRC staff believes that the recommendations developed by the BWOG are responsive to the operating experience at B&W plants and that implementation of the recommendations should reduce the reactor trips initiated by the CRD system. Thus, the staff finds the recommendations acceptable. However, the BWOG should reconsider its rejection of the recommendation to check out replacement components before their installation.

10.3.5 BWOG Recommendations Related to Reducing High- and Low-Pressure RCS Trips by Improving Transient Response Capability

The seven recommendations listed below were developed by the BWOG to reduce the frequency of reactor trips by improving the RCS thermal-hydraulic response to initiating events.

(1) Toledo Edison should evaluate lowering the low-pressure RCS trip set point from 1985 psig to 1900 psig.

(2) The BWOG Analysis Committee should re-evaluate the variable low-pressure RCS trip set point.

(3) The BWOG Analysis Committee should evaluate lowering the initial, steady-state RCS pressure set point from its present value of 2155 psig to provide additional margin to the high-pressure RCS trip set point.

(4) The BWOG should investigate methods for increasing the capacity of pressurizer spray flow to improve plant response to high-pressure transients.

(5) The BWOG should investigate methods for lowering the temperature of pressurizer spray flow to improve the plant response to high-pressure transients.

(6) The pressurizer spray control circuitry should be modified to automatically open the valve to the fully open position on initiation of high-pressure transients, or a fast-acting spray valve should be installed.

(7) An anticipatory spray signal should be developed to open the pressurizer spray valve for predefined events that could lead to a high-pressure transient.

Only the first recommendation has been entered into the RTS as TR-205-RPS. The remaining recommendations are still being processed by the BWOG.

The first two recommendations involve change in set points for the RPS. The first recommends an evaluation of the low-pressure RCS trip set point for Davis-Besse. The Davis-Besse low-pressure RCS trip set point is 1985 psig while the other B&W plants have trip set points of either 1800 or 1900 psig. Thus, the BWOG concluded that this change could probably be implemented at Davis-Besse without a significant decrease in safety margins. The objective of the second recommendation is to eliminate the variable low-pressure RCS trip in the RPS. The BWOG notes that only three reactor trips have occurred since 1980 from this trip function and that other trip functions in the RPS would have actuated and safely mitigated these events. The BWOG report contains insufficient information for the staff to conclude that these changes are appropriate. However, these modifications would require changes to the plant Technical Specifications and would have to be approved by the staff. Thus, pursuit of the studies recommended by the BWOG is acceptable to the staff since the utilities must perform and submit safety evaluations for staff review and approval before implementing RPS set point changes.

The third recommendation is to evaluate lowering of the steady-state operating pressure on the B&W plants in order to provide additional margin to the high-pressure RCS trip set point. Implementation of such a change in the normal plant operating conditions would affect the plant safety analyses and would probably require RPS set point changes and changes to the plant Technical Specifications. Should the BWOG decide to pursue this modification, the staff will require the BWOG to submit a plant safety evaluation for the change for staff review and approval before its implementation.

The remaining recommendations are focused on improving the effectiveness of the pressurizer spray system and thereby limiting the magnitude of overpressure transients. Such action will provide enhanced capability for B&W plants to respond to some plant upsets, such as loss of a single MFW pump, without resulting in a reactor trip. The staff believes that modifications to the pressurizer spray system will prevent unnecessary reactor trips for some events. However, the BWOG must ensure that any changes that may be implemented will not affect the plant safety analyses.

The recommendations made by the BWOG in this category will require careful study to ensure plant safety margins are not significantly reduced. However, the staff believes that, with proper analyses and engineering, some of these changes can probably be implemented in such a way as to reduce unnecessary reactor trips while not compromising plant safety.

10.4 Summary of Staff Findings

The NRC staff concludes that the implementation of BWOG recommendations will reduce the number of reactor trips. However, the staff believes that the BWOG should reconsider its rejection of the recommendations to (1) place protective covers over local level/trip switches that can directly result in a turbine or reactor trip and (2) ensure that replacement components are properly checked out before their use. In addition, the utilities should submit plant safety evaluations for staff review and approval before implementating changes to the RPS or the steady-state operating pressure of B&W plants.

11 BWOG PROGRAMMATIC AND MANAGEMENT ACTIONS

11.4 Safety and Performance Recommendation Integration Group

As stated in the SER, the NRC staff had not completed its review of the key recommendations and therefore would not comment on the acceptability of those key recommendations until the staff's review of ICS/NNI, reactor trip initiating events, and the emergency feedwater initiation and control system was completed. After the staff completed its review of these issues, it proceeded to review the recommendations contained in the BWOG RTS report dated September 1987 to determine the adequacy of the key recommendations identified by the BWOG. As mentioned in the SER, the Safety and Performance Recommendation Integration Group (SPRIG) developed specific criteria to determine the key recommendations. The staff has determined that the criteria used by the BWOG are acceptable and therefore the key recommendations identified by the BWOG are generally appropriate.

In addition, the staff reviewed all the recommendations in the RTS to determine whether any of those recommendations not designated as non-key recommendations have sufficient safety significance to be considered high-priority recommendations and thus should be given additional attention during the staff's review of the utility's implementation plans. Following is the staff's list of additional RTS recommendations it considers as high priority and its bases:

(1) TR-115-PES - Test diesel generators to ensure they will carry loads under expected sequential loading conditions. If the diesel generators have not been tested to ensure they can carry loads, then a potential safety concern exists since diesel generators provide the required emergency power in the event of a loss of offsite power.

(2) TR-122-IAS - Systematically inspect the instrument air system for leaks. This can be done easily and may significantly improve the performance of the instrument air system since air leaks can lead to contamination and overextension of air system capacity.

(3) TR-128-IAS - Review training and loss of air response procedures for instrument air systems. A potentially large benefit in ensuring safe plant shutdown can be gained from a relatively small effort by improving the capability to recover from a loss-of-instrument-air event.

(4) TR-143-IAS - Inspect accumulators and their check valves in the instrument air system. If those accumulators supplying backup air for safety-related equipment have not been inspected/tested on a regular basis, then a potential safety concern exists since accumulators are relied on to ensure valve operability following a loss of normal instrument air.

(5) TR-144-IAS - Develop or upgrade loss-of-instrument-air procedures (applicable to only three BWOG plants). The concern is the same as indicated under item 3 above.

(6) TR-157-OPS - Validate emergency operating procedures (EOPs) to determine if adequate staffing and prioritization exists. The BWOG review of Category B and C events shows increased operator attention is required to establish plant control for these events. Thus, validation of the EOPs will provide assurance that the operator guidance is available for mitigating these events.

(7) TR-174-MSS - Improve response of modulating turbine bypass valves. As discussed in Section 6.4 of the SER, main steam pressure control has not been ideal at B&W plants and has led to increased operator burden. Improved response of the modulating turbine bypass valves will reduce the complexity of and increase the safety margins of post-trip steam pressure control.

(8) TR-179-MFW - Identify areas for enhancing the reliability of the MFW and condensate systems and controls. The BWOG studies indicate that the frequency of MFW system-initiated reactor trips has not declined despite many MFW system modifications. Therefore, improving MFW/condensate system performance warrants priority consideration.

(9) TR-181-OPS - Verify adequacy of instrumentation and displays used to assess and control the stability parameters in the Abnormal Transient Operator Guidelines. Adequate instrumentation and display are required to ensure proper response by control room operators.

(10) TR-190-ICS - Develop backup manual and automatic controls for pressurizer level and pressure control powered from another power source. The staff evaluation of the capability of the B&W plants to reach a known safe state (KSS), following a partial or full loss of NNI/ICS power, found that in certain conditions the operator burden in controlling the plant is excessive and beyond that allowed by the BWOG definition for KSS. When NNI-X power is lost, the operator is required to manually control pressurizer level and pressurizer pressure when all indications in the control room of these two plant variables and all operable controls in the control room for these variables may have been lost.

(11) TR-201-MTS - Review EHC overspeed and fast control and intercept valve circuits. Since these circuits have led to seven reactor trips, improving their reliability could lead to a significant decrease in the number of turbine-related trips.

As noted in Section 12.2 of the SER, the NRC staff will review each utility's implementation plans. The staff will ensure that the key recommendations identified in the SPRIG report and the high-priority recommendations noted above are implemented in a timely manner. Furthermore, the staff expects each utility to evaluate all recommendations, not just the key recommendations. The staff will perform plant-specific audits and will monitor the utility's implementation programs to ensure proper disposition and timely implementation of all the recommendations in the RTS.

11.5 BWOG Evaluation Program To Assess Quality of Implementation of the SPIP Recommendations

In a letter to D. Crutchfield dated November 24, 1987, the BWOG stated that a program has been initiated to evaluate how each member utility dispositions the recommendations produced by SPIP. To ensure the quality of implementation, the BWOG has taken the following three-phase approach: (1) to evaluate a member utility's program for managing the SPIP recommendations, (2) to assess the technical adequacy of the implementation details for selected recommendations, and (3) to monitor the attainment of the SPIP goals. The BWOG Executive Committee has formed an evaluation team and has established mid-February as the completion date for reviewing Phase 1 of the program for each of the seven B&W utilities.

The staff believes there is a need for a program such as that developed by the BWOG. The majority of the recommendations developed by the BWOG do not provide specific design details. Rather, the recommendations generally define design criteria with the engineering of the detailed plant-specific modifications left to the utility. This approach is appropriate in light of the specific differences that exist among B&W plants. The BWOG's plans to audit the implementation of the recommendations to ensure that the interpretation of the recommendations and the quality of the implementation are commensurate with the intent of the BWOG SPIP. Therefore, the staff concludes that the BWOG program to audit the implementation of the recommendations should contribute toward achieving the goals of the SPIP.

One of the concerns identified by an NRC staff member (SER Section 2.2, item 4) relates to the fact that neither the BWOG nor the utilities have effectively analyzed the proposed SPIP recommendations to determine the effects on other parts of the plant. During a meeting with the BWOG on the reassessment program, the Advisory Committee on Reactor Safeguards (ACRS) asked a similar question related to whether the SPIP involved any systems interactions studies. The BWOG response, provided in BAW-1919, Section XI, p. XI-77, was that systems interactions were indirectly addressed through a detailed systems review of several major systems and their dependence upon and relationship to one another. The BWOG also stated that the sensitivity study, risk assessment, and failure modes and effects analysis in the ICS/NNI system evaluation, in combination with the detailed systems reviews, involved systems interaction activities.

The staff agrees with the BWOG that the reviews performed involve some systems interaction studies. However, because many of the SPIP recommendations are general in nature, detailed systems interaction studies cannot be performed at this time. As detailed engineering is performed by the individual utilities to implement the recommendations, systems interactions can be better defined and more effectively analyzed.

The staff notes that one of the guidelines for the audit program established by the BWOG is to evaluate, on a plant-specific basis, the potential for introducing new failure modes or systems interactions. It is the staff's opinion that because of plant-specific differences in B&W-designed plants, this is an acceptable approach and if properly evaluated should provide more confidence that any potentially adverse systems interactions will be addressed. In addition, the staff stated in Section 12.2 of the SER that it intends to audit the plant-specific implementation of the approved BWOG recommendations at selected B&W plants. As part of this audit, the staff will assess, where appropriate, if systems interactions have been considered.

The staff believes that the combination of actions described above provides the assurance that significant systems interactions related to the recommendations approved by the BWOG will be effectively addressed.

