October 5, 1989

Docket No. 50-289

Mr. Henry D. Hukill, Vice President
and Director - TMI-1
GPU Nuclear Corporation
P. O. Box 480
Middletown, Pennsylvania 17057

Dear Mr. Hukill:

SUBJECT: SAFETY AND PERFORMANCE IMPROVEMENT PROGRAM (SPIP) IMPLEMENTATION AUDIT AT THREE MILE ISLAND UNIT 1 (TMI-1) (TAC NO. 68206)

REFERENCE: Letter from Dennis M. Crutchfield, NRC, to Walter S. Wilgus, BWOG, dated May 4, 1988, "Status of the evaluations of previous NRC requirements, recommendations and concerns applicable to B&W designed plants."

As you are aware, the NRC staff has determined that a series of audits is necessary to verify the proper implementation of the Babcock & Wilcox Owners Group (BWOG) Safety and Performance Improvement Program (SPIP) recommendations at each B&W designed facility. This audit series consists of a SPIP programmatic audit (which has been completed already), a SPIP recommendation implementation audit, and a followup audit, if necessary. The implementation audit for TMI-1 is planned for the week of April 2, 1990. In conjunction with this audit the audit team will also evaluate your disposition of previous NRC issues applicable to B&W designed plants (See the Enclosure to this letter).

It is anticipated that the audit will commence with an entrance meeting at 9:00 am on April 2, 1990 and conclude with an exit meeting on April 5, 1990. We plan to conduct the audit at the TMI site in Middletown, PA. I will firm up specifics with your staff as the week of the audit approaches.

Sincerely,

/s/

Ronald W. Hernan, Senior Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosure: As stated

cc w/enclosure:
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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STATUS OF EVALUATIONS OF PREVIOUS NRC REQUIREMENTS, RECOMMENDATIONS, AND CONCERNS APPLICABLE TO BABCOCK & WILCOX-DESIGNED PLANTS

The staff has performed computer searches using the Babcock & Wilcox (B&W) docket numbers and key words from specific subjects addressed in the Babcock & Wilcox Owners Group (BWOG) "Safety and Performance Improvement Program" (SPIP) report, BAW-1919, to identify the documents the NRC staff believes should have been reviewed during the SPIP review. In addition, the staff has identified safety-related issues that should have been evaluated on a routine basis by the B&W utilities. 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 have been adequately covered. In addition, previous concerns related to the integrated control system/non-nuclear instrumentation (ICS/NNI) system are not included. These were addressed in Appendix E of the SSER, issued March 1988.

The documents identified in the computer search that contain recommendations and identify concerns are primarily NUREG reports, NRC orders, and Inspection and Enforcement bulletins, circulars, and information notices. This enclosure contains the issues derived from these documents. In addition, Nuclear Safety Analysis Center - 3 (March 1980) recommendations are included where appropriate.

This information has been divided into the eight categories listed below.

(1) reactor coolant system and emergency core cooling systems
(2) safety-related electrical systems and/or components
(3) instrument auxiliary systems
(4) decay heat removal system
(5) valves (including safety/relief)
(6) main feedwater system
(7) auxiliary/emergency feedwater system
(8) administrative controls

For simplification and to avoid duplication, each issue that had been identified was placed in what appeared to be the most applicable category (i.e., one entry).

The results of this search are presented in the table below, which provides a number column that categorizes the issue for easy reference, a source column that identifies the reference document, and an issue column that describes the requirements, recommendations, or concerns contained in the source document. The staff requests that the BWOG in conjunction with the individual utilities where required, provide for each entry a response that describes the action taken to address each issue and identifies the document or method by which the issue was addressed. Each entry should be referenced by category, number, and, if applicable, item.

CATEGORY 1: REACTOR COOLANT SYSTEM AND EMERGENCY CORE COOLING SYSTEMS

No.	Source	Issue
1-1	NUREG-0667 Recommendation (Rec.) 2.2 (9) May 1980	This report recommends the following: • Following reactor trip, pressurizer level should remain on scale and pressurizer pressure should remain above the high-pressure injection (HPI) actuation set point. • These objectives should be met independent of all manual operator actions (e.g., control of feedwater, letdown isolation, startup of a makeup pump).
	Rec. 2.2 (11)	Plant modifications should be made to reduce or eliminate manual immediate actions from emergency procedures.
1-2	NRC Office of Inspection and Enforcement Information Notice (IEIN) 86-19 Mar. 21, 1986	This notice concerns failure of the reactor coolant pump (RCP) shaft at Crystal River. The initial indications were motor frame vibration and RCP thrust bearing high temperature; the operators manually tripped the RCP. Further symptoms related to such an event are listed below. • A shaft fracture was found at a nonfunctional groove below thermal barrier. • Ultrasonic testing (UT) of three other RCPs showed cracks at same locations. • All four capscrews joining the shaft to impeller had broken as a result of intergranular stress corrosion cracking (IGSCC). • UT at Davis-Besse found cracks in one RCP and probable cracks in three other RCPs. • Similar capscrew failures to those above were found on the RCPs at the Palisades plant because the prescribed preloading could not be achieved because of rough threads.
1-3	IE Bulletin IEB) 86-03 Oct. 8, 1986	This bulletin addresses the potential failure of multiple pumps of the emergency core cooling system (ECCS) as a result of the single failure of the air-operated valve in the minimum-flow recirculation line.
	Item 1	Promptly determine whether or not the facility has a single-failure vulnerability in the minimum flow recirculation line of any ECCS pumps that could cause the failure of more than one ECCS train.

