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

June 25, 1990

TO: ALL PRESSURIZED WATER REACTOR LICENSEES AND CONSTRUCTION PERMIT HOLDERS

SUBJECT: RESOLUTION OF GENERIC ISSUE 70, "POWER-OPERATED RELIEF VALVE AND BLOCK VALVE RELIABILITY," AND GENERIC ISSUE 94, "ADDITIONAL LOW-TEMPERATURE OVERPRESSURE PROTECTION FOR LIGHT-WATER REACTORS," PURSUANT TO 10 CFR 50.54(f) (GENERIC LETTER 90-06)

The purpose of this generic letter is to advise pressurized water reactor (PWR) licensees and construction permit (CP) holders of the staff positions delineated in Enclosures A and B to this letter. Enclosure A presents the staff position resulting from the resolution of Generic Issue 70 (GI-70) and is applicable to all Westinghouse and Babcock and Wilcox (B&W)-designed plants and Combustion Engineering (CE)-designed plants with power-operated relief valves (PORVs). Enclosure B presents the staff position resulting from the resolution of Generic Issue 94 (GI-94) and is applicable to all Westinghouse-designed and CE-designed plants whether or not they have PORVs and block valves. Enclosure B does not apply to B&W-designed plants.

The technical findings and the regulatory analysis related to GI-70 are discussed in NUREG-1316, "Technical Findings and Regulatory Analysis Related to Generic Issue 70--Evaluation of Power-Operated Relief Valve and Block Valve Reliability in PWR Nuclear Power Plants" (Enclosure C). In Enclosure D, the staff prepared a regulatory analysis for GI-94 based on the work performed by Battelle Pacific Northwest Laboratory (PNL) and reported in NUREG-1326, "Regulatory Analysis for the Resolution of Generic Issue 94, Additional Low-Temperature Overpressure Protection for Light-Water Reactors."

On the basis of technical studies for GI-70, the staff requests that to enhance safety, actions identified in Section 3 of Enclosure A be taken by all PWR licensees and CP holders that use or could use PORVs to perform any of the safety-related functions identified in Section 2 of Enclosure A. These actions result from the staff interpretation of safety-related equipment (see 10 CFR 50.49 and 10 CFR Part 100, Appendix A).

On the basis of technical studies for GI-94, the staff also requests that to enhance safety, actions identified in Section 3 of Enclosure B be taken by all Combustion Engineering and Westinghouse PWR licensees and CP holders. These actions result from the staff interpretation of General Design Criteria 15 and 31 in 10 CFR Part 50, Appendix A. The information requested by this letter is directed at addressing these concerns.

Note that the staff's requests are based on the performance of PORVs and PORV block valve designs used to date on U.S. power reactors. Currently, certain valve manufacturers are developing modified designs with the goal of improving reliability. The use of more reliable valves should result in less frequent corrective maintenance and can result in longer inservice testing intervals as delineated in Section XI of the ASME Boiler and Pressure Vessel Code.

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Accordingly, pursuant to Section 182 of the Atomic Energy Act and 10 CFR 50.54(f), you, as a PWR licensee or CP holder, are required to advise the NRC staff under oath or affirmation, within 180 days of the date of this letter, of your current plans relating to PORVs and block valves and to low-temperature overpressure protection, in particular whether you intend to follow the staff positions included in Enclosures A and B as applicable, or propose alternative measures, and your proposed schedule for implementation.

For PWR plants with an operating license, staff positions 1 and 2 in Section 3.1 of Enclosure A should be implemented by the end of the first refueling outage that starts 6 months or later from the date of this letter. Requests for the technical specification modifications in staff position 3 in Section 3.1 of Enclosure A and in Section 3 of Enclosure B should be submitted by the end of the first refueling outage that starts 6 months or later from the date of this letter.

For PWR CP holders, staff positions 1 and 2 in Section 3.1 of Enclosure A should be implemented before initial criticality or within 6 months of the date of this letter, whichever is later. The technical specification modifications in staff position 3 in Section 3.1 of Enclosure A and in Section 3 of Enclosure B should be submitted by the end of the first refueling outage that starts 6 months or later from the date of this letter.