12 IMPLEMENTATION

12.2 NRC Audit of the BWOG Recommendation Approval Process

In this section of the SER, the staff stated that the NRC intends to send an inspection team to selected B&W plants early in 1988 to audit plant-specific implementation of approved BWOG recommendations. In a letter dated December 21, 1987, the BWOG stated that most of the recommendations require further evaluations that may require followup actions once the evaluations are completed. Since most of the recommendations are not at a stage where the acceptability of the utilities' interpretation of the implementation of the recommendations can be evaluated by the staff, these audits will be deferred. New dates will be established at a later time based on the status of the recommendations at the utilities selected to be audited. Before the implementation audit, the staff plans to audit the program in place at selected utilities to evaluate the SPIP recommendations for implementation at the plants.

This audit will be similar to that performed by the BWOG as described in its letter of November 24, 1987, and discussed in Section 11.5 of this supplement. This audit will provide the staff with additional assurance that the BWOG is properly monitoring the plans of its individual members.

12.3 Matrix of Approved Recommendations

In the SER, the staff stated that Table 12.1, which identified each of the 207 approved recommendations, and Table 12.2, which identified the 22 recommendations rejected by the BWOG, will be updated in supplements to the SER as the staff completes its review of the remaining sections of the BWOG SPIP and the BWOG submits updated information in the recommendation tracking system (RTS) report. Tables 12.1 and 12.2 have been updated as a result of the staff's completion of the review of integrated control system/non-nuclear instrumentation and reactor trip initiating events reports, the addition of eight recommendations in the January 1988 update of the RTS report, and the reconsideration of two previously rejected recommendations.

12.4 Implementation of Recommendations

In a letter dated December 21, 1987, the BWOG provided comments on the staff's SER of the BWOG plant reassessment program. The BWOG stated that it may not be possible for each utility to provide an implementation schedule by June 1, 1988, for most of the recommendations because further evaluation and potential followup actions may be required by the utilities. In addition, it stated that formal orders are not needed since the BWOG is fully committed to timely resolution of all appropriate recommendations. The BWOG also committed to provide periodic updates of the RTS, which show the status and progress and schedular information requested by the staff in the SER.

The staff believes that this RTS submittal will keep the staff cognizant of the overall implementation of the recommendations. However, because the BWOG lacks the authority to enforce individual utility commitments to the NRC, the staff still believes that its request in the SER for each utility to provide its schedule for implementation of the recommendations is appropriate. Based on the BWOG's comment that it may not be possible to provide an implementation schedule for most of the recommendations by June 1, 1988, the staff recommends that the individual utilities provide, no later than that date, a schedule for implementation of those recommendations that have been evaluated, a tentative schedule for implementation of those recommendations requiring further evaluation, and justification for the delays in implementation. The staff will review these submittals to determine whether the implementation schedules are appropriate. The staff further recommends that these schedules be regularly updated to include the remaining recommendations not yet approved by the BWOG Steering Committee and that early notification of schedular slippages be made with adequate justification. The NRC Project Manager for each B&W-designed utility would be responsible for tracking these implementation schedules. The BWOG RTS updates will be reviewed regularly by the staff to determine the overall program progress and the progress made by each individual utility.

Since the issuance of the SER, by letters dated January 5 and March 2, 1988, the BWOG has submitted two updated versions of the RTS report (November 1987 and January 1988). The January 1988 report added eight new recommendations to the RTS. These are the only additions since the September 1987 RTS report update was issued. In addition, two of the previously rejected recommendations (T-34 and TIR-2) have been re-evaluated by the BWOG and added to the list of approved recommendations (TR-211-ICS and TR-213-ADM are two of the eight added in the January 1988 RTS report). These reports contain an update on the status of those recommendations already in the RTS. Summarized below is a comparison of the changes in the status of the recommendations as reported in Appendix J to BAW-1919, Revision 5, issued July 1987, and the September and November 1987 and January 1988 updates to the RTS.

	All recommendations (%)				Key recommendations (%)			
	BAW-1919	RTS updates			BAW-1919	RTS updates		
Recommendation status category	July 1987	Sept 1987	Nov 1987	Jan 1988	July 1987	Sept 1987	Nov 1987	Jan 1988
Evaluating	38	34	44	44	33	36	37	41
Implementing	17	18	19	12	22	31	31	22
Closed operable	16	15	17	20	22	22	23	28
Closed not applicable	15	17	18	18	7	8	8	8
Closed rejected	1	1	1	1	2	1	1	1
No report	13	15	2	4	14	2	-	-

While this table indicates some improvement in implementation of the recommendations, a more detailed review of the reports shows that from July 1987 to

January 1988, with the exception of Rancho Seco, little progress was made in implementing and closing recommendations currently in the RTS. It appears that very little progress has been made recently toward approval of the remaining recommendations by the BWOG Steering Committee. Although this may not be representative of actual progress made, the staff is concerned about the apparent lack of substantive progress. Since it is the BWOG's intent to use the RTS to keep the staff informed of progress, the staff recommends that either more detail be provided in the RTS report or supplemental information be provided as an enclosure to the updated RTS reports so that the staff may more accurately evaluate program progress.

The staff will meet with the BWOG and the individual utilities to discuss the details of the schedular commitments for implementation of the recommendations and to identify the additional information needed to augment the updated RTS reports. Should the staff find at a later date that this approach does not provide the desired results (i.e., if there were significant schedular slippages in the implementation of the recommendations), the use of regulatory action on a plant-specific basis to ensure that the recommendations are implemented in a timely manner will be considered.

12.5 Impact Assessment of Proposed BWOG Recommendations

As discussed in Section 12.5 of the SER, the staff had not completed its review of the BWOG SPIP evaluation of the ICS/NNI and thus was unable to complete its assessment of the SPIP recommendations on the safety of B&W plants. Within most of the sections of the SER and this supplement, the staff has evaluated the individual BWOG tasks and the recommendations developed from them. In this section, specific safety issues have been defined and those RTS recommendations appropriate to that issue are collected. This collection of recommendations is then examined to provide another perspective of the SPIP efforts for B&W plant safety. The conclusions contained herein do not diminish any specific staff comments made in the individual sections of the SER or this supplement.

12.5.1 Reduction in Reactor Trips

One of the goals of the SPIP is to develop recommendations that will decrease the frequency of reactor trips. Areas of improvement and the associated recommendations in the RTS include those listed below.

(1) reduce the number of reactor trips to ICS/NNI that are caused by input signal failures (TR-001-ICS, TR-003-ICS, TR-005-ICS, and TR-104-ICS)

(2) improve power supplies to the ICS/NNI (TR-013-ICS, and TR-102-ICS)

(3) improve ICS performance by improved maintenance and tuning (TR-008-ICS, TR-009-ICS, TR-038-ICS, TR-107-ICS, TR-186-ICS, and TR-204-ICS)

(4) improve reliability of main feedwater system by minimizing trips to the MFW pump (TR-015-MFW, TR-016-MFW, TR-052-SFI, TR-053-SFI, TR-066-MFW, TR-067-MFW, TR-073-MFW, TR-076-MFW, TR-082-MFW, TR-083-MFW, TR-084-MFW, TR-170-MFW, TR-171-OPS, and TR-179-MFW)

(5) improve MFW control systems (TR-017-MFW, TR-021-MFW, TR-068-MFW, TR-069-MFW, TR-070-MFW, TR-072-MFW, TR-075-MFW, TR-085-MFW, and TR-088-MFW)

(6) improve maintenance and operator training for the MFW system (TR-018-MFW, TR-020-MFW, TR-068-MFW, TR-074-MFW, TR-079-MFW, TR-173-MFW, and TR-202-MFW)

(7) reduce the number of reactor trips caused by the turbine systems (TR-025-MTS, TR-030-MTS, TR-100-MTS, TR-101-MTS, TR-168-MTS, TR-169-MTS, TR-200-MTS, TR-201-MTS, and TR-206-MTS)

(8) modify reactor trip set points to provide increased capability to accommodate RCS response to upsets without resulting in a reactor trip (TR-027-ADM, TR-031-RPS, and TR-205-RPS)

(9) improve reliability of plant electrical systems (TR-112-PES, TR-113-PES, TR-117-PES, TR-119-PES, and TR-203-PES)

The staff expects that the actions proposed by the BWOG will, when implemented, have a significant impact on the frequency of reactor trips at B&W plants.

12.5.2 Challenges to Safety Systems

Challenges to safety systems have been considered from several perspectives by the BWOG. As noted above, the BWOG has proposed several actions that will reduce the challenges to the reactor trip system. Additionally, several of the actions discussed above will result in enhanced reliability of the MFW system and thereby will decrease the need for actuation of the auxiliary/emergency feedwater (AFW/EFW) system. Other actions recommended to reduce unnecessary AFW/EFW actuations include those listed below.

(1) review and modify AFW/EFW actuation set points to prevent unnecessary challenges to the system (TR-022-EFW and TR-161-EFW)

(2) improve operator control of the MFW system following a reactor trip (TR-069-MFW, TR-070-MFW, TR-071-MFW, and TR-207-OPS)

To minimize actuations of the emergency core cooling systems (ECCS), those actions listed below have been proposed to minimize the potential for overcooling of the RCS.

(1) improve performance of the main steam pressure control systems (TR-023-MSS, TR-024-MSS, TR-048-MSS, TR-109-MSS, TR-174-MSS, and TR-176-POV)

(2) reduce overcooling of the RCS as a result of excessive MFW or AFW/EFW flow (TR-098-MFW, TR-099-OPS, TR-155-EFW, TR-159-OPS, and TR-176-POV)

(3) minimize RCS response to loss of ICS/NNI power events or restoration of ICS/NNI power following such events (TR-032-ICS, TR-036-ICS, TR-037-ICS, TR-096-MSS, TR-097-EFW, and TR-178-ICS)

(4) reduce probability of PORV opening and ensure ability to isolate a stuck PORV (TR-051-OPS, TR-172-PRV, and TR-175-PRV)

The staff expects that the combinations of actions proposed by the BWOG will reduce the number of challenges to the plant safety systems and thus represent an improvement in the safety of B&W plants.

12.5.3 Operator Burden

As noted in Section 5 of the SER, the NRC staff concluded that for reactor trips, B&W plants do not pose greater burden on the plant operator if steam and feed flows to the steam generator are controlled within reasonable bounds. As discussed above, SPIP proposed several recommendations to ensure that steam and feed flows would be properly controlled following a reactor trip. Additionally, implementation of the actions proposed, most notably TR-024-MSS and TR-174-MSS, should reduce the need for B&W plant operators to take manual control to ensure that the main steam safety valves fully reseat following a reactor trip. Thus, these recommendations should reduce operator burden.

One of the areas of increased operator burden at B&W plants, noted in Section 5 of the SER, is the operator's difficulty in diagnosing and responding to failures of automatic control systems (ICS/NNI). Several of the recommendations discussed above were developed by the BWOG to address this issue. Additional recommendations related to the adequacy of instrumentation available and the availability of emergency operating procedures for a loss of ICS/NNI power include TR-012-ICS, TR-034-ADM, and TR-154-ICS.

Beyond these specific areas, the BWOG made additional recommendations to improve operator performance and reduce operator burden including those listed below.

(1) improve emergency notifications (TR-058-OPS, TR-059-OPS, and TR-156-OPS)

(2) identify high priority tasks during emergencies for specific training (TR-061-OPS, TR-064-OPS, TR-157-OPS, and TR-177-OPS)

(3) improve information available to operator in control room (TR-062-OPS, TR-063-OPS, and TR-158-OPS)

While the staff noted in Section 7 of the SER that the BWOG evaluation of operator burden would have been enhanced by the use of human factors expertise, the staff believes that the implementation of the SPIP recommendations will have a positive effect on reducing the operator burden at B&W plants.

12.5.4 Decay Heat Removal

B&W plants have several means of ensuring decay heat removal. First of these is the use of the MFW system. Actions taken by the BWOG to improve the reliability of the MFW system have been discussed above.