No.	Source	Issue
	Item 2	If the problem exists: • Promptly instruct all operating shifts of the problem and measures to recognize and mitigate the problem. • Promptly develop and implement corrective actions to bring the facility into compliance with General Design Criterion (GDC) 35.

CATEGORY 2: SAFETY-RELATED ELECTRICAL SYSTEMS AND/OR COMPONENTS

No.	Source	Issue
2-1	IEB 80-06 Mar. 13, 1980	This bulletin addresses engineered safety feature (ESF) reset controls and recommends that the licensee:
	Item 1	Review the drawings for systems serving safety-related functions at the schematic level to determine whether or not all associated safety-related equipment remains in its emergency mode if the ESF actuation signal is reset.
	Item 2	Verify that the actual installed instrumentation and controls at the facility are consistent with the drawings reviewed by conducting a test to demonstrate that all equipment remains in its emergency mode if the actuating signal is removed and/or the various isolating or actuation signals are reset manually.
	Item 3	If any safety-related equipment does not remain in its emergency mode when an ESF signal is reset, describe the proposed system modification, design change, or other corrective action planned to resolve the problem.
2-2	NUREG-0667 Rec. 2.2 (4) May 1980	This report indicates that: • Steam line break detection and mitigation systems should be modified as necessary to eliminate adverse interactions with the auxiliary feedwater (AFW) system. • Steam line break detection and mitigation systems should be re-evaluated and modified so that they are capable of differentiating between an actual steam line break and undercooling/overcooling caused by feedwater transients.
2-3	IEB 80-16 June 27, 1980	This b[illegible]etin covers misapplication of pressure transmitter[illegible] indicates that the licensee:

No.	Source	Issue
	Item 1	Determine if the facility has installed or plans to install Rosemount Inc. Model 1151 or 1152 pressure transmitters with output codes "A" or "D" in any safety-related application.
	Item 2	If it is determined that the facility has the transmitters described in item 1 above in any safety-related application, determine whether they can be exposed to input pressures that could result in anomalous output signals during normal operation, anticipated transients, or design-basis accidents. If affected transmitters can be exposed to input pressures that could result in anomalous output signals, perform a worst-case analysis to determine whether the anomalous signals could result in violating any design-basis assumption. The safety-related application shall include control, protective, or indication functions. If any safety-related application does not conform to these requirements, address the basis for continued plant operation until the problem can be resolved and provide an analysis of all potential adverse system effects that could occur as a result of the postulated pressure transmitter maloperation described in enclosure 1 of the bulletin. In each instance, the analysis should include the effects of the postulated transmitter maloperation as it relates to indication, control, and protective functions. The analysis shall address both incorrect automatic system operation and incorrect operator actions caused by the erroneous indications. Address conformance to Institute of Electrical and Electronics Engineers (IEEE) Standard 279, Section 4.20, in the analysis. The analysis should include the specific information required by the bulletin.
	Item 3	Submit a complete description of all corrective actions required as a result of the analyses and evaluations, along with the schedule for accomplishing the corrective actions.
2-4	IEIN 86-10 Feb. 13, 1986	This notice alerts licensees to the concern of safety parameter display system (SPDS) operability. It was found that two out of five plants had declared the SPDS to be operational when the SPDS was not operational. The major deficiencies that were found in the SPDS survey are listed below. • lack of availability because of gross system malfunctions • display of unreliable or invalid data and alarms

No.	Source	Issue
		• poor acceptance by operators because of reliability problems • management failure to integrate SPDS into operational environment • inadequate documentation and failure to control system testing and modifications • slow system response to some operator commands.
2-5	IEB 79-25 Nov. 2, 1979 Item 1	This bulletin requires licensees to determine if Westinghouse model BFD/NBFD relays of the type identified in the bulletin are used at their facilities. If they are used, identify the safety-related systems involved, the function of the relays, and plans for test and/or replacement programs.
	Item 2	Establish a program to ensure performance of affected relays. This program should include periodic testing and/or replacement, the basis for test intervals, development of approved procedures for testing and/or replacement, and documentation of relay failures found during testing.
	Item 3	A written report should be submitted to address the actions taken under items 1 and 2 above.
2-6	IEB 79-09 Apr. 17, 1979 Item 1	This bulletin requires licensees to determine if GE type AK-2 breakers are used at their facilities and, if used, identify the safety system involved and the plans for developing a preventive maintenance program.
	Item 2	The preventive maintenance program should include a preventive maintenance schedule, the necessary qualifications for personnel to perform the maintenance, and the status of the recommended corrective actions described in GE Service Alert Letter No. 175.
	Item 3	A written report should be submitted to address the actions taken under items 1 and 2 above.
2-7	IE Circular (IEC) 81-12 July 22, 1981	This circular urges licensees to review the specific items presented in the "Description of Circumstances" Section of this IEC as they relate to circuit breakers. Also review the procedure for surveillance testing of circuit breakers to ensure that the procedure provides for independent testing of each trip function. If the procedure does not have provisions for independent testing of each trip function, then modifications should be made to include such features.