If the applicable schedule cannot be met, the licensee or the CP holder shall advise the staff of a proposed revised schedule, justification for any delay, and any planned compensating measures to be taken during the interim. Alternatives to schedules and the guidance provided herein will be evaluated on their merits on an individual case basis. Based on its review and the acceptability of these responses, the staff will close out GI-70 and GI-94 for each plant.

Your response shall include the following specific items.

1. A statement by licensees and CP holders as to whether they will commit to incorporate improvements 1, 2, and 3 in Section 3.1 of Enclosure A. With respect to improvement 3 in Section 3.1 of Enclosure A, licensees and CP holders shall state whether they will commit to use those modified limiting conditions of operation of PORVs and block valves in the technical specifications for Modes 1, 2, and 3 in Attachment A-1 of Enclosure A for Westinghouse-designed and CE-designed plants with two PORVs, or in Attachment A-2 of Enclosure A for Westinghouse-designed plants with three PORVs, or in Attachment A-4 of Enclosure A for B&W-designed plants.[1] In addition to this 10 CFR 50.54(f) request, if the licensees and the CP holders commit to implement these recommended technical specifications, it is requested that they submit modifications to their current technical specifications in a license amendment in accordance with the schedule noted above.

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[1] Plants that already have staff-issued technical specifications consistent with these requirements need merely state this in their response. No further action will be required for this aspect of the Commission's position.

2. A statement by licensees and CP holders as to whether they will submit a license amendment request to modify the technical specifications and commit to use the modified technical specifications for the low-temperature overpressure protection system concerning the limiting conditions of operation in Modes 5 and 6 as identified in Attachment B-1 of Enclosure B to this generic letter for Westinghouse-designed or CE-designed plants, as appropriate. In addition to this 10 CFR 50.54(f) request, if the licensees and CP holders commit to implement these recommended technical specifications, it is requested that they submit modifications to their current technical specifications in a license amendment in accordance with the schedule noted above.

The actions to incorporate technical specification (TS) requirements for the resolution of GI-70 and GI-94 are considered to be consistent with the Commission's Policy Statement on Technical Specification Improvements. This policy statement captures existing requirements under Criterion 3 (Mitigation of Design-Basis Accidents or Transients) or under the provisions to retain requirements that operating experience and probabilistic risk assessment show to be important to the public health and safety. Although it is recognized that PORVs for older plants may not have been classified as safety-related components that are used to mitigate a design-basis accident and, therefore, may not have been included in TS as part of the plant's licensing basis, this is not an acceptable basis for not implementing the proposed actions to incorporate TS requirements for PORVs consistent with the guidance provided. Likewise, such requirements would be retained in TS when implementing improvements in TS consistent with the Commission policy statement on the basis of Criterion 3 or risk considerations noted above.

Backfit Discussion

For GI-70, the actions proposed by the NRC staff to improve the reliability of PORVs and block valves, as identified in Section 3 of Enclosure A, represent new staff positions for some licensees and CP holders, and this request is considered a backfit in accordance with NRC procedures. This backfit is a cost-justified safety enhancement. Therefore, an analysis of the type described in 10 CFR 50.109(a)(3) and 50.109(c) was performed, and a determination was made that there will be a substantial increase in overall protection of the public health and safety and that the attendant costs are justified in view of this increased protection. The analysis and determination will be made available in the Public Document Room with the minutes of the 167th and 168th meetings of the Committee to Review Generic Requirements.

It is noted that most of the recommended actions for GI-70 may already be implemented by those plants that have received operating licenses in recent years and would, therefore, represent less of a backfit than for older PWR plants that currently do not include PORVs and block valves in the ASME Section XI Inservice Testing Program and do not have technical specifications for PORVs and block valves or that operate with the block valves closed because of leaking PORVs.