In the event that all main feedwater is lost, the AFW/EFW system is used to ensure the steam generator heat sink. Several recommendations developed for this system have been discussed previously. Additional recommendations developed to improve the reliability of the AFW/EFW system include those listed below.

(1) improve performance of motor-operated valves (TR-041-MOV, TR-042-MOV, TR-043-MOV, TR-044-MOV, TR-045-MOV, TR-046-MOV, and TR-047-MOV)

(2) improve operational reliability of EFW system (TR-160-EFW and TR-166-EFW)

(3) improve maintenance, testing, and surveillance of EFW system (TR-163-EFW, TR-164-EFW, and TR-165-EFW)

The staff also notes that it has an ongoing activity under GI-124 to evaluate the reliability of certain plants having only two AFW/EFW pumps, including Rancho Seco, Crystal River Unit 3, and ANO-1. The staff will develop evaluations/recommendations for the AFW/EFW systems in these plants, separate from the BWOG SPIP, to ensure that they demonstrate acceptable reliability for a loss of MFW event.

In the event that all sources of feedwater are unavailable, B&W plants have an alternate way of removing decay heat by feed-and-bleed cooling, as discussed in Section 5.3.8 of the SER. No specific recommendations have been developed by SPIP to enhance feed-and-bleed systems at B&W plants. However, the staff noted that the Davis-Besse plant, which unlike the other B&W plants has a low shutoff head for the high-pressure injection system, is pursuing separate plant-specific modifications to enhance its feed-and-bleed capability. To provide additional assurance that the feed-and-bleed system would be manually actuated by the plant operators when required, SPIP recommended, as TR-177-OPS, that operators be given specific training to reinforce the need to take action when specified by the emergency operating procedures (EOPS).

The NRC staff finds that implementation of the recommendations developed under SPIP will provide added assurance that adequate decay heat removal will be provided at B&W plants.

12.5.5 Pressurized Thermal Shock

Pressurized thermal shock (PTS) is a situation wherein the RCS overcools while it either remains pressurized or repressurizes. The BWOG proposed several actions aimed at reducing the potential for overcooling the RCS, as discussed in Section 12.5.2 above.

Examination of previous analytical results of PTS for B&W plants shows that one of the most frequent, and the most severe, PTS events is the plant response to a loss of ICS/NNI power. SPIP recommendation, TR-178-ICS, is directly responsive to this transient scenario and should have a significant effect on the PTS risk at B&W plants.

As noted in Section 9 of the SER, PTS risk at B&W plants is not a significant contributor to core damage even before the implementation of the recommendations. The staff believes that the actions proposed by the BWOG will further reduce any PTS risk posed at the B&W plants.

Table 12.1 SPIP Recommendations

Recommendation No.	Subject	Source: BAW-1919 and other reports	SER/SSER Section
TR-001-ICS	Replace RC flow signal input to ICS with RC pump status.	Vol. 1, pp. IV-14 to IV-16, §A.5.c.1	SSER 6.1.3
TR-002-ICS	Automatically detect invalid RC temperature inputs to ICS (superseded by Recommendation TR-104-ICS).	Superseded by Recommendation TR-104-ICS	SSER 6.1.3
TR-003-ICS	Remove startup FW flow correction to main FW flow function from the ICS.	Vol. 1, pp. IV-17 & 18, §A.5.c.3	SSER 6.1.3
TR-004-ICS	Automatically detect an invalid input to ICS of turbine header pressure (superseded by Recommendation TR-104-ICS).	Superseded by Recommendation TR-104-ICS	SSER 6.1.3
TR-005-ICS	Remove neutron flux signal auctioneering circuitry from RPS and relocate in the ICS.	Vol. 1, pp. IV-19 & 20, §A5.c.5, & Appendix J, p. 3-51	SSER 6.1.3
TR-006-ICS	Delete FW temperature correction to FW demand from ICS.	Vol. 1, pp. IV-20 & 21, §A.5.c.6, & Appendix J, p. 3-51	SSER 6.1.3
TR-007-ICS	Remove Btu limits from ICS.	Vol. 1, pp. IV-9 & 10, §A.4.6.1	SSER 6.1.4
TR-008-ICS	Improvements to reactor runback capability.	Vol. 1, pp. IV-11 & 12, §A.4.c.2	SSER 6.1.4
TR-009-ICS	Improvements to ICS tune control circuits.	Appendix J, §1.n., p. 24	SSER 6.1.4
TR-010-ICS	Modify ICS control circuit.	Appendix C, p. 2-5, §2.4.2 & Appendix R, §a., p. 7, §2.1.6	SSER 6.1.4
TR-011-ICS	Determine if the grid frequency error circuit has been detuned.	Vol. 1, p. IV-13, §A.4.c.5	SSER 6.1.4
TR-012-ICS	Determine if operator has necessary information from procedures, indicators, etc. to detect loss of NNI power.	Appendix J, §IX.d(1.3.3.f)	SSER 6.1.6, 6.1.9

Table 12.1 (Continued)

Recommendation No.	Subject	Source: BAW-1919 and other reports	SER/SSER Section
TR-013-ICS	Prevent loss of power to the ICS or NNI.	Appendix J, §I.a.5 & §IX.d	SSER 6.1.7
TR-014-MFW	Install monitoring system on MFW pumps to document causes of pump trips.	Appendix H, pp. 3-44 & 3-53	SER 6.2/SSER 6.1.12
TR-015-MFW	Determine if a low MFW pump suction pressure is needed.	Appendix B-4, p. 87, §6.2 & Appendix H, p. 3-45	SER 6.2
TR-016-MFW	Investigate oil system pressure in MFW pump.	Appendix B-4, pp. 90 & 91, §6.3	SER 6.2/SSER 6.1.12
TR-017-MFW	Evaluate MFW pump control systems.	Appendix B-4, pp 90 & 93, §6.4 & §6.4.2	SER 6.2/SSER 6.1.12
TR-018-MFW	Provide training on MFW system components.	Appendix B-4, p. 90, §6.2 & Appendix H, p. 3-35	SER 6.2, 7.3/SSER 6.1.9
TR-019-MFW	Ensure there are sufficient annunciator and trip signals for MFW supply system.	Document No. 47-1158723-00, "BWOG Operating Experience Summary Report for 1984," dated June 1985, p. 19	SER 6.2/SSER 6.1.12
TR-020-MFW	Establish procedures for switching of MFW pump oil supply.	Document No. OC-1-85-02, "BWOG TAP Report for Oconee 1 Reactor Trip on December 3, 1984," §1, p.1	SER 6.2
TR-021-MFW	Identify causes for MFW pump control problems.	Document No. DB-1-84-05 (B&W) Document No. 12-1151048-00), BWOG TAP Report, "D-B Rx Trip on 1/8/84," §1.D3, p. 2 & §II.E, p. 7	SER 6.2/SSER 6.1.12
TR-022-MFW	Review EFIC system low SG level set points.	Document No. D-B-1-85-03, "D-B Events of 3/16/85 and 3/21/85," §1.D.1, p. 2	SER 6.3
TR-023-MSS	Determine need to replace MSSV release nut cotter pins.	Document No. 12-1151244-00, "D-B TAP Report, D-B Reactor Trip of 3/2/84," §II.E.1, p. 13	SER 6.4

Table 12.1 (Continued)

Recommendation No.	Subject	Source: BAW-1919 and other reports	SER/SSER Section
TR-024-MSS	Determine causes to correct anomalous post-trip performance of MSSVs.	Document No. OC-1-85-01, "BWOG TAP Report, Oconee 1 Reactor Trip on December 2, 1984," §III.B.1, p. 13	SER 6.4
TR-025-MTS	Review EHC system for loss of input power.	"BWOG TAP Operating Experience Summary Report, Jan-Dec 1983," p. 2-4	SER 8.3/SSER 6.1.12
TR-026-OPS	Operability of SG shell thermocouples.	Document No. 12-1151244-00, "BWOG TAP Report, Davis-Besse Reactor Trip on March 2, 1984, " §II.E.2, p. 13	SER 4.2
TR-027-ADM	Calibration techniques for power range imbalance.	"BWOG Transient Assessment Program Operating Experience Summary Report, Jan-Dec 1983," p. 2-5	SER 4.2
TR-028-ADM	Training on power/imbalance control.	"BWOG Transient Assessment Operating Experience Summary Report, January-December 1983," pp. 3-1 & 3-2	SER 7.3
TR-029-ADM	Include human error information in TAP reports.	Document No. 77-1156349-00, "BWOG Report, A Study of Human Interface Events at B&W Operating Plants," February 1985, §5.1, p. 27	SER 7.3
TR-030-MTS	Raise ART on turbine trip arming point.	Appendix B-1, p. 6-1 & Vol. 1, pp. IV-3 to IV-5, §A.1	SER 8.1
TR-031-RPS	Increase setpoint for high pressure reactor trip.	Appendix B-3, p. 3-2 & Vol. 1, pp. IV-7 & 8, §A.3	SER 8.2
TR-032-ICS†	Evaluate restoration of ICS/NNI power.	Appendix R VII.B.1, p. VII-4	SSER 6.1.6
TR-033-ICS†	Ensure that plant will go to a safe state on loss of ICS/NNI power.	Appendix R VII.B.1, pp. VII-4 & VII-5	SSER 6.1.6

†Recommendations marked with a dagger are commitments to the NRC.

Table 12.1 (Continued)

Recommendation No.	Subject	Source: BAW-1919 and other reports	SER/SSER Section
TR-034-ICS†	Training for loss of ICS power.	Appendix R VII.B.1, p. VII-5	SSER 6.1.9
TR-035-ICS†	Familiarize operators with Rancho Seco event.	Vol. 1, p. IV-29, §5	SSER 6.1.9
TR-036-ICS†	Evaluate turbine bypass valve position on loss of ICS.	Appendix R VII.B.1, p. VII-5	SSER 6.1.6
TR-037-ICS†	Evaluate MFW pump speed control on loss of ICS power.	Appendix R VII.B.1, p. VII-5	SSER 6.1.6
TR-038-ICS	Develop and implement a preventive maintenance program for the ICS/NNI.	Appendix R VII.B.1, p. VII-5	SSER 6.1.8
TR-039-ICS	Wire the power supply monitor in the ICS/NNI directly to the output bus after the auctioneering diodes.	Appendix R VII.B.1, p. VII-5	SSER 6.1.7
TR-040-ADM	Use the TA Committee's Trip Investigation/Root Cause Determination Program	Appendix H, p. 4-2, Action Item 1	SER 11.1
TR-041-MOV	Confirm by field inspection data required to size operators and valves for motor-operated valves.	Appendix H, p. 4-2, Action Item 3.1	SER 8.4
TR-042-MOV	Obtain analytic methods used by valve and operator vendors.	Appendix H, p. 4-2, Action Item 3.2	SER 8.4
TR-043-MOV	Ensure that torque switch bypass limit switch is set to open after valve is unseated.	Appendix H, p. 4-2, Action Item 3.3	SER 8.4
TR-044-MOV	For wedge seating valves, position open direction torque switches to the highest allowable setpoints.	Appendix H, p. 4-2, Action Item 3.4	SER 8.4
TR-045-MOV	Ensure that maintenance procedures provide for properly setting torque switches and bypass limit switches.	Appendix H, p. 4-2, Action Item 3.5	SER 8.4

†Recommendations marked with a dagger are commitments to the NRC.