No.	Source	Issue
2-8	IEC 81-14 Nov. 5, 1981	This circular urges licensees to review operating experience with main steam isolation valves (MSIVs) to identify problems related to the failure of the valves to close and to equipment degradation that prevents a valve from closing and would require other than routine maintenance to correct. Evaluate the corrective actions identified in the maintenance records to ensure that the actions were adequate to solve the root-cause problems; if not adequate, develop plans for further corrective action. If air quality control is suspected of contributing to problems with the MSIVs, review the air system to ensure that measures have been or will be taken to prevent future air quality degradation and consider the installation of monitors and/or alarms to provide warning of air quality deterioration. If system binding is suspected of contributing to problems with the MSIVs, review maintenance procedures to ensure that they include precautions to be taken against detrimental effects (such as those caused by inappropriate lubricants) and that the procedures include tests that demonstrate the valves will perform under operating conditions before being placed in service.
2-9	IEIN 82-35 Aug. 25, 1982	This notice identifies a concern that stop check valves manufactured by Velan would fail to pass flow because of maintenance and design problems.
2-10	IEIN 82-48 Dec. 3, 1982	This notice identifies potentially significant deficiencies with Agastat CR 0095 relay sockets. In 1979 General Electric tested 2200 of these relay sockets and found a significant number exhibited contact retention problems and potential electrical connection problems. These relay sockets were redesigned and modified by the manufacturer and all relay sockets shipped after March 1979 were of the new type. Another means of correcting the problem in the old relay sockets is to install cardboard insulator strips behind the relay sockets.
2-11	IEIN 82-50 Dec. 20, 1982	This notice identifies a potentially significant problem pertaining to misapplication of solid-state ac undervoltage relays type ITE-27, Series 211B and 211L, manufactured by Brown Bovery Electric Inc. These relays were found to be used on Class 1E switchgear that requires a source of dc control power for proper operation and these relays were used in some plants to monitor ac bus undervoltage conditions. Instead,

No.	Source	Issue
		ITE-27, Series 211R, relays should be used since they do not drop out on loss of dc power and do not result in an inadvertent isolation of Class 1E switchgear.
2-12	IEIN 82-54 Dec. 27, 1982	This notice identifies potentially significant problems with a certain batch of Westinghouse NBFD relays that appear to have a higher-than-expected failure rate. The notice contains a copy of a Westinghouse technical letter that discusses the problem and provides inspection and test methods for verifying operability of the relays as well as suggested corrective action.
2-13	IEB 83-04 Mar. 11, 1983 Item 1	This bulletin requires PWR licensees with other than WDB-type breakers in the reactor protection system (RPS) to: • Perform surveillance tests of the undervoltage trip function that are independent of tests of the shunt trip function. • Review their maintenance program to ensure conformance with manufacturer's recommendation. • Ensure that the appropriate emergency operating procedures for the event of failure-to-trip and other operating events are reviewed with each operator. • Provide a written report containing results of the above actions, a description of all RPS breaker malfunctions that have not been reported previously, and verification that procurement, testing, and maintenance activities related to the RPS breaker and undervoltage devices are treated as safety related.
2-14	IEIN 83-76 Nov. 2, 1983	This notice suggests that utilities using General Electric type AK-2-25 breakers with undervoltage trip devices, visually inspect each undervoltage armature to ensure it is in its proper position after each operation (i.e., the fully down position and not the mid position).
2-15	IEB 84-02 Mar. 12, 1984 Item 1	This bulletin requires licensees to: Develop plans and schedules for replacing nylon or Lexan coil spool-type HFA relays that are used in energized safety-related applications and nylon coil spool-type HFA relays that are used in normally de-energized applications.
	Item 2	In the interim, before the relays are replaced, develop and implement surveillance plans that include monthly functional tests and visual inspections.

No.	Source	Issue
	Item 3	Provide the basis for continuing operation until the normally energized relays are replaced.
	Item 4	Provide a written report describing the above actions and including completion schedules.
2-16	IEIN 84-20 Mar. 21, 1984	This notice identifies a problem pertaining to the service life of relays in safety-related systems. It specifically identifies the earlier-than-anticipated, end-of-service life failures of Agastat G-P series relays and Sylvania GTE ac relays. The notice suggests that utilities review their safety-related systems to determine if these relays have been installed or are being held as spare parts. Preventive maintenance programs should recognize the application-dependent (energized/de-energized) service life of relays and the current surveillance interval should be compared with the service life of the relays as used in the system to determine if it is acceptable. The notice indicated that these problems are similar to those discussed in IEB 84-02 and the general concerns associated with HFA relay failures discussed in that bulletin apply.
2-17	IEIN 84-37 May 10, 1984	This notice provides the following guidance to eliminate problems encountered in the use of lifted leads and jumpers during maintenance and surveillance testing: • Install permanent test hardware. • Include additional procedural checks of system configuration during surveillance and maintenance testing. • Review procedures to ensure instructions for surveillance and maintenance clearly specify the reconnection of any lifted leads and the removal of any jumpers. • Use at least two qualified operators to independently verify proper system configuration before safety-related equipment is returned to service. • Perform functional tests to verify proper system configuration is restored before safety-related equipment is returned to service. • Review with operators and maintenance personnel specific instances of errors involving lifted leads or jumpers and the safety impact of such errors.
2-18	IEIN 85-58 July 17, 1985 Supplement 1 Nov. 19, 1985	This notice identifies a potentially significant problem pertaining to the failure of General Electric (GE) type AK-2-25 reactor trip breakers (RTBs) that are installed in facilities designed by B&W and Combustion