For GI-94, the actions proposed by the NRC staff to improve the availability of the low-temperature overpressure protection (LTOP) system, as identified in Section 3 of Enclosure B, represent a new interpretation of existing requirements for some licensees and CP holders, and this request is considered a backfit in accordance with NRC procedures. This backfit is a cost-justified safety enhancement. Therefore, an analysis of the type described in 10 CFR 50.109(a)(3) and 50.109(c) was performed, and a determination was made that there will be a substantial increase in overall protection of the public health and safety and that the attendant costs are justified in view of this increased protection. The analysis and determination will be made available in the Public Document Room with the minutes of the 167th and 168th meetings of the Committee to Review Generic Requirements.

This request is covered by Office of Management and Budget Clearance Number 3150-0011, which expires January 31, 1991. The estimated average burden hours is 320 person-hours per licensee response, including assessment of the new recommendations, searching data sources, gathering and analyzing the data, and preparing the required reports. Comments on the accuracy of this estimate and suggestions to reduce the burden may be directed to the Office of Management and Budget, Room 3208, New Executive Office Building, Washington, D.C. 20503, and the U.S. Nuclear Regulatory Commission, Information and Records Management Branch, Office of Information Resources Management, Washington, D.C. 20555.

Sincerely,

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Enclosures:
A. Staff Positions Resulting from Resolution of Generic Issue 70
B. Staff Positions Resulting from Resolution of Generic Issue 94
C. NUREG-1316, "Technical Findings and Regulatory Analysis Related to Generic Issue 70--Evaluation of Power-Operated Relief Valve and Block Valve Reliability in PWR Nuclear Power Plants"
D. NUREG-1326, "Regulatory Analysis for the Resolution of Generic Issue 94, Additional Low-Temperature Overpressure Protection for Light-Water Reactors"

Enclosure A to Generic Letter 90-06

Staff Positions Resulting from
Resolution of Generic Issue 70 -
PORV and Block Valve Reliability

1. BACKGROUND

Generic Issue 70 (GI-70), "Power-Operated Relief Valve and Block Valve Reliability," involves the evaluation of the reliability of power-operated relief valves (PORVs) and block valves and their safety significance in PWR plants. The technical findings and regulatory analysis related to GI-70 are discussed in NUREG-1316, "Technical Findings and Regulatory Analysis Related to Generic Issue 70--Evaluation of Power-Operated Relief Valve and Block Valve Reliability in PWR Nuclear Power Plants" (Enclosure C). This report identifies those safety-related functions that may be performed by PORVs and also identifies potential improvements to PORVs and block valves. In support of the resolution of GI-70, the Oak Ridge National Laboratory (ORNL) performed a study of PORV and block valve operating experience. A report, prepared by ORNL, was issued as NUREG/CR-4692, "Operating Experience Review of Failures of Power Operated Relief Valves and Block Valves in Nuclear Power Plants," dated October 1987.

Traditionally, the PORV and its block valve are provided for plant operational flexibility and for limiting the number of challenges to the safety-related pressurizer safety valves. The operation of the PORVs has not been classified as a safety-related function, i.e., one on which the results and conclusions of the safety analysis are based and that invokes the highest level of quality and construction. For overpressure protection of the reactor coolant pressure boundary (RCPB) at normal operating temperature and pressure, the operation of PORVs has not been explicitly considered as a safety-related function. Also, an inadvertent opening of a PORV or safety valve has been analyzed in the Final Safety Analysis Reports as an anticipated operational occurrence with acceptable consequences. For these reasons, most PWRs, particularly those licensed prior to 1979, do not classify PORVs as safety-related components.

The Three Mile Island Unit 2 (TMI-2) accident focused attention on the reliability of PORVs and block valves since the malfunction of the PORV at TMI-2 contributed to the severity of the accident. On other occasions, PORVs have stuck open when called upon to function. Also, there are PORVs in many operating plants that have leakage problems so that the plants must be operated with the upstream block valves in the closed position. The technical specifications governing PORVs on most operating PWRs, which deal with closing the block valve and removing power, were developed to allow continued plant operation with degraded PORVs, but did not consider the need for the PORVs to perform the safety functions discussed below.