Table 12.1 (Continued)

Recommendation No.	Subject	Source: BAW-1919 and other reports	SER/SSER Section
TR-046-MOV	Challenge valves to open and close under differential pressures that simulate worst operational and accident conditions.	Appendix H, p. 4-2, Action Item 3.6	SER 8.4
TR-047-MOV	Institute formal training programs on motor-operated valves.	Appendix H, p. 4-2, Action Item 3.7	SER 8.4
TR-048-MSS	Revise turbine bypass valve preventive maintenance program.	Appendix H, p. 4-3, Action Item 6.1	SER 6.4
TR-049-MSS	Review and revise steam trap preventive maintenance program.	Appendix H, p. 4-3, Action Item 6.2	SER 6.4
TR-050-MSS	Include in plant operating procedures provisions for opening steam trap bypass valves during startup, and draining turbine bypass header valves before startup or cooldown.	Appendix H, p. 4-3, Action Item 6.3	SER 6.4
TR-051-OPS	Conduct postmaintenance and surveillance PORV testing.	Appendix H, p. 4-4, Action Item 7.1	SER 8.4
TR-052-SFI	Filter steam generator level signals in steam feedwater rupture control system.	Appendix H, p. 4-4, Action Item 10.1	SSER 6.1.12
TR-053-SFI	Correct overheating problems that can lead to electronic power supply malfunctions.	Appendix H., p. 4-4, Action Item 10.2	SSER 6.1.7
TR-054-SFI	Redesign MSIV pneumatic hardware to ensure this equipment is exercised during surveillance testing.	Appendix H, p. 4-4, Action Item 10.3	SER 8.4
TR-055-ADM	Coordinate the activities of plant operations, security, and radcon personnel to facilitate timely access to critical equipment.	Appendix H, p. 4-4, Action Item 11.1	SER 7.4
TR-056-ADM	Move chain link fences to provide better access to critical components.	Appendix H, p. 4-5, Action Item 11.2	SER 7.4

Table 12.1 (Continued)

Recommendation No.	Subject	Source: BAW-1919 and other reports	SER/SSER Section
TR-057-ADM	Consider ways to improve access to critical components.	Appendix H, p. 4-5, Action Item 11.3	SER 7.4
TR-058-OPS	Use highest emergency classification level when making initial notification to NRC.	Appendix H, p. 4-5, Action Item 12.1	SER 7.4
TR-059-OPS	Training for personnel who make emergency notifications.	Appendix H, p. 4-5, Action Item 12.2	SER 7.4/SSER 6.1.9
TR-060-OPS	Stress in personnel training that drastic actions shall be taken if required by procedures (superseded by Recommendation TR-177-OPS).	Appendix H, p. 4-5, Action Item 12.3	SER 7.4
TR-061-OPS	For specific training, identify high priority operator tasks during emergencies.	Appendix H, p. 4-5, Action Item 12.4	SER 7.4
TR-062-OPS	Maintain a high SPDS availability by corrective and preventive maintenance.	Appendix H, p. 4-5, Action Item 12.5	SER 7.4/SSER 6.1.6
TR-063-OPS	Ensure that P/T graphs are provided in the control room.	Appendix H, p. 4-5, Action Item 12.6	SER 7.4
TR-064-OPS	Training for resetting turbine driven EFW pumps.	Appendix H, p. 4-5, Action Item 12.7	SER 6.3, 7.4
TR-065-OPS	Improve communications between control room and certain plant areas at Rancho Seco.	Appendix H, p. 4-5, Action Item 12.8	SER 7.4
TR-066-MFW	Ensure that a single electrical failure will not cause a loss of both feedwater trains.	Appendix H, p. 4-6, Action Item 14.1	SER 6.2/SSER 6.1.12
TR-067-MFW	Wherever possible, eliminate automatic MFW pump trip functions.	Appendix H, p. 4-6, Action Item 14.2	SER 6.2/SSER 6.1.9
TR-068-MFW	Develop a postmaintenance testing program for the MFW pump turbines and governor controls.	Appendix H, p. 4-6, Action Item 14.3	SER 6.2/SSER 6.1.8
TR-069-MFW	Eliminate automatic control of the MFW block valve except during a reactor trip.	Appendix H, p. 4-7, Action Item 14.6	SER 6.2/SSER 6.1.9, 6.1.12

Table 12.1 (Continued)

Recommendation No.	Subject	Source: BAW-1919 and other reports	SER/SSER Section
TR-070-MFW	Provide capability to override a "close" signal to the MFW block valve.	Appendix H, p. 4-7, Action Item 14.7	SER 6.2/SSER 6.1.9, 6.1.12
TR-071-MFW	Install valve position indication for the startup and MFW regulating valves.	Appendix H, p. 4-7, Action Item 14.8	SER 6.2/SSER 6.1.12
TR-072-MFW	Eliminate the transfer from the startup to the MFW flowmeter when the MFW block valve opens.	Appendix H, p. 4-8, Action Item 14.9	SER 6.2/SSER 6.1.12
TR-073-MFW	Eliminate high MFW pump discharge pressure trips as a common occurrence.	Appendix H, p. 4-8, Action Item 14.10	SER 6.2
TR-074-MFW	Schedule I&C calibration and inspection work to minimize the number of times the MFW pump and turbine instrumentation and controls are disturbed.	Appendix H, p. 4-8, Action Item 14.11	SER 6.2
TR-075-MFW	Modify control scheme for the heater drain pump recirculation control valves (for ANO-1 only).	Appendix H, p. 4-8, Action Item 14.12	SER 6.2
TR-076-MFW	Eliminate automatic trip of the "preferred" MFW pump after a reactor trip (for ANO-1 only).	Appendix H, p. 4-8, Action Item 14.13	SER 6.2
TR-077-MFW	Review and upgrade preventive maintenance on auxiliary boilers.	Appendix H, p. 4-9, Action Item 14.18	SER 6.2
TR-078-MFW	Add an indication near the MFW pump controls for MFW pump discharge pressure.	Appendix H, p. 4-9, Action Item 14.19	SER 6.2, 7.4
TR-079-MFW	Put MFW regulating valves, main block valves and startup control valves on a refueling frequency for an operational check.	Appendix H, p. 4-9, Action Item 14.20	SER 6.2/SSER 6.1.8
TR-080-MFW	Instrumentation to determine performance of MFWPT shaft-driven oil pump.	Appendix H, p. 4-9, Action Item 14.21	SER 6.2

Table 12.1 (Continued)

Recommendation No.	Subject	Source: BAW-1919 and other reports	SER/SSER Section
TR-081-MFW	Movem control room MFW flow indication from back panel to apron (for Rancho Seco only).	Appendix H, p. 4-9, Action Item 14.22	SER 6.2
TR-082-MFW	Add automatic bypass of Powdex (or condensate demineralizer) units on high differential pressure.	Appendix H, p. 4-10, Action Item 14.24	SER 6.2
TR-083-MFW	Add MFW pump turbine lube oil purifiers.	Appendix H, p. 4-10, Action Item 14.25	SER 6.2
TR-084-MFW	Correct feed pump turbine shaft sealing problems.	Appendix H, p. 4-10, Action Item 14.26	SER 6.2
TR-085-MFW	Modify main FW pump recirculation valve for automatic control during startup and shutdown.	Appendix H, p. 4-10, Action Item 14.27	SER 6.2/SSER 6.1.12
TR-086-MFW	Improper draining of first-stage FW heaters	Appendix H, p. 4-10, Action Item 14.28	SER 6.2
TR-087-MFW	Add capability for flushing the feed pump turbine governor control oil system.	Appendix H, p. 4-10, Action Item 14.29	SER 6.2
TR-088-MFW	Eliminate automatic plant runback on low MFW pump discharge pressure or establish setpoint to achieve a successful runback.	Appendix H, p. 4-10, Action Item 14.30	SER 6.2/SSER 6.1.12
TR-089-MFW	Eliminate potential for physical damage to condensate and MFW pneumatic valve operator air supply lines.	Appendix H, p. 4-10, Action Item 14.31	SER 6.2
TR-090-MFW	Add valve position indication in control room for daer-rator feedwater tank inlet valves (for Davis-Besse only).	Appendix H, p. 4-11, Action Item 14.32	SER 6.2/SSER 6.1.12
TR-091-MFW	Eliminate need for an auxiliary operator to open a deaerator feedwater tank drain line after reactor trips (for Davis-Besse only).	Appendix H, p. 4-11, Action Item 14.33	SER 6.2/SSER 6.1.9
TR-092-MFW	Assess the cause for frequent feed booster pump low-suction pressure alarms (for Davis-Besse only).	Appendix H, p. 4-11, Action Item 14.34	SER 6.2

Table 12.1 (Continued)

Recommendation No.	Subject	Source: BAW-1919 and other reports	SER/SSER Section
TR-093-MFW	Allow full power operation using only two hot well pumps (for Oconee units only).	Appendix H, p. 4-11, Action Item 14.35	SER 6.2
TR-094-MFW	Reduce the effects of flashing of fourth stage FW heater drains (for Davis-Besse only).	Appendix H, p. 4-11, Action Item 14.36	SER 6.2
TR-095-MFW	Clean/flush the condensate pump motor coolers (for Davis-Besse only).	Appendix H, p. 4-11, Action Item 14.37	SER 6.2
TR-096-MSS	Evaluate design of turbine bypass and atmospheric dump systems.	Appendix I, p. III-2, No. PD-C1	SER 4.2, 6.4/SSER 6.1.6
TR-097-EFW	Evaluate design of EFW flow control valves.	Appendix R, §e p. 4; Appendix R VIII c.1, p. VIII-3; & Appendix I, p. III-3, No. PD-C2	SER 4.2, 6.3/SSER 6.1.6
TR-098-MFW	Overfill protection for MFW system.	Appendix I, p. III-5, No. PD-SB2	SER 4.2, 6.2/SSER 6.1.10
TR-099-OPS	Include guidence on excessive MFW, throttling AFW, and throttling HPI in plant procedures.	Appendix I, p. III-19, No. HI-C4	SER 6.3, 7.3/SSER 6.1.9
TR-100-MTS	Review MSR drain tank level control and drain line configuration.	Appendix B-2, p. 8-4, §8.2.2	SER 8.3/SSER 6.1.12
TR-101-MTS	Operator training on main generator excitation, voltage control and operation.	Appendix B-2, p. 8-5, §8.2.3	SER 8.3
TR-102-ICS	Install redundant dc power supplies for NNI-Y (for AP&L only).	Appendix R VII.B.1, p. VII-5	SSER 6.1.7
TR-103-ICS	Fuse external power learning ICS/NNI cabinets (for FPC only).	Appendix R VII.B.1, p. VII-5	SSER 6.1.7
TR-104-ICS	Incorporate automatic selection of valid inputs for ICS/NNI.	Appendix R VII.B.1, p. VII-5	SSER 6.1.3

Table 12.1 (Continued)

Recommendation No.	Subject	Source: BAW-1919 and other reports	SER/SSER Section
TR-105-ICS	Perform field verification of ICS/NNI drawings.	Appendix R VII.B.1, p. VII-5	SSER 6.1.7, 6.1.8
TR-106-ICS	Remove unused hardware from ICS/NNI cabinets.	Appendix R VII.B.1, p. VII-6	SSER 6.1.4, 6.1.8
TR-107-ICS	Improved maintenance and tuning of ICS.	Appendix R VII.B.1, p. VII-6; Appendix I, P. III-10, No. PM-5B2; and Appendix I, p. III-12, No. PM-081.	SSER 6.1.8
TR-108-MSS	Investigate using maximum allowable set pressure for the lowest set MSSVs (or for TMI-1 only).	Appendix N, p. 12-14, Item 11, §12.5	SER 6.4
TR-109-MSS	Ensure that relief valves not automatically isolated from main steam system post trip are in a preventive maintenance and test program.	Appendix N, p. 12-13, Item 10, §12.5	SER 6.4
TR-110-MSS	Davis-Besse should provide continuous EFW flow as a function of level.	Appendix N, p. 12-13, Item 6, §12.5 & Appendix N, p. 12-14, Item 11, §12.5	SER 6.4
TR-111-RPS	Review safety system surveillance procedures for checking which channel is available for testing before initiation of test.	Appendix K, p. III-1, §1	SER 4.4/SSER 6.1.8
TR-112-PES	Review switchyard maintenance procedures to ensure there is no mechanism for loss of offsite power.	Appendix K, p. III-1, §2	SER 4.4
TR-113-PES	Review breaker control power distribution to determine effects of a loss of the battery bus.	Appendix K, p. III-2, §3	SER 4.4/SSER 6.1.7
TR-114-PES	Evaluate hardware to ensure diesel generators cannot be synchronized to grid out of phase.	Appendix K, p. III-2, §5	SER 4.4
TR-115-PES	Test diesel generators to assure they will carry loads under expected sequential loading conditions.	Appendix K, p. III-2, §6	SER 4.4