No.	Source	Issue
		Engineering (CE). During a test at Rancho Seco one of the RTBs failed to trip open when its undervoltage attachment was actuated. The RTBs at Rancho Seco that were sent to GE for refurbishment, were only visually inspected when returned. The utility had developed procedures to perform checks of critical parameters of breakers based on B&W guidance. Supplement 1 to this notice identifies additional GE type AK-2-25 breaker failures resulting in slow closure times that occurred at Calvert Cliffs and Oconee. These failures were related to laminated sections of the armature that slipped down causing contact between the laminations and pole face at Calvert Cliffs. At Oconee, a new undervoltage device in the RTB reduced clearances between the armature and heads of mounting bracket studs and could have caused contact. This notice calls attention to GE's Service Advice Letter No. 300, which outlines corrective actions.
2-19	IEIN 85-93 Dec. 6, 1985	This notice alerts utilities that the electric circuit breaker closing function of Westinghouse (W) type DS circuit breakers would not operate if the spring release latch levers were broken. W issued Technical Bulletin No. NSID-TB-85-17 advising utilities of this potential malfunction. The W bulletin identifies the following corrective actions: (1) advise personnel breaker may be closed manually; (2) user should evaluate function to determine if it affects safety; and (3) inspect latch lever during normal scheduled maintenance/inspection of type DS circuit breakers.
2-20	IEIN 87-08 Feb. 4, 1987	This notice alerts utilities to potentially defective dc motors that were manufactured by H. F. Porter (now Peerless-Winsmith) between December 1984 and December 1985 and installed in Limitorque motor operators.
2-21	IEIN 87-12 Feb. 13, 1987	This notice alerts utilities to potential problems with GE type AKF-2-25 circuit breakers failing to fully open. These circuit breakers are susceptible to failures as a result of binding within the breaker cam mechanism unless proper maintenance procedures are developed and followed by trained individuals. The notice contains the following maintenance information that was provided by GE to be incorporated in utility programs for type AKF-2-25 breakers: (1) maintenance/inspection intervals and complete overhaul should be every 12 months or each refueling outage; (2) only specified lubricants should be used; (3) only qualified properly trained personnel should perform maintenance; and (4) breakers that have not yet been converted to a specified lubricant should be cycled.

No.	Source	Issue
2-22	IEIN 87-24 June 4, 1987	This notice alerts utilities to potential problems involving inverter losses. The notice refers to an NRC case study report, AEOD/C605, on operational experience involving losses of electrical inverters in which three failure mechanisms were identified. The notice suggests that utilities consider (1) monitoring temperature and/or humidity internal to inverter enclosures and input and output voltages of the inverter unit during steady-state and transient conditions and (2) reviewing maintenance and testing procedures and practices.

CATEGORY 3: INSTRUMENT AIR SYSTEMS

No.	Source	Issue
3-1	IEIN 81-38 Dec. 17, 1981	This notice informs licensees about an NRC staff review of a number of problems and instances related to contamination of air systems in operating plants. The review indicated that air-operated components and systems will occasionally become inoperable because of contamination with oil, water, desiccant, rust, or other corrosion product. The notice described the following actions, which are known to minimize air system problems: • frequently monitor the dew point of the instrument air • periodically check the desiccant cartridges to determine if they need regenerating or replacing • periodically blow down lines to remove oil, moisture, and crud in the instrument air system • periodically inspect filters downstream of the desiccant cartridges to ascertain that the desiccant has not been pulverized to the point that it is escaping from the cartridge and possibly clogging the filters • avoid using service air as a backup to the instrument air system when alternative backups are available • frequently monitor the instrument air system to ensure that it has not been contaminated with oil, moisture, or crud when service air has been used as a backup to the instrument air system

CATEGORY 4: DECAY HEAT REMOVAL (DHR)