Following the TMI-2 accident, the staff began to examine transient and accident events in more detail, particularly with respect to required operator actions and equipment availability and performance. As a result, the staff initiated an evaluation of the role of PORVs to perform certain safety-related functions.

2. SAFETY FUNCTIONS OF PORVs AND BLOCK VALVES

The staff, in its evaluation, determined that over a period of time the role of PORVs has changed such that PORVs are now relied upon by many Westinghouse, B&W, and CE designed plants with PORVs to perform one, or more, of the following safety-related functions:

1. Mitigation of a design-basis steam generator tube rupture accident,
2. Low-temperature overpressure protection of the reactor vessel during startup and shutdown, or
3. Plant cooldown in compliance with Branch Technical Position RSB 5-1 to SRP 5.4.7, "Residual Heat Removal (RHR) System."

Where PORVs are used or could be used to perform one, or more, of the safety-related functions identified above or to perform any other safety-related function that may be identified in the future, it is appropriate to reconsider the safety classification of PORVs and the associated block valves. For certain PWR plants receiving an operating license in recent years, the staff has required these valves to be classified as safety-related components if they perform one, or more, safety-related functions.

For operating PWR plants, the staff has concluded that it is not cost effective to replace (backfit) existing non-safety-grade PORVs and block valves (and associated control systems) with PORVs and block valves that are safety grade even when they have been determined to perform any of the safety-related functions discussed above. Subsequent to the TMI-2 accident, a number of improvements were required of PORVs and block valves, such as requirements to be powered from Class IE buses and to have valve position indication in the control room. For operating plants, the greatest immediate benefits can be derived from implementing items 1 through 3 identified below, which can increase the reliability of these components and provide assurance they will function as required.

3. IMPROVEMENTS TO ALL PORVs AND BLOCK VALVES

3.1 Operating PWR Plants and Construction Permit Holders

Based on the analysis and findings for GI-70, the staff concludes that the following actions should be taken to improve the reliability of PORVs and block valves:

1. Include PORVs and block valves within the scope of an operational quality assurance program that is in compliance with 10 CFR Part 50, Appendix B. This program should include the following elements:

   a. The addition of PORVs and block valves to the plant operational Quality Assurance List.

   b. Implementation of a maintenance/refurbishment program for PORVs and block valves that is based on the manufacturer's recommendations or guidelines and is implemented by trained plant maintenance personnel.

c. When replacement parts and spares, as well as complete components, are required for existing non-safety-grade PORVs and block valves (and associated control systems), it is the intent of this generic letter that these items may be procured in accordance with the original construction codes and standards.

2. Include PORVs, valves in PORV control air systems, and block valves within the scope of a program covered by Subsection IWV, "Inservice Testing of Valves in Nuclear Power Plants," of Section XI of the ASME Boiler and Pressure Vessel Code. Stroke testing of PORVs should only be performed during Mode 3 (HOT STANDBY) or Mode 4 (HOT SHUTDOWN) and in all cases prior to establishing conditions where the PORVs are used for low-temperature overpressure protection. Stroke testing of the PORVs should not be performed during power operation. Additionally, the PORV block valves should be included in the licensees' expanded MOV test program discussed in NRC Generic Letter 89-10, "Safety-Related Motor Operated Valve Testing and Surveillance," dated June 28, 1989.

3. For operating PWR plants, modify the limiting conditions of operation of PORVs and block valves in the technical specifications for Modes 1, 2, and 3 to incorporate the position adopted by the staff in recent licensing actions. Attachments A-1 through A-3 are provided for guidance. The staff recognizes that some recently licensed PWR plants already have technical specifications in accordance with the staff position. Such plants are already in compliance with this position and need merely state that in their response. These recent technical specifications require that plants that run with the block valves closed (e.g., due to leaking PORVs) maintain electrical power to the block valves so they can be readily opened from the control room upon demand. Additionally, plant operation in Modes 1, 2, and 3 with PORVs and block valves inoperable for reasons other than seat leakage is not permitted for periods of more than 72 hours.