Table 12.1 (Continued)

Recommendation No.	Subject	Source: BAW-1919 and other reports	SER/SSER Section
TR-116-PES	Review dc charging system and ensure the charging voltage does not exceed plant equipment voltage rating.	Appendix K, p. III-3, §7	SER 4.4/SSER 6.1.7
TR-117-PES	Notify inverter overcurrent protection to ensure the breaker/fuses open on overcurrent before inverters fail.	Appendix K, p. III-3, §8	SER 4.4/SSER 6.1.7
TR-118-PES	Evaluate loadings on ac and dc vital buses to ensure adequate margins exist without trip of equipment.	Appendix K, p. III-3, §9	SER 4.4/SSER 6.1.7
TR-119-PES	Implement preventive maintenance for electrical buses.	Appendix K, p. III-3, §10	SER 4.4/SSER 6.1.7
TR-120-IAS	Check O-rings in critical air-operated valves.	Appendix M, p. 5-2, §5.2.1	SER 6.5
TR-121-IAS	Make appropriate personnel aware of importance of instrument air system, prohibition of use for tools, and need to report air system damage.	Appendix M, p. 5-2, §5.2.2	SER 6.5/SSER 6.1.9
TR-122-IAS	Systematically inspect instrument air systems for leaks.	Appendix M, p. 5-3, §5.2.4	SER 6.5
TR-123-IAS	For instrument air systems, protect against failures possible with desiccant-type driers.	Appendix M, p. 5-3, §5.2.5	SER 6.5
TR-124-IAS	Identify metal lines with high vibration in instrument air systems, and when cracks are found, replace metal lines with flexible tubine.	Appendix M, P. 5-4, §5.2.6	SER 6.5
TR-125-IAS	Test critical air-operated valves for operability in the preventive maintenance program.	Appendix M, p. 5-4, §5.2.8	SER 6.5
TR-126-IAS	Compare instrument air system configuration with functional target criteria.	Appendix M, p. 5-5, §5.2.10	SER 6.5
TR-127-IAS	For instrument air system, review preventive maintenance program, identifying parameters for trending to determine maintenance requirements.	Appendix M, p. 5-5, §5.2.11	SER 6.5

Table 12.1 (Continued)

Recommendation No.	Subject	Source: BAW-1919 and other reports	SER/SSER Section
TR-128-IAS	Review training and loss of air response procedures for instrument air system.	Appendix M, p. 5-5, §5.2.12	SER 6.5/SSER 6.1.9
TR-129-IAS	Install automatic bypass line around driers and filters (for ANO-1 only).	Appendix M, p. 5-5, §5.3.1.1	SER 6.5
TR-130-IAS	Expand procedure for the loss of instrument air (for ANO-1 only).	Appendix M, p. 5-6, §5.3.1.2	SER 6.5/SSER 6.1.9
TR-131-IAS	Investigate feasibility of routing instrument air compressor intakes to the exterior (for Oconee units only).	Appendix M, p. 5-6, §5.3.2.1	SER 6.5
TR-132-IAS	Add an after-drier to the instrument air line (for Oconee units only).	Appendix M, pp. 5-6 & 5-7, §5.3.2.2	SER 6.5
TR-133-IAS	Add a filtration system downstream of the last drier in the instrument air system (Oconee units only).	Appendix M, p. 5-7, §5.3.2.3	SER 6.5
TR-134-IAS	Install control room-operated isolation valves with manual bypass at the key line feeding each unit's auxiliary building instrument air system header (for Oconee units only).	Appendix M, p. 5-8, §5.3.2.4.1	SER 6.5
TR-135-IAS	Install automatic isolation valves that could limit instrument air system leaks (for Oconee units and CR-3).	Appendix M, p. 5-8, §5.3.2.4.2	SER 6.5
TR-136-IAS	Install a dewpoint monitor downstream of instrument air system driers (for Duke, FPC and TED only).	Appendix M, p. 5-8, §5.3.2.5; Appendix M, P. 5-11, §5.3.3.3; & Appendix M, p. 5-19, §5.3.7.1	SER 6.5
TR-137-IAS	Check accumulators in instrument air system for water buildup. Install drain valves where necessary (for all operating plants except ANO-1).	Appendix M, p. 5-8, §5.3.2.6 Appendix M, p. 5-12, §5.3.3.6; & Appendix M, p. 5-15, §5.3.4.3	SER 6.5

Table 12.1 (Continued)

Recommendation No.	Subject	Source: BAW-1919 and other reports	SER/SSER Section
TR-138-IAS	Install a check valve after each compressor aftercooler in instrument air system (for Duke and FPC only).	Appendix M, p. 5-9, §5.3.2.7 & Appendix M, p. 5-13, §5.3.3.7	SER 6.5
TR-139-IAS	Install on/off status and remote start of instrument air compressors in the control room (for Duke and FPC only).	Appendix M, p. 5-9, §5.3.2.8 & Appendix M, p. 5-10, §5.3.3.1	SER 6.5
TR-140-IAS	Assign high maintenance priority to an out-of-service air compressor and maintain sufficient spare parts to repair a compressor within a week (for Oconee units only).	Appendix M, p. 5-9 & 5-10, §5.3.2.9	SER 6.5
TR-141-IAS	Install an automatic bypass valve to bypass driers filters upon loss of instrument air header pressure (for FPC, WNP, and TVA).	Appendix M, pp. 5-10, 5-11, §5.3.3.2; Appendix M, p. 5-18, §5.3.6.1; & Appendix M, p. 5-20, §5.3.8.1	SER 6.5
TR-142-IAS	Design the components of the instrument air system to withstand maximum flow generated by all the compressors (for FPC only).	Appendix M, p. 5-11, §5.3.3.4	SER 6.5
TR-143-IAS	Inspect accumulators and their check valves in the instrument air system (for FPC, GPUN, SMUD, and TED only).	Appendix M, p. 5-12, §5.3.3.5; Appendix M, p. 5-15, §5.3.4.2; & Appendix M, pp. 5-16 & 5-17, §5.3.5.2	SER 6.5
TR-144-IAS	Develop or upgrade a loss-of-instrument air procedure (for FPC, SMUD and TVA only).	Appendix M, p. 5-13, §5.3.3.8; Appendix M, p. 5-17, §5.3.5.4; & Appendix M, p. 5-22, §5.3.8.4	SER 6.5
TR-145-IAS	Install automatic isolation valves in instrument air-lines (for CR-3 only).	Appendix M, p. 5-14, §5.3.3.9;	SER 6.5
TR-146-IAS	Loss of air procedure for instrument air system should note importance of quickly bypassing driers and filters when excessive flow rates are experienced (for TMI-1 only).	Appendix M, p. 5-14, §5.3.4.1	SER 6.5
TR-147-IAS	Normally open or closed positions are recommended for certain valves (for TMI-1 only).	Appendix M, p. 5-16, §5.3.4.4	SER 6.5

Table 12.1 (Continued)

Recommendation No.	Subject	Source: BAW-1919 and other reports	SER/SSER Section
TR-148-IAS	Install automatic isolation valves at specified points in instrument air system (for SMUD only).	Appendix M, p. 5-16, §5.3.5.1	SER 6.5
TR-149-IAS	Design instrument air system components to withstand maximum flow generated by all compressors (for TED only).	Appendix M, p. 5-19, §5.3.7.2	SER 6.5
TR-150-IAS	Eliminate the ESFAS signal to close specified valves and isolate service and control air (for TVA only).	Appendix M, p. 5-21, §5.3.8.2	SER 6.5
TR-151-IAS	Eliminate apparent inconsistencies in instrument air valve designations on various drawings (for TVA only).	Appendix M, p. 5-21, §5.3.8.3	SER 6.5
TR-152-IAS	Establish same run time for the various compressors in the instrument air system (for (TVA only).	Appendix M, p. 5-22, §5.3.8.5	SER 6.5
TR-153-IAS	Evaluate a plant-specific air system failure.	Appendix M, p. 6-10, §6.7.1	SER 6.5
TR-154-ICS	Provide operator with unambiguous status of indicators and recorders in main control room on loss of ICS/NNI power or signal.	Appendix R, VII.B.1, p. VII-6	SSER 6.1.3, 6.1.6, 6.1.9
TR-155-EFW	Limit maximum flow rate delivered by the EFW system.	Appendix P, Vol. II, p. B-12, §3.A; Appendix Q, p. 2-2, §2.1.3; & Appendix Q, p. 2-3, §2.1.4	SER 6.3, 4.2, 5.4/ SSER 6.1.10
TR-156-OPS	Provide a sesignated "phone talker" to relay Emergency Plan messages.	Appendix S, p. 3-2, §3.1.2	SER 7.5/SSER 6.1.9
TR-157-OPS	Validate EOPs to determine if adequate staffing and prioritization exists	Appendix S, p. 3-2, §3.4.2	SER 7.5
TR-158-OPS	Re-evaluate annunciator designs to ensure key alarms do not go unnoticed.	Appendix S, p. 3-5, §3.6.2	SER 7.5/SSER 6.1.6

Table 12.1 (Continued)

Recommendation No.	Subject	Source: BAW-1919 and other reports	SER/SSER Section
TR-159-OPS	Evaluate secondary system controls to achieve remote manual control in the main control room of all post-trip steam flow paths, MFW and EFW.	Appendix N, p. 12-13, Item 7, §12.5 & Appendix S, p. 3-8, §3.9.2	SER 4.2/SSER 6.1.6
TR-160-EFW	Evaluate ability to extend the time to achieve design EFW flow.	Appendix Q, p. 2-1, §2.1.2	SER 6.3
TR-161-EFW	Evaluate the margin between the EFW and MFW low-level control points to prevent unneeded EFW actuations.	Appendix Q, p. 2-3, §2.1.5	SER 6.3
TR-162-EFW	Modify EFW flow control to provide smoother flow control rather than on-off control.	Appendix N, p. 12-13, Item 6, §12.5; Appendix N, p. 12-14, Item 11, Appendix Q, p. 2-4, § 2.1.6; & Appendix S, p. 3-9, §3.10.2	SER 6.3
TR-163-EFW	Review EFW surveillance and test procedures to ensure that components used in the EOPs are included in the test program.	Appendix Q, p. 2-5, §2.2.2	SER 6.3/SSER 6.1.8
TR-164-EFW	Review EFW Preventive Maintenance Program, including minimizing potential for common cause failures arising from maintenance and testing procedures.	Appendix Q, p. 2-5. §2.2.3; Appendix Q, p. 2-6, §2.2.6; & Appendix Q, p. 2-6, §2.4	SER 6.3/SSER 6.1.8
TR-165-EFW	Review EFW maintenance and testing procedures to eliminate conflicting and confusing instructions.	Appendix Q, p. 2-6, §2.2.4	SER 6.3/SSER 6.1.8
TR-166-EFW	Implement a program to improve and maintain the availability and performance of the EFW systems.	Appendix Q, p. 2-6, §2.3	SER 6.3/SSER 6.1.8
TR-167-PES	Include in operating procedures guidance on restoration of power to electrical buses, especially if the ICS or ICS controlled equipment is affected.	Document No. OC-1-85-06, "TAP Report," p. 7; Document No. TMI-86-02, "TAP Report," p. 19; & Document No. RS-86-01, "TAP Report," p. 46	SSER 6.1.6