No.	Source	Issue
4-1	IEIN 80-20 May 9, 1980	This notice describes an event at Davis-Besse that occurred while in a refueling mode and that resulted in a loss of DHR capability for approximately 2½ hours. The following three factors contributed to this loss: (1) inadequate procedures and/or administrative controls, (2) extensive maintenance activities, and (3) the two out of four safety features actuation system (SFAS) logic. It was suggested that licensees evaluate the susceptibility of their plants to lose DHR capability by these causes.
4-2	IEIN 86-39 May 20, 1986	This notice alerts licensees to a potential common-mode failure of multiple residual heat removal (RHR) pump motors and pump internals. An event occurred in which the high temperature of a lower guide bearing went unnoticed because of several other alarms--3 days later the motor caught fire. It was found that lower pump impeller wear rings on multiple RHR pumps had separated from the impeller as a result of IGSCC. Motor guide bearing failures are significant because of the potential failure of pump internals. Other potential causes of internal damage include inadequate flow and lubrication.
4-3	IEC 81-10 July 2, 1981	This circular advises licensees to: • Review their operating procedures for plant cooldown, emergency, and abnormal operations as they relate to natural circulation to ensure sufficient information is available for operators to recognize symptoms of reactor coolant system (RCS) voiding and take appropriate actions to recover from a voided condition. • Inform each licensed operator of the information discussed in the circular. • Consider including this information in operator training and retraining classes.
4-4	GL 87-12 July 9, 1987	The subject of this letter is the loss of RHR capability while the RCS is partially filled. The GL transmits NUREG-1269, a report on the Diablo Canyon loss of RHR system incident of April 10, 1987. It requests licensees to provide a description of plant operations with a partially filled RCS. The description should address such topics as the initial conditions, instrumentation and alarms, available pumps, containment closure capability, procedures, training, additional resources, requirements for other modes, any changes to be made and the schedule for making them.

CATEGORY 5: VALVES (INCLUDING SAFETY/RELIEF VALVES)

No.	Source	Issue
5-1	IEC 79-22 Nov. 16, 1979	This circular advises licensees to conduct a review to determine if periodic surveillance of power-operated relief valves (PORVs) is necessary to ensure that the PORVs will perform as intended.
5-2	IEIN 80-41 Nov. 6, 1980	This notice describes the failure of a Velan check valve in the decay heat removal system at Davis-Besse. The valve disk and arm separated from the valve body and were lodged under the valve cover plate. The bolts and locking mechanism that holds the arm to the valve body were missing.
5-3	IEB 81-02 Apr. 8, 1981 Supplement Aug. 18, 1981 Item 1	This bulletin requires licensees to ascertain whether any W-EMD motor-operated gate valves have been installed or are maintained as spares for installation in safety-related systems where they are required to close against differential pressure. If the affected valves have been installed, licensees should take corrective action and evaluate the effect on system operability if the valves fail to close. If affected valves are spares, licensees should replace or modify the valves before installation. Licensees should provide a written report listing the affected valves, service, and maximum differential pressure required to close and describing the safety consequences if the valves fail to close and the corrective actions taken or planned along with a schedule for completing these actions.
5-4	IEIN 81-35 Dec. 2, 1981	This notice addresses Metropolitan Edison's report of loose valve internals in the high-pressure injection pump discharge crane 3-inch, 1500-pound tilt check valves that resulted from corrosion of the seat holddown devices. As a result of these findings, a continuing inspection program for TMI-1 was developed and implemented which led to the discovery that some tilt check valves could not prevent back flow because the hinge pin and ring seat retention devices failed. Many valve fabrication inconsistencies also were discovered that may have initiated or contributed to the failures.
5-5	IEIN 82-20 June 28, 1982	This notice alerts licensees to a potentially significant problem pertaining to internal damage to swing check valves of the same or similar design and service as those manufactured by Alloy Steel Products Company and Pacific Company.

No.	Source	Issue
5-6	IEIN-83-57 Aug. 31, 1987	This notice alerts licensees to a potential problem with ASCO three-way solenoid-operated pilot valves, type NP-8316 in 3/8- and 1/2-inch national pipe thread sizes. The manufacturer's installation instruction bulletin, issued in 1978, provides incorrect assembly instructions for certain parts of this valve.
5-7	IEIN 84-33 Apr. 20, 1984	This notice alerts licensees to a potential problem with main steam safety valves that have failed because of cotter pin failures.
5-8	IEIN 84-66 Aug. 17, 1984	This notice identifies events where turbine-driven AFW pumps were unavailable because the steam supply was isolated (trip and throttle valve was not latched). The NRC recommends that licensees review these events and consider the following preventive actions: • design change to provide positive control room indication of a trip valve "latched" condition • regular adjustment and testing of the limit switches to ensure operability • local verification of position after resetting trip valve • visual verification daily or once per shift to see that the valve is not tripped • local mechanical valve position indication installed and permanent tags attached to the valve providing instructions for operation • on-shift training in operation of the trip valve for all personnel who are required to operate the valve • improved housekeeping to prevent fouling external valve linkages • warning sign installed near the trip lever
5-9	IEIN 84-48 June 18, 1984 Supplement 1 Nov. 16, 1984	This notice alerts licensees to a potential deficiency in the design, application, or maintenance of Rockwell International globe valves that have resulted in two types of failures: (1) the stem separating from the disk and (2) the disk being backed off its disk nut. The manufacturer attributes these failures to the high cavitation loads the valve disk experiences when used in severe throttling conditions and recommends that these valves be replaced with smaller size valves so that the valve disk will be in a more fully open position.