Enclosure A to Generic Letter 90-06

Attachment A-1

Modified Standard Technical Specifications
for Combustion Engineering and Westinghouse Plants

REACTOR COOLANT SYSTEM

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

The following is to be used when two PORVs are provided:

3.4.4 Both power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or both PORVs inoperable because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one PORV inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV to OPERABLE status or close its associated block valve and remove power from the block valve; restore the PORV to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With both PORVs inoperable due to causes other than excessive seat leakage, within 1 hour either restore at least one PORV to OPERABLE status or close its associated block valve and remove power from the block valve and be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

d. With one or both block valves inoperable, within 1 hour restore the block valve(s) to OPERABLE status or place its associated PORV(s) in manual control. Restore at least one block valve to OPERABLE status within the next hour if both block valves are inoperable; restore any remaining inoperable block valve to operable status within 72 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

e. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at least once per 18 months by:

a. Operating the PORV through one complete cycle of full travel during MODES 3 or 4, and

Enclosure A To Generic Letter 90-06

Attachment A-2

Modified Standard Technical Specifications
for Westinghouse Plants with Three PORVs

REACTOR COOLANT SYSTEM

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

The following is to be used when three PORVs are provided:

3.4.4 All power-operated relief valves (PORVs) and their associated block valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or more PORVs inoperable because of excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) with power maintained to the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and HOT SHUTDOWN within the following 6 hours.

b. With one or two PORVs inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV(s) to OPERABLE status or close the associated block valve(s) and remove power from the block valve(s); restore the PORV(s) to OPERABLE status within the following 72 hours or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With three PORVs inoperable due to causes other than excessive seat leakage, within 1 hour either restore at least one PORV to OPERABLE status or close the block valves and remove power from the block valve(s) and be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

d. With one or more block valves inoperable, within 1 hour restore the block valve(s) to OPERABLE status or place its associated PORV in manual control. Restore at least one block valve to OPERABLE status within the next hour if three block valves are inoperable; restore any remaining inoperable block valve(s) to operable status within 72 hours; otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

e. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at least once per 18 months by:

a. Operating the PORV through one complete cycle of full travel during MODES 3 or 4, and

b. Where applicable, operating solenoid air control valves and check valves on associated air accumulators in PORV control systems through one complete cycle of full travel for plants with air-operated PORVs, and

c. Performing a CHANNEL CALIBRATION of the actuation instrumentation.

4.4.4.2 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed in order to meet the requirements of ACTION b, or c in Specification 3.4.4.

4.4.4.3 The emergency power supply for the PORVs and block valves shall be demonstrated OPERABLE at least once per 18 months by:

a. Manually transferring motive and control power from the normal to the emergency power bus, and

b. Operating the valves through a complete cycle of full travel.

Enclosure A to Generic Letter 90-06

Attachment A-3

Applicable to Combustion Engineering and Westinghouse Plants

3/4.4.4 RELIEF VALVES

Bases of the Limiting Condition for Operation (LCO) and Surveillance Requirements:

The OPERABILITY of the PORVs and block valves is determined on the basis of their being capable of performing the following functions:

A. Manual control of PORVs to control reactor coolant system pressure. This is a function that is used for the steam generator tube rupture accident and for plant shutdown. This function has been classified as safety related for more recent plant designs.

B. Maintaining the integrity of the reactor coolant pressure boundary. This is a function that is related to controlling identified leakage and ensuring the ability to detect unidentified reactor coolant pressure boundary leakage.

C. Manual control of the block valve to: (1) unblock an isolated PORV to allow it to be used for manual control of reactor coolant system pressure (Item A), and (2) isolate a PORV with excessive seat leakage (Item B).

D. Automatic control of PORVs to control reactor coolant system pressure. This is a function that reduces challenges to the code safety valves for overpressurization events.