Table 12.1 (Continued)

Recommendation No.	Subject	Source: BAW-1919 and other reports	SER/SSER Section
TR-168-MTS	Provide guidance in procedures when troubleshooting the EHC.	Document No. OC-2-85-02, "TAP Report," p. 6 & Document No. OC-2-86-01, "TAP Report," p. 6	SER 8.3/SSER 6.1.8
TR-169-MTS	Evaluate possibility for defeating the high vibration trip during main turbine valve testing (for GE turbines only).	Document No. D-B-1-85-04, "TAP Report, p. 5 & Document No. 47-1168891-00, "Review of Reactor Trip Initiating Events at the BWOG Plants, 1980-1985," pp. V-3, V-7, & V-8	SER 8.3/SSER 6.1.12
TR-170-MFW	Evaluate placing orifice snubbers in the MFW pump control oil system.	Document No. ANO-85-06, "TAP Report," p. 4	SER 6.2/SSER 6.1.12
TR-171-OPS	Evaluate alarm setpoints to determine if adequate time is provided for operator response.	Document No. OC-3-86-02, "TAP Report," p. 6	SSER 6.1.9
TR-172-PRV	Evaluate PORV circuitry to determine if momentary loss of power or restoration of power can cause PORV to open.	Document No. TMI-86-02, "TAP Report," p. 27 & Document No. OC-3-86-04, "TAP Report," p. 12	SSER 6.1.6
TR-173-MFW	Ensure in procedures that MFW pump status to ARTS/RPS is reset after each MFW pump is operational.	Document No. ANO-85-05, "TAP Report," p. 5	SER 6.2/SSER 6.1.12
TR-174-MSS	Improve response of modulating turbine bypass valves.	Appendix N, p. 12-14, Item 11, §12.5	SER 6.4
TR-175-PRV	Ensure the PORV block valve functions as designed under transient conditions.	"B&W Owners Group Review of June 9, 1985 Davis-Besse Loss of Feedwater Transient," BWOG-1154 Task Force, August 1986, p. 3-20 & Document No. 47-1165733-00, Review of Category B and C Events at the B&WOG Plants 1980-1985," §7.0, Item 2.0	SER 8.4

Table 12.1 (Continued)

Recommendation No.	Subject	Source: BAW-1919 and other reports	SER/SSER Section
TR-176-POV	Ensure power-operated valves employed in controlling post-trip energy balance between primary and secondary system are subjected to confirmation of ability to perform their function.	Appendix H, "BWOG Review of the June 9, 1985 Davis-Besse Loss of Feedwater Transient," B&WOG Task Force 1154 & Appendix I, "Review of Category B and C Events at the B&WOG Plants 1980-1985," Document No. 47-1165733-00	SER 8.4
TR-177-OPS	Review EOPs to ensure that wherever drastic actions are specified, plant conditions require the action.	Appendix S, p. 3-5, §3.5.1	SER 7.5
TR-178-ICS	Ensure plant goes to a known safe state on loss of power to the ICS/NNI.	B&W Owners Group I&C Committee Meeting Minutes, March 20, 1986; B&WOG SPIP-IAB Meeting Minutes, March 31-April 1, 1987; & Appendix R, ICS/NNI Evaluation Final Report, p. LC-1 (Known Safe State on the Loss of ICS/NNI Power)	SSER 6.1.6, 6.1.9
TR-179-MFW	Identify areas for enhancing the reliability of the MFW and condensate systems and controls.	Appendix P, Vol. II, p. B-8, §2.A & p. B-9, §2.B.7 & §2.C	SER 6.2
TR-180-MTS	Provide a monitoring capability for the EHC system for purpose of root cause determination.	Appendix B-2, p. 8-3, §8.2.1	SER 8.3/SSER 6.1.12
TR-181-OPS	Verify adequacy of instrumentation and displays used to assess and control the ATOG stability parameters.	Appendix I, p. III-14, No. HI-C2	SER 7.5/SSER 6.1.6, 6.1.9
TR-182-ICS	Evaluate installing automatic bus transfer switches of MFW pump controllers (or Davis-Besse only).	Appendix R VII.B.2, p. VII-6	SSER 6.1.7
TR-183-ICS	Preventive maintenance and testing for ABT switches.	Appendix R VII.B.2, p. VII-6	SSER 6.1.7, 6.1.8
TR-184-ICS	Provide separate fuses for hand stations that use ac power.	Appendix R VII.E, p. VII-11	SSER 6.1.7
TR-185-ICS	Power feedwater flow recorders directly from NNI.	Appendix R VII.E, p. VII-11	SSER 6.1.6

Table 12.1 (Continued)

Recommendation No.	Subject	Source: BAW-1919 and other reports	SER/SSER Section
TR-186-ICS	Minimize access to ICS/NNI cabinets during operation, and train maintenance personnel on location of power distribution components.	Appendix R VII.B.2, p. VII-8	SSER 6.1.7, 6.1.8
TR-187-ICS	Install current and voltage meters for NNI power supplies (for Davis-Besse only).	Appendix R VII.B.2, p. VII-8	SSER 6.1.7
TR-188-ICS	Maintain dc power supply current balance and perform a periodic full load test for each power supply.	Appendix R VII.B.2, p. VII-8	SSER 6.1.7, 6.1.8
TR-189-ICS	Set selector switches to select maximum NNI dependence.	Appendix R VII.B.2, p. VII-7	SSER 6.1.3
TR-190-ICS	Develop backup controls for puressurizer level and pressure control.	Appendix R VII.B.2, p. VII-10	SSER 6.1.4, 6.1.6
TR-191-ICS	Separate condensate flow from NNI power (for CR-3 only).	Appendix R VII.B.2, p. VII-7	SSER 6.1.6
TR-192-ICS	Remove/modify NNI power supply and signal select logic (for Rancho Seco only).	Appendix R VII.E, p. VII-12	SSER 6.1.7
TR-193-ICS	Review/test pressurizer heater low-low level interlock logic.	Appendix R VII.E, p. VII-12	SSER 6.1.4
TR-194-ICS	Buffer hand powered indicators and recorder inputs from automatic power signals.	Appendix R Vol. I, pp. 5-3 & 27-2	SSER 6.1.4
TR-195-ICS	Supply hand and automatic powered circuits from separate panels.	Appendix R VII.B.2, p. VII-6	SSER 6.1.6
TR-196-ICS	Set pressurizer level signal select relays to automatic powered transmitters.	Appendix R VII.B.2, p. VII-7	SSER 6.1.3
TR-197-ICS	Provide automatic power transfer for the modulating pressurizer heater converters.	Appendix R VII.B.2, p. VII-7	SSER 6.1.6

Table 12.1 (Continued)

Recommendation No.	Subject	Source: BAW-1919 and other reports	SER/SSER Section
TR-198-ICS	Automatic powered reactor inlet and out temperature sensors should be selected, or logic changed to automatically select auto powered sensor on loss of hand power.	Appendix R VII.B.2, p. VII-7	SSER 6.1.3
TR-199-ICS	Failure of inputs to RC pump interlocks must not prevent pump restart.	Appendix R VII.E, p. VII-12	SSER 6.1.4
TR-200-MTS	Install a time-delay relay or an orifice between the EHC oil system and the ARTS sensing line to prevent oil pressure perturbations.	Document No. 47-1168891-00, Sept. '87, "Review of Reactor Trip Initiating Events at the B&WOG Plants 1980-1986," p. IV-4, §IV.B.1.C	SER 8.3/SSER 6.1.12
TR-201-MTS	Review EHC overspeed and fast control and intercept valve circuits.	Document No. 47-1168891-00, Sept. '87, "Review of Reactor Trip Initiating Events at the B&WOG Plants 1980-1986," p. IV-4, §IV.B.1.D	SER 8.3/SSER 6.1.12
TR-202-MFW	Review FW system procedures to determine if switchover of steam supplies can be made at lower power levels.	Document No. 47-1168891-00, Sept. '87, "Review of Reactor Trip Initiating Events at the B&WOG Plants 1980-1986," p. IV-5, §IV.B.2.F	SER 6.2
TR-203-PES	Establish preventive maintenance to increase reliability of inverters.	Document No. 47-1168891-00, Sept. '87, "Review of Reactor Trip Initiating Events at the B&WOG Plants 1980-1986," p. IV-6, §IV.B.3.b	SSER 6.1.7, 6.1.8, 10.3
TR-204-ICS	Eliminate or reduce automatic ICS runback rate on asymmetric rod conditions.	Document No. 47-1168891-00, Sept. '87, "Review of Reactor Trip Initiating Events at the B&WOG Plants 1980-1986," p. IV-6, §IV.B.5.c	SSER 6.1.4
TR-205-RPS	Evaluate lowering the low RCS pressure trip setpoint (for Davis-Besse only).	Document No. 47-1168891-00, Sept. '87, "Review of Reactor Trip Initiating Events at the B&WOG Plants 1980-1986," p. IV-7, §IV.B.6.f	SSER 10.3

Table 12.1 (Continued)

Recommendation No.	Subject	Source: BAW-1919 and other reports	SER/SSER Section
TR-206-MTS	Improve purity of lubricating oil in the EHC system, and standpipe in lubricating oil tank, and add sample line at bottom tank.	Appendix B-2, p. 8-6, §8.3	SER 8.3
TR-207-OPS	Review operator training with regard to the manual control of MFW post-trip.	Appendix I, p. III-22, No. HI-SB1	SER 6.2/SSER 6.1.9
TR-208-ICS	Establish program to monitor control system.	Appendix R, p. VII-12, Section VII.E, Suppl. Doc. d. §3.5, p. IV-8, Section IV.A	SSER 6.1.8
TR-209-ICS	Add signal limiters to prevent control integrals from ongoing into saturation.	Appendix R, p. 18, Suppl. Doc. e	SSER 6.1.4
TR-210-ICS	Verify that standards for circuit board repair are included in maintenance training program.	Appendix R, p. I-3, Suppl. Doc. r	SSER 6.1.8
TR-211-ICS	Develop modification to remove automatic ICS Trip on NNI single power failure.	Appendix R, Suppl. Doc. q Vol. 1	SSER 6.1.6
TR-212-ICS	Label ICS/NNI switches S1 and S2 to detect energized vs. tripped positions.	Appendix R, p. VII-12, Section VII.E	SSER 6.1.9
TR-213-ADM	Protective devices should be placed over local level/trip switches that can directly result in turbine or reactor trips.	Document No. 47-1168891-00, Sept. 1987, "Review of Reactor Trip Initiating Events at the B&WOG Plants 1980-1986," p. V-2, V-3, V-7, and V-8	SSER 10.3
TR-214-CRD	Revise CRD malfunction procedure to provide instructions for how to stop recommended control rod group insertions.	Document No. 47-1168891-00, Sept. 1987, "Review of Reactor Trip Initiating Events at the B&WOG Plants 1980-1986," p. V-82. Transient Assessment Program Report No. ANO-83-01, Sept. 1982	SSER 10.3
TR-215-RCP	Ensure low RC pressure ES activation does not isolate essential pump services.	Document No. 47-1164768-00, "Operator Maintenance Personnel Interview Project Compilation of Responses"	SSER 4.4

Table 12.2 SPIP recommendations rejected by the BWOG steering commmittee

Note: The 21 recommendations listed here were rejected by the BWOG steering committee; thus, permanent numbers were never assigned to the recommendation and working (temporary) numbers identify the rejected recommendation. The source document cited by the steering committee as its basis for rejection, the date of the steering committee meeting at which the recommendation was rejected, and the SER or SSER section that treats the rejection are given.