No.	Source	Issue
5-10	IEIN 85-35 Apr. 30, 1985	This notice identifies a potentially significant problem related to Parker-Hannifur Corporation check valves supplied by Anchor/Darling Valve Company. These valves may degrade the capability for closing main steam isolation valves (MSIVs) or feedwater isolation valves or inhibit other safety functions. An event at Byron Unit 1 resulted in the failure of two MSIVs to close on an isolation signal because the instrument air check valves failed to seat in response to gradually decreasing air pressure.
5-11	IEIN 85-59 July 17, 1985	This notice identifies a potentially significant problem related to stress corrosion failures of valve stems and shafts that are not routinely examined. Four instances are described where cracks were found in 410 stainless steel valve stems. Each instance involved a different plant and valve manufacturer, and the cracks were discovered after failure or disassembly.
5-12	IEIN 85-84 Oct. 30, 1985	This notice discusses the possible failure of MSIVs to close under low- or no-steam flow conditions and the testing of these valves with non-safety-related motive power (instrument air) in place. • determine the need for a test program to establish reliability
5-13	IEB 85-03 Nov. 15, 1985	The common-mode failures of motor-operated valves with improper switch settings during plant transients led the NRC to request licensees to: Develop and implement a program to ensure that valve operator switches are selected, set, and maintained properly for MOVs in high-pressure coolant injection/core spray and emergency feedwater systems [reactor core isolation cooling (RCIC) system for BWRs] that are required to be tested for operational readiness in accordance with 10 CFR 50.55a(g). This should include the following components: (1) Review and document the design basis for the operation of each valve. This documentation should include the maximum differential pressure expected during opening and closing the valve for normal and abnormal events to the extent that these valve operations and events are included in the existing, approved design basis (e.g., the design-basis documented in pertinent licensee submittals such as FSAR analyses and fully approved operating and emergency procedures). When determining the maximum

No.	Source	Issue
		differential pressure, those single equipment failures and inadvertent equipment operations (such as inadvertent valve closures or openings) that are within the plant design basis should be assumed.
		(2) Using the results of item (1) above, establish the correct switch settings. This shall include a program to review and revise, as necessary, the methods for selecting and setting all switches (i.e., torque bypass, position limit, overload) for each valve operation (opening and closing). If the licensee determines that a valve is inoperable, the licensee shall also make an appropriate justification for continued operation in accordance with the applicable technical specification.
		(3) Individual valve setting shall be changed, as appropriate, to those established in item (2), above. Whether the valve setting is changed or not, the valve will be demonstrated to be operable by testing the valve at the maximum differential pressure determined in item (1) above with the exception that testing MOVs under conditions simulating a break in the line containing the valve is not required. Otherwise, justification should be provided for any cases where testing with the maximum differential pressure cannot practicably be performed. This justification should include the alternate to maximum differential pressure testing which will be used to verify the correct settings. Each valve shall be stroke tested, to the extent practical, to verify that the settings defined in item (2) above have been properly implemented even if testing with differential pressure cannot be performed.
		(4) Prepare or revise procedures to ensure that correct switch settings are determined and maintained throughout the life of the plant. Ensure that applicable industry recommendations are considered in the preparations of these procedures.
5-14	IEIN 86-05 Jan. 31, 1986	This notice discusses incorrect factory-set ring settings (not allowing full disk travel and hence, relief capacity) for main steam safety valves (MSSVs) and for PWR primary system safety valves that may not be known because full-flow tests are not performed or required.

No.	Source	Issue
5-15	NUREG-1195 Feb. 1986 Section 10.1 Item 3	This report discusses the need for a maintenance program for manual isolation valves to ensure continued operability.
5-16	IEIN 86-29 Apr. 25, 1986	This notice discusses the importance of fully understand-standing the effects of changes to MOV switch settings. For instance, the readjustment of a switch that was on same shaft as the torque bypass switch to the closed position in response to IEB 85-03 caused an excessive cooldown rate because the S/D HX isolation valves were shown to be closed when they were really up to 16 percent open. Even though maintenance and operations personnel were aware of this situation, such settings could adversely affect other plant equipment.
5-17	IEIN 86-56 July 10, 1986	This notice lists more causes of MSSV malfunctions as detailed below • majority of problems related to not actuating/reseating at set point • second largest group of problems related to failure to open and failure to close properly • several cases showed the actual lift point to be substantially higher than desired set point • in one case, the lift point of 11 of 16 valves was excessive, indicating a multiple common-mode problem • several cases where the reseat value was substantially below set point • cases indicated that there were more problems with leaking valves than with properly functioning valves
5-18	IEIN 86-92 Nov. 4, 1986	This notice discusses pressurizer code safety valves: • one found 350 psi above set point because of a hole in the bellows causing boron contamination and corrosion • one found leaking too much to test as a result of bad steam cutting • one found lifted prematurely, causing system depressurization to 1800 psig before reseating, as a result of cocked spring and improper adjustment of ring settings by maintenance personnel