E. Manual control of a block valve to isolate a stuck-open PORV.

Surveillance Requirements provide the assurance that the PORVs and block valves can perform their functions. Specification 4.4.4.1 addresses PORVs, 4.4.4.2 the block valves, and 4.4.4.3 the emergency (backup) power sources. The latter are provided for either PORVs or block valves, generally as a consequence of the TMI ACTION requirements to upgrade the operability of PORVs and block valves, where they are installed with non-safety-grade power sources, including instrument air, and are provided with a backup (emergency) power source. The block valves are exempt from the surveillance requirements to cycle the valves when they have been closed to comply with the ACTION requirements. This precludes the need to cycle the valves with full system differential pressure or when maintenance is being performed to restore an inoperable PORV to operable status.

Surveillance requirement 4.4.4.1.b has been added to include testing of the mechanical and electrical aspects of control systems for air-operated PORVs.

Testing of PORVs in HOT STANDBY or HOT SHUTDOWN is required in order to simulate the temperature and pressure environmental effects on PORVs. In many PORV designs, testing at COLD SHUTDOWN is not considered to be a representative test for assessing PORV performance under normal plant operating conditions.

The Modified Standard Technical Specification (STS) requirements include the following changes from prior STS guidance:

1. Clarify the statement of LCO by replacing "All" with "Both" where the design includes two PORVs.

2. ACTION statement a. includes the requirement to maintain power to closed block valve(s) because removal of power would render the block valve(s) inoperable and the requirements of ACTION statement c. would apply. Power is maintained to the block valve(s) so that it is operable and may be subsequently opened to allow the PORV to be used to control reactor pressure. Closure of the block valve(s) establishes reactor coolant pressure boundary integrity for a PORV that has excessive seat leakage. (Reactor coolant pressure boundary integrity takes priority over the capability of the PORV to mitigate an overpressure event.) However, the APPLICABILITY requirements of the LCO to operate with the block valve(s) closed with power maintained to the block valve(s) are only intended to permit operation of the plant for a limited period of time not to exceed the next refueling outage (MODE 6) so that maintenance can be performed on the PORVs to eliminate the seat leakage condition. The PORVs should normally be available for automatic mitigation of overpressure events and should be returned to OPERABLE status prior to entering STARTUP (MODE 2).

3. ACTION statements b. and c. include the removal of power from a closed block valve as additional assurance to preclude any inadvertent opening of the block valve at a time in which the PORV may not be closed due to maintenance to restore it to OPERABLE status. (In contrast, ACTION statement a. is intended to permit continued plant operation for a limited period of time with the block valves closed, i.e., continued operation is not dependent on maintenance at power to eliminate excessive PORV leakage, and, therefore, ACTION statement a. does not require removal of power from the block valve.)

4. ACTION statements a., b., and c. have been changed to terminate the forced shutdown requirements with the plant being in HOT SHUTDOWN rather than COLD SHUTDOWN because the APPLICABILITY requirements of the LCO do not extend past the HOT STANDBY mode.

5. ACTION statement d. has been modified to establish remedial measures that are consistent with the function of the block valves. The prime importance for the capability to close the block valve is to isolate a stuck-open PORV. Therefore, if the block valve(s) cannot be restored to operable status within 1 hour, the remedial action is to place the PORV in manual control to preclude its automatic opening for an overpressure event and to avoid the potential for a stuck-open PORV at a time that the block valve is inoperable. The time allowed to restore the block valve(s) to operable status is based upon the remedial action time limits for inoperable PORVs per ACTION statements b. and c. since the PORVs

are not capable of mitigating an overpressure event when placed in manual control. These actions are also consistent with the use of the PORVs to control reactor coolant system pressure if the block valves are inoperable at a time when they have been closed to isolate PORVs that have excessive seat leakage. The modified ACTION statement does not specify closure of the block valves because such action would not likely be possible when the block valve is inoperable. Likewise, it does not specify either the closure of the PORV, because it would not likely be open, or the removal of power from the PORV. When the block valve is inoperable, placing the PORV in manual control is sufficient to preclude the potential for having a stuck-open PORV that could not be isolated because of an inoperable block valve. For the same reasons, reference is not made to ACTION statements b. and c. for the required remedial actions.