No.	Subject	Source document	Basis for rejection (date)	SER/SSER Section
16	Evaluate the adequacy of plant communication systems, to include: a. The ability to contact plant auxiliary operators following a trip. b. The availability of communications equipment at key operating locations, such as the EFW and TBV control stations and other components/locations important to the control of FW and steam flow where manual control actions may be required.	BAW-1919, Appendix I, "Review of Category 'B' and Category 'C' Events at B&WOG Plants, 1980-1985," 47-1165733-00, September 1986, p. III-18.	This item was addressed on a plant-by-plant basis for the 1154 Task Force and is documented on p. 3-40 of that report. (October 28-29, 1986)	SER 4.2
22	Establish a program to reduce reactor trips and improve plant post-trip response. The program should focus on two key areas: a. Identification and correction of recurring problems at their plant. b. Preclusion of problems with potentially severe consequences based on the collective experience of the BWOG plants.	*Ibid*, p. III-25. program is being conducted.	This is the purpose of SPIP and the (October 28-29, 1986)	SER 4.2

Table 12.2 (Continued)

No.	Subject	Source document	Basis for rejection (date)	SER/SSER Section
23	The Transient Assessment Committee should review the TAP Report guidelines to determine what additional information is necessary to document and communicate the transient experience and lessons learned. Areas for potential inclusion in TAP Reports are: a. Listing of events of a similar nature. b. Discussion of transient response had the event occurred under different initial conditions, such as: startup mode vs. full power; low decay heat vs. high; backshift vs. day shift.	Ibid, p. III-25	This review is within the capability of the Transient Assessment Committee. (October 28-29, 1986)	SER 4.2
25	The BWOG should undertake a study of reactor trips following outages or previous reactor trips (repeat reactor trip) to identify the causes of, and recommendations that could be made to reduce the number of, such reactor trips.	Ibid, p. III-28	This study is within the capability of the Transient Assessment Committee. (October 28-29, 1986)	SER 4.2
A-1	1. Establish a BWOG Task Force to develop and implement an MSSV performance improvement program as described in Section 12.1 of the SPIP report. "Main Steam Pressure Control Review." 2. Based on the results of Phase I, raise MSSV design pressure if necessary. 3. Based on monitoring and evaluation of future performance, initiate work to reduce lifting of MSSVs.	BAW-1919, Appendix N, "B&WOG Availability Committee: Main Steam Pressure Control Review," Section 12.1, 47-1165943-00, p. 12-2.	A Valve Task Force was established to handle these and other valve-related issues. (January 14-15, 1987)	SER 6.4

Table 12.2 (Continued)

No.	Subject	Source document	Basis for rejection (date)	SER/SSER Section
5.2.7	The compressors should be isolated and/or separated as part of the redesigning and upgrading of the air supply system.	BAW-1919, Appendix M, "Instrument Air System Review Report," Section 5.2.7, 47-1165965-00, December 1986, p. 5-4	Other recommendations deal with improving response on loss of air. This is not a feasible solution.	SER 6.5
5.2.9	The status of leakage in the air system can be determined by observing the run times of the compressors and the discharge from the receivers. However, some plants have the compressors located in different places throughout the buildings. This layout makes it difficult to quickly determine the total air flow in a loss-of-air event. Flowmeters should be installed to enhance system diagnostics.	Ibid, Section 5.2.9, p. 5-4.	This is not feasible or practical. (January 14-15, 1987)	
3b	If EFW flow-limiting means are not adopted, analysis and tests should be performed to demonstrate, if possible, that a head adequate to maintain natural circulation of reactor coolant is developed by injection of emergency feed, without raising the level of condensed liquid in the steam generator above its normal low level value (roughly 5% full).	BAW-1919, Appendix P, "A Comparative Study of the Sensitivity of B&W Reactor Plants," Vol. II, MPR 948, March 1987, p. B-13.	Recommendation is not justified, since it is important to limit the maximum EFW by such means, as for example, cavitating venturis, an automatic control based on obtaining a pre-selected EFW flowrate, or incorporation into existing automatic level controls, of feedback from one of the existing EFW flow instruments. (May 5-7, 1987)	SER 5.4
2.1.7	Although each utility has taken action to reduce the propensity for turbine overspeed trips, further actions are recommended to enhance resistance. It is recommended that the BWOG undertake some or all of the following activities:	BAW-1919, Appendix Q, "Auxiliary/Emergency Feedwater System Review Final Report," 47-1168159-00, May 1987, p. 2-4.	Not necessary beyond what was done by the 1154 Task Force based on operating history and AP&L and TED. (May 5-7, 1987)	SER 6.3/SSER 6.1.12

Table 12.2 (Continued)

No.	Subject	Source document	Basis for rejection (date)	SER/SSER Section
	a. Sponsor an R&D* effort to determine if more reliable controls are available, including an investigation of non-traditional sources such as aerospace turbo-pump controls.			
	b. Develop an information exchange program that provides each owner with results of successful changes made to improve the system arrangement and configuration.			
2.2.5	Training of plant personnel should be conducted specifically in areas where repetitive problems have occurred (for example, motor-operated valve (MOV) testing and maintenance, and turbine controls testing and maintenance). Repetitive problems may also be indicative of design weaknesses, and design reviews may be necessary.	Ibid, p. 2-6	Although this is a good idea, it is not limited to EFW but is included in the SPIP recommendations covering a broader range of plant areas. (May 5-7, 1987)	SER 6.3/SSER 6.1.9
6	CR-3 experienced a steam generator overfill due to the resetting of the EFIC system trip by maintenance personnel after its inadvertent actuation. Review this event as part of the Operations, Maintenance, and I&C personnel training programs. Emphasize the need to establish communication with the control room before taking action on safety system equipment if an unplanned actuation occurs during maintenance, testing, etc.	TAP Report CR-85-01, p. 6.	This is included in general training at the utilities. (May 5-7, 1987)	SSER 6.1.10

*Research and development.

Table 12.2 (Continued)

No.	Subject	Source document	Basis for rejection (date)	SER/SSER Section
11	Provide indication of the status of the MFW pump and turbine trip bistables into ARTS/RPS in the main control room.	TAP Reports OC-2-82-01, p. 3 OC-3-84-04, p. 5; & OC-2-87-03, p. 2.	Not justified (May 5-7, 1987)	SSER 6.1.12
T-5	Implement a program that identifies the cause of turbine trips and records data for identification of pattern development.	BAW-1919, Appendix B-2, "An Evaluation of Turbine Generator Caused Reactor Trips in B&W Nuclear Power Plants," Pickard, Lowe, and Garrick Inc., PLG-0444, Section 8, December 1985, p. 6.	This is part of the Transient Assessment Committee's charter. (October 28-29, 1986)	SER 8.3
TIR-7	Include the plant response for a turbine trip runback in the operator training program.	Ibid, pp. VI-31 & VI-32	Standard industry practice is to retrain after modifications. (September 9-10, 1987)	SSER 10.3
TIR-8	Review ICS tuning to ensure control settings are compatible with a turbine trip runback from less than 45% power.	Ibid, pp. VI-31 & VI-32.	ICS tuning is not a function of ARTS setpoint. Retuning is accomplished by TR-107-ICS. (September 9-10, 1987)	SSER 6.1.8, 10.3
TIR-19	The plants that trip both MFW pumps on high SG level should evaluate sequential vs. simultaneous MFW pump trip.	Ibid, pp. VI-29, VI-36, & VI-37.	Trip is the correct consequence for this situation. (September 9-10, 1987)	SSER 6.1.10, 10.3
TIR-24	The BWOG I&C Committee should develop a post-trip or post-transient troubleshooting procedure to aid in diagnosing ICS module failures.	Ibid, pp. V-70 & V-71	This has no benefit to reduce trips or complex transients (September 9-10, 1987)	SSER 6.1.6

Table 12.2 (Continued)

No.	Subject	Source document	Basis for rejection (date)	SER/SSER Section
TIR-26	Review and upgrade maintenance procedures as necessary to ensure that proper check-out of replacement modules and assemblies in the CRDS is made prior to their use. This recommendation should also be extended to ICS components and EHC components, circuit boards, etc.	Ibid, pp. V-82 & V-83.	Not a feasible or practical solution to the problem. (September 9-10, 1987)	SSER 6.1.8, 10.3
TIR-28	Each BWOG plant should review it CRD cabinets for proper labeling and consistency with procedures.	Ibid, pp. V-80 through V-84.	This industry issue is being addressed on a braoder plant-wide basis. (September 9-10, 1987)	SSER 10.3

APPENDIX A

BIBLIOGRAPHY

BAW-1919	Babcock & Wilcox Owners Group, "Safety and Performance Improvement Program," BAW-1919, Revision 4, April 1987, and Revision 5, July 22, 1987.
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BWOG 47-1168891-00	Babcock & Wilcox Owners Group, "Review of Initiating Events at the BWOG Plants, 1980-1986," BWOG Report 47-1168891-00, September 1987.
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IE Bulletin 79-27	U.S. Nuclear Regulatory Commission, "Loss of Non-Class 1E Instrumentation and Control Power System Bus During Operation," IE Bulletin 79-27, November 30, 1979.
IE Bulletin 80-04	U.S. Nuclear Regulatory Commission, "Analysis of a PWR Main Steam Line Break With Continued Feedwater Addition," IE Bulletin 80-04, February 8, 1980.
IEEE Standard 279-1971	Institute of Electrical and Electronics Engineers, "Criteria for Protection Systems for Nuclear Power Generating Stations."
NUREG-0667	U.S. Nuclear Regulatory Commission, "Transient Response of Babcock & Wilcox-Designed Reactors," NUREG-0667, May 1980.
NUREG-0737	U.S. Nuclear Regulatory Commission, "Clarification of TMI Action Plan Requirements," NUREG-0737, November 1980.
NUREG-0800	U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," NUREG-0800, July 1981.
NUREG-0844	U.S. Nuclear Regulatory Commission, "NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity," draft report for comment, April 1985.
NUREG-1154	U.S. Nuclear Regulatory Commission, "Loss of Main and Auxiliary Feedwater Events at the Davis-Besse Plant on June 9, 1985," NUREG-1154, July 1985.

NUREG-1177	U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Restart of the Davis-Besse Nuclear Power Station, Unit 1, Following the Event of June 9, 1985," NUREG-1177, June 1986.
NUREG-1195	U.S. Nuclear Regulatory Commission, "Loss of Integrated Control System Power and Overcooling Transient at Rancho Seco on December 25, 1985," NUREG-1195, February 1986.
TAP Report No. ANO-1 85-06	Babcock & Wilcox Owners Group, Transient Assessment Report (TAP) for Arkansas Nuclear One, Unit 1, "Reactor Trip on MFW ARTS Trip," January 23, 1986.