No.	Source	Issue
		• one found actuating spuriously as a result of the wrong test equipment being used to set the lift pressure. • 20 other events involving 32 valves that had drift in the set point • large group with seat leakage
5-19	IEIN 86-93 Nov. 3, 1986	This notice discusses the importance of [illegible]ly understanding effects of changes to MOV switch settings. The generic correlation available several years ago that torque varies linearly from 40 to 100 percent with torque switch settings of 1 to 5 is not valid for many actuators. In some cases, at a setting of 1, torque varies from 11 to 55 percent. As a result, an individual calibration curve or bench test is required. Arbitrarily raising the torque switch setting to maximum can cause damage since thermal overloads are often removed. Analysis showed that two valves in the normal charging line that have to close during ECCS actuation would not be able to close against the differential pressure if the generic correlation was used to establish switch set points. In addition, improperly set thermal overloads can render the HPCI systems inoperable.

CATEGORY 6: MAIN FEEDWATER SYSTEM (MFW)

No.	Source	Issue
6-1	NUREG-0667 May 1980 Rec 2.2 (10)	This report suggests that B&W licensees should perform sensitivity studies of possible modifications that could reduce the response of the once-through steam generator (OTSG) to secondary coolant flow perturbations. Both active and passive measures should be investigated to mitigate overcooling and undercooling events.
6-2	GL 81-28 July 31, 1981 (formerly GL 81-16)	This letter transmits the NRC/AEOD report related to steam generator overfill. Licensees are requested to: • determine which scenarios are credible at their plant • determine possible consequences of steam generator overfill • include this information in an overall operator training program Recommendations of the AEOD report:

No.	Source	Issue
		(1) Overfill should be considered an unresolved safety issue (USI) because of the lack of safety-grade equipment to prevent or mitigate overfill and the potential severity of consequences.
		(2) Overfill should be treated as either a separate USI or made a part of an existing USI because consideration of combined blowdown of primary and secondary systems resulting from a steam generator tube rupture (SGTR) is not included in the present USI, which also assumes low probability and allows credit for operator actions.
		(3) An audit should be conducted to determine if reactor operators are aware of the potential seriousness of overfill situations. If the audit shows subjects that are not covered in training programs, interim actions should be initiated.

CATEGORY 7: AUXILIARY/EMERGENCY FEEDWATER SYSTEM (AFW)

No.	Source	Issue
7-1	NUREG-0667 May 1980 Rec. 2.2 (1)	This report discusses the transient response of B&W-designed reactors and contains several recommendations, including those listed below. The AFW system on operating B&W plants should be classified as an ESF system and upgraded as necessary to meet safety-grade requirements.
	Rec. 2.2 (2)	• AFW should be automatically initiated and controlled by ESF (safety-grade) that are independent of non-nuclear instrumentation/integrated control system (NNI/ICS) and other non-safety systems. • The selection of signals used to initiate AFW flow should be re-evaluated to permit automatic initiation in a more timely manner to preclude steam generator dryout. • The level in steam generators should be automatically controlled by the AFW to prevent overcooling and to terminate flow before overfilling.
	Rec. 2.2 (3)	Installation of a diverse-driven AFW pump should be expedited at Davis-Besse.

No.	Source	Issue
	Rec. 2.2 (7)	Provide the flexibility to substitute combinations of in-core thermocouples for loop resistance temperature detectors (devices) (RTDs) used for input to subcooling meter and the capability of continuous or trending display of in-core thermocouples.
	Rec 2.2 (8)	Provide safety-grade containment high-radiation signal to initiate containment vent and purge isolation.
	Rec. 2.2 (21)	The need to introduce AFW through the top sparger during anticipated transients should be re-evaluated by licensees; consider the reduced depressurization response if AFW could be introduced through the main feedwater (MFW) nozzle and could enter the tube region from the bottom of the unit.
7-2	IEIN 80-23 May 23, 1980	This notice describes an event at Arkansas Nuclear One following a loss of offsite power and a reactor trip. Emergency feedwater (EFW) pumps, which started and provided feedwater to the steam generators, list suction as a result of flashing in the main feedwater train which forced hot water through the startup and flowdown demineralizers to the EFW pump suction where it flashed to steam and caused pump cavitation. Action to prevent reoccurrence included revising the EFW system operating procedure and plant startup procedure to require shutting EFW suction valve from startup and flowdown demineralizers during plant startup after the steam generator feedwater source has been shifted to main feedwater pump.
7-3	GL 81-28 July 31, 1981	See item 6-2 above.
7-4	IEB 85-01 Oct. 29, 1985 Item 1	This bulletin relates to steam binding of AFW pumps and requests licensees to develop procedures for monitoring fluid conditions within the AFW system on a regular basis during times when the system is required to be operable. This monitoring should ensure that fluid temperature at the AFW pump discharge is maintained at about ambient temperature. Monitoring of fluid conditions, if used and the primary basis for precluding steam binding, is recommended each shift.
	Item 2	Develop procedures for recognizing steam binding and for restoring the AFW system to operable status, should steam binding occur.
	Item 3	Procedural controls should remain in effect until completion of hardware modification to substantially reduce the likelihood of steam binding or until superseded by action implemented as a result of resolution of Generic Issue 93.