APPENDIX B

ABBREVIATIONS

ABT	automatic bus transfer
ACRS	Advisory Committee on Reactor Safeguards
AFW	auxiliary feedwater
ANO-1	Arkansas Nuclear One, Unit 1
AP&L	Arkansas Power & Light Co.
ARTS	anticipatory reactor trip system
ASME	American Society of Mechanical Engineers
ATOG	Abnormal Transient Operator Guideline
ATWS	anticipated transient without scram
B&W	Babcock & Wilcox
BOP	balance of plant
Btu	British thermal unit
BWOG	Babcock & Wilcox Owners Group
CE	Combustion Engineering
CR-3	Crystal River Nuclear Plant, Unit 3
CRD	control rod drive
CST	condensate storage tanks
D-B	Davis-Besse Nuclear Power Station
ECCS	emergency core cooling system
EFIC	emergency feedwater initiation and control
EFW	emergency feedwater
EHC	electrohydraulic control
EOP	emergency operating procedure
ESF	engineered safety feature(s)
ESFAS	engineered safety feature(s) actuation system
FMEA	failure modes and effects analysis (analyses)
FOGG	feed only good generator
FPC	Florida Power Corporation
FSAR	Final Safety Analysis Report
FW	feedwater
GE	General Electric
GI	generic issue
GL	generic letter
GPUN	General Public Utility Nuclear Corp.
HPI	high-pressure injection
HSPS	heat sink protection system

I&C	instrumentation and control
ICS	integrated control system
IE	NRC Office of Inspection and Enforcement
IEB	IE Bulletin
IEIN	IE information notice
IGSCC	intergranular stress corrosion cracking
KSS	known safe state
LER	licensee event report
MCR	main control room
MDFWP	motor-driven feedwater pump
MFW	main feedwater
MFWP	main feedwater pump
MOV	motor-operated valve
MPR	MPR Associates
MSIV	main steam isolation valve
MSSV	main steam safety valve
NNI	non-nuclear instrumentation (system)
NRC	U.S. Nuclear Regulatory Commission
NSAC	Nuclear Safety Analysis Center
ORNL	Oak Ridge National Laboratory
OTSG	once-through steam generator
PIC	previously identified concerns
PORV	power-operated relief valve
PSM	power supply monitor
PTS	pressurized thermal shock
PWR	pressurized-water reactor
RC	reactor coolant
RCIC	reactor core isolation cooling
RCP	reactor coolant pump
RCPPM	reactor coolant pump power monitor
RCS	reactor coolant system
RHR	residual heat removal
RPS	reactor protection system
RTB	reactor trip breakers
RTS	recommendation tracking system (BWOG)
SAIC	Science Applications International Corporation
SASS	smart analog signal selector
SER	safety evaluation report
SFAS	safety features actuation system
SFRCS	steam and feedwater rupture control system
SG	steam generator
SGTR	steam generator tube rupture
SMUD	Sacramento Municipal Utility District

SPDS	safety parameter display system
SPIP	Safety and Performance Improvement Program
SPRIG	Safety and Performance Recommendation Integration Group
SRP	Standard Review Plan
STA	shift technical advisor
TAP	Transient Assessment Program
TED	Toledo Edison
TMI	Three Mile Island Nuclear Station
ULD	unit load demand
USI	unresolved safety issue
UT	ultrasonic testing
W	Westinghouse

APPENDIX C

NRC STAFF CONTRIBUTORS AND CONSULTANTS

This supplement is a product of the NRC staff and its contractors. The staff members listed below were principal contributors to this report. A list of contractors follows the list of staff members.

NRC STAFF

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APPENDIX E

PREVIOUSLY IDENTIFIED ICS/NNI CONCERNS*

Concern	Source Document	Guidance/Recommendation
1. Adequacy of power distribution system fault protection	SMUD report June 1978	Fuse all loads powered from NNI-X and NNI-Y buses and evaluate the possibility of using lower size fuses.
2. Failure of controls that could initiate transient or inhibit mitigation of transient	NUREG-0560 May 1979	Re-evaluate control systems and their significance to safety.
3. Steam generator overfill and overcooling resulting from ICS failure	BAW-1564 Sec. 3.2.c Aug 1979	Review for possible changes to enhance reliability and safety by preventing or mitigating the consequences of an inadvertently opened feedwater startup valve and by providing means to prevent or mitigate the consequences of an open turbine bypass valve.
4. Loss of power to controls and to control room instrumentation	IE Bulletin 79-27 Nov 1979	Review buses supplying power to I&C systems that could affect the ability to achieve cold shutdown; identify I&C loads and evaluate loss of power to these loads; describe proposed design modifications.
5. Consequences of loss of NNI power	NRC Generic Letter Mar 1980	Address the proposed corrective actions identified following the Feb 26, 1980 loss of NNI power event at CR-3 in terms of applicability to other B&W plants and expand the IE Bulletin 79-27 review to include implications of the CR-3 event.

*These are concerns typically common to the more significant B&W plant transients involving the ICS/NNI systems.

Concern	Source Document	Guidance/Recommendation
6. Loss of power supply	Industry review of CR-3 event NSAC-3/INPO-1 Rec. III.A.1 Mar 1980	Investigate the need for backup or bus transfer capabilities upon loss of instrumentation and control power supplies resulting from faults.
7. Close electrical coupling	NSAC-3/INPO-1 Rec. III.A.2 Mar 1980	Investigate and take corrective action regarding coupling of indication, control, and computer input signals.
8. Spurious control system actions	NSAC-3/INPO-1 Rec. III.A.4 Mar 1980	Investigate and take corrective action regarding control system susceptibility to spurious/inappropriate actions based on incorrect information (caused by electrical faults).
9. Electrical cross-contamination of instrument channels and also of controls	NSAC-3/INPO-1 Rec. III.A.5 Mar 1980	Investigate (including field tests) to identify problems concerning cross-contamination of power supplies affecting redundant channels of instrument and control functions.
10. Effects of loss of single bus	NSAC-3/INPO-1 Rec. III.A.7 Mar 1980	Investigate and take corrective action regarding assignment of instruments to specific buses to ensure as much redundancy as possible.
11. Impact of single bus failure	NUREG-0667 Rec. 5.a May 1980	Power buses and signal paths for ICS and NNI should be separated and channelized.
12. Undesirable failure modes (of power source, signal source, and ICS itself) causing substantial plant upsets	NUREG-0667 Rec. 5.d May 1980	ICS should have provisions for detecting gross failures and taking appropriate defensive action.
13. Implementation of existing recommendations	NUREG-0667 Rec. 5.h May 1980	Prompt action should be taken on IE Bulletin 79-27.
14. Avoiding unnecessary failure mechanisms	IE Information Notice 84-80 Nov 1984	Consider scheduling calibration of NNI and ICS dc power supply alarm and trip set points every refueling outage.

Concern	Source Document	Guidance/Recommendation
15. Availability of minimum indicators for safe shutdown	IE Bulletin 79-05A Apr 1979	Ensure that indications for the minimum set of plant variables needed for safe shutdown will be available.
16. Alarm on loss of I&C bus power	IE Bulletin 79-27 Nov 1979	Review buses supplying power to I&C systems that could affect ability to reach cold shutdown; identify and review for each bus the alarm/ indications provided in the control room to alert operators to the loss of power to the bus; describe any proposed design modifications.
17. Loss of control room indications	NRC Generic Letter Mar 1980	Address information available to the operator, including how the opertor determines which information is reliable, and identify what information is needed to bring the plant to cold shutdown.
18. Indications of lost control functions	NRC Orders Apr 1980	Make changes in equipment and control systems to give clear indications of functions that are lost or unreliable.
19. Numerous and complicated operator action necessary	NUREG-0667 Rec. 11 May 1980	Modifications should be made to reduce post-reactor trip immediate manual actions.
20. Confusing or misleading indications to operator	NUREG-0667 Rec. 5.b May 1980	Power supply arrangements should be reconsidered to eliminate "mid-scale" failure mode for instrumentation.
21. Backup instrumentation	NUREG-0667 Rec. 5.c May 1980	Multiple instrument failures should be unambiguously indicated to guide operator selection and use of alternate instrumentation unaffected by the failure(s).
22. Adequacy of remaining or backup instrumentation	NUREG-0667 Rec. 6 May 1980	Establish the minimum set of parameters needed (recommended list include); instrumentation for these should be reliable, redundant, and meet

Concern	Source Document	Guidance/Recommendation
		safety system standards; at least one channel for each parameter shall be recorded for trending etc.
23. Procedures needed for loss of power to I&C bus	IE Bulletin 79-27 Nov 1979	Ensure adequate emergency procedures exist to achieve cold shutdown upon loss of power to each bus supplying power to I&C systems; include: (a) alarms; and indicators of the loss, (b) use of alternate instruments/controls powered by other buses, (c) restoration of power to bus; describe any design modifications or administrative controls necessary to implement such procedures.
24. Implementation of existing recommendations	NUREG-0667 Rec. 14 May 1980	Develop and promptly implement plant-specific emergency procedures for loss of NNI or ICS power; B&W development of ATOG endorsed; full utility support for ATOG recommended.
25. Adequacy of safe shutdown capability	SMUD Report June 1978	Procedures should be developed for safe shutdown upon total loss of NNI-X or NNI-Y power.

APPENDIX F

NRC LETTERS TO THE BWOG CONCERNING THE ICS/NNI REASSESSMENT

1. Letter dated June 24, 1986, from D. M. Crutchfield (NRC) to C. Doyel (BWOG I&C Committee). This letter provides staff feedback to the BWOG concerning the ICS/NNI reassessment portion of the SPIP based on information presented by the BWOG during meetings with the staff on April 29 and May 21, 1986.

2. Letter dated August 1, 1986, from D. Crutchfield (NRC) to H. B. Tucker (BWOG Executive Committee). This letter requested clarification of the BWOG reassessment program with regard to retrospective issues raised by the Incident Investigation Team in its report on the loss of ICS power event at Rancho Seco on December 26, 1985 (NUREG-1195).

3. Letter dated September 12, 1986, from D. M. Crutchfield (NRC) to C. Doyel (BWOG I&C Committee). This letter provided staff feedback to the BWOG concerning the ICS/NNI reassessment based on information provided in preliminary documents developed by the I&C Committee and discussed during a meeting with the I&C Committee on August 26, 1986.

4. Letter dated December 17, 1986, from D. M. Crutchfield (NRC) to H. B. Tucker (BWOG). This letter provided a list of concerns identified during the staff's review of the BWOG SPIP, as documented in BAW-1919 and its revisions, and requested that the BWOG address these concerns in a future update of BAW-1919.

5. Letter dated February 13, 1987, from D. M. Crutchfield (NRC) to L. Stalter (BWOG I&C Committee). This letter provided staff feedback to the BWOG concerning the ICS/NNI reassessment based on information provided in revised documents developed by the I&C Committee and discussed with the I&C Committee during a meeting on December 10, 1986.

6. Letter dated April 7, 1987, from D. M. Cruchfield (NRC) to G. R. Skillman (BWOG). This letter provided staff comments and questions concerning the BWOG ICS/NNI reassessment and requested that the BWOG address the associated concerns in a future update of BAW-1919.

7. Letter dated November 24, 1987, from J. A. Calvo (NRC) to R. P. Rogers (BWOG). This letter requested additional information needed for the staff to complete its review of Appendix R, "ICS/NNI Evaluation Final Report," to BAW-1919, "B&W Owners Group Safety and Performance Improvement Program (SPIP)."

APPENDIX G

ERRATA

The following errata apply to the "Safety Evaluation Report Related to Babcock & Wilcox Owners Group Plant Reassessment Program."

Page 9-43, Table 9.22	Change first column:	"C" to "C_5"
		"T" to "T_4"
		"D" to "D_2"
		"T" to "T_2"
		"C" to "C_3"
		"T" to "T_1"
		"D" to "D_1"
		"C" to "C_2"
		"D" to "C_4"
		"C" to "C_1"
		"T" to "T_3"
		"D" to "D_3"
	Change Davis-Besse Score "39" to "9"	

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Supplement No. 1 to the Safety Evaluation Report (SER) related to the Babcock & Wilcox Owners Group (BWOG) Plant Reassessment Program has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. This supplement contains the NRC staff's evaluation of the BWOG reassessment of the integrated control system/non-nuclear instrumentation system, the emergency feedwater initiation and control system, reactor trip initiating events, several additional open items identified in the SER, and the BWOG comments on the SER.

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