No.	Source	Issue
7-5	IEIN 84-06 Jan. 25, 1984	This notice indicates that leakage into the AFW system from the MFW system constitutes a common mode that can lead to a loss of all AFW capability as a result of steam binding. In addition there is a potential for water hammer damage of AFW pump if relatively cold water discharges into a region of the piping system that contains steam.
	Item 14	It is not clear that overcooling transients such as occurred at Rancho Seco are within the bounds of the FSAR analyses.
7-6	IEIN 86-14 Mar. 10, 1986 Supplement 1 Dec. 17, 1986	This notice discusses overspeed trip/lockout of AFW turbines and the various causes listed below. • immediate clearing and reset of overspeed trip signal caused trip on restart of AFW • leaking steam supply valve caused non-zero initial speed • undrained condensate in long steam supply lines • trip on restart because the governor is designed to start with no initial control of oil pressure (low pressure) (Because the oil pressure does not decay quickly and the provision to dump the oil is local and manual, the turbine will not restart until the oil pressure is reduced.) The supplement to this notice addresses an NRC/AEOD report on turbine overspeed. The dominant causes were problems related to governor speed control, trip valve, and overspeed trip mechanism. The report recommends a procedural change to start up the turbine by warming it with a small steam flow before exposing it to full steam flow. Both procedural inadequacies and human errors were found to contribute to improper setting of governor speed.

CATEGORY 8: ADMINISTRATIVE CONTROLS

No.	Source	Issue
8-1	NUREG-0667 May 1980 Rec 2.2 (15)	This report recommends that mandatory 1-week simulator training should be required for all licensed B&W operators, oriented toward undercooling and overcooling events, solid waste system operation, and natural circulation cooling.
	Item 7	Evaluate procedures and training for reporting events to the NRC Operations Center. Review the adequacy of

No.	Source	Issue
		shift staffing for ensuring that knowledgeable individuals will be available for properly implementing the emergency plan during complex and long operational events.
	Item 13	Verify that plant procedures involving "drastic" actions are sufficiently precise and clear to ensure proper implementation.
8-2	NUREG-1195 Feb. 1986 Section 10.1 Item 4	This report states that the anticipated transient operating guidelines (ATOG) supplied by the BWOG include an explicit procedure for loss of ICS power. However, this procedure may not be included in the utilities' emergency operating procedures (EOPs), as it should be.
	Item 5	EOPs direct operators to trip the appropriate pumps to terminate flow if feedwater flow cannot be isolated; however, operators seem reluctant to do so.
	Item 6	Operator training and procedures should be adequate to resolve conflict between avoiding the pressurized thermal shock region and regaining pressurizer level.
	Item 7	Operators should receive classroom and/or simulator training on overall plant response to either loss of ICS dc power or the restoration of ICS dc power.
	Item 9	Non-licensed operators may only be receiving walk-through or talk-through training where hands-on training may be necessary.
	Item 10	Radiological control and emergency preparedness programs and training may not be adequate if events occur which result in other than minor radiological consequences.
8-3	NUREG-1195 Feb. 1986 Section 10.2 Item 2	This report covers the adequacy of annunciator procedures manual concerning implications of ICS alarm and value to operators in recognizing or restoring a loss of ICS dc power.
	Item 5	Protective clothing or respiratory protection should be readily available in the event of an incident requiring emergency entry into a contaminated area.
	Item 6	Following plant modifications, review procedures, other than those directly affected, to determine applicability.
	Item 7	Determine whether the staffing required by Technical Specifications and other regulatory commitments is adequate to mitigate the effects of an overcooling transient such as occurred at Rancho Seco.

No.	Source	Issue
8-4	Nuclear Safety Analysis Center (NSAC)-3 March 1980 Rec. II.C	This report states that procedures for orderly plant shutdown following loss of power supply should be prepared or reviewed/revised, as necessary. Reactor system cooldown limits, and the basis for those limits should also be reviewed.
	Rec. II.D	The industry should further analyze and resolve with the NRC the current reactor coolant pump trip procedures to be followed during a small-break loss-of-coolant accident (LOCA).
	Rec. II.E	The industry should review the current high-pressure injection pump requirements and resolve any procedural issues with the NRC. Procedures that avoid or minimize challenges to safety valves, primary system, and eventually to the containment building itself are needed.
	Rec. II.F	Procedures for declaring an emergency should be reviewed to determine if responsibility for monitoring plant conditions, which lead to declaring a specific category of emergency, should be assigned to a specific individual.
	Rec. III.B.1	Data handling and display systems should be reviewed to determine their adequacy.
	Rec I.B	Power supply failures and their effects on control systems should be reviewed.
	Rec. I.C	The work practices of instrumentation technicians and their effects on plant safety should be reviewed in plant training sessions.
	Rec. II.A	Written procedures should be established for switching instruments between power supplies in the event of power supply failures, including designating the preferred bus for each instrument.
	Rec. II.B	Procedures for steam generator rupture matrix or its equivalent should be reviewed in conjunction with post-TMI requirements on steam-driven emergency feedwater pumps to determine if aggravating effects exist during loss of heat sink.