6. Surveillance requirement 4.4.4.2 has been modified to remove the exception for testing the block valves when they are closed to isolate an inoperable PORV. If the block valve is closed to isolate a PORV with excessive seat leakage, the operability of the block valve is of importance, because opening the block valve is necessary to permit the PORV to be used for manual control of reactor pressure. If the block valve is closed to isolate an otherwise inoperable PORV, the maximum allowable outage time is 72 hours, which is well within the allowable limits (25 percent) to extend the block valve surveillance interval (92 days). Furthermore, these test requirements would be completed by the reopening of a recently closed block valve upon restoration of the PORV to operable status, i.e., completion of the ACTION statement fulfills the required surveillance requirement.

Enclosure A to Generic Letter 90-06

Attachment A-4

Modified Technical Specifications
for Babcock and Wilcox Plant

REACTOR COOLANT SYSTEM

3/4.4.4 RELIEF VALVE

LIMITING CONDITION FOR OPERATION

3.4.4 The power-operated relief valve (PORV) and its associated block valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the PORV inoperable because of excessive seat leakage, within 1 hour either restore the PORV to OPERABLE status or close the associated block valve with power maintained to the block valve; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With the PORV inoperable due to causes other than excessive seat leakage, within 1 hour either restore the PORV to OPERABLE status or close the associated block valve and remove power from the block valve, and be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With the block valve inoperable, within 1 hour restore the block valves to OPERABLE status or place the associated PORV in manual control and restore the block valve to operable status within the next hour; otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of Specification 4.0.5, the PORV shall be demonstrated OPERABLE at least once per 18 months by:

a. Operating the PORV through one complete cycle of full travel during MODES 3 or 4, and

b. Performing a CHANNEL CALIBRATION of the actuation instrumentation.

4.4.4.2 The block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed in order to meet the requirements of ACTION b in Specification 3.4.4.

4.4.4.3 The emergency power supply for the PORV and block valve shall be demonstrated OPERABLE at least once per 18 months by:

a. Manually transferring motive and control power from the normal to the emergency power bus, and

b. Operating the valve through a complete cycle of full travel.

BABCOCK & WILCOX PLANTS

Enclosure A to Generic Letter 90-06

Attachment A-5

Applicable to Babcock and Wilcox Plants

3/4.4.4 RELIEF VALVE

Bases of the Limiting Condition for Operation (LCO) and Surveillance Requirements:

The OPERABILITY of the PORV and block valve is determined on the basis of their being capable of performing the following functions:

A. Manual control of the PORV to control reactor coolant system pressure. This is a function that is used for the steam generator tube rupture accident and for plant shutdown. This function has been classified as safety related for more recent plant designs.

B. Maintaining the integrity of the reactor coolant pressure boundary. This is a function that is related to controlling identified leakage and ensuring the ability to detect unidentified reactor coolant pressure boundary leakage.

C. Manual control of the block valve to: (1) unblock an isolated PORV to allow it to be used for manual control of reactor coolant system pressure (Item A), and (2) isolate the PORV with excessive seat leakage (Item B).

D. Automatic control of the PORV to control reactor coolant system pressure. This is a function that reduces challenges to the code safety valves for overpressurization events.

E. Manual control of a block valve to isolate a stuck-open PORV.

Surveillance Requirements provide the assurance that the PORV and block valve can perform their functions. Specification 4.4.4.1 addresses the PORV, 4.4.4.2 the block valve, and 4.4.4.3 the emergency (backup) power source. The latter is provided for either the PORV or block valve, generally as a consequence of the TMI ACTION requirements to upgrade the operability of PORVs and block valves, where they are installed with non-safety-grade power sources, including instrument air, and are provided with backup (emergency) power sources. The block valve is exempt from the surveillance requirements to cycle the valve when it has been closed to comply with the ACTION requirements. This precludes the need to cycle the valve with full system differential pressure or when maintenance is being performed to restore an inoperable PORV to operable status.

Testing the PORV in HOT STANDBY or HOT SHUTDOWN is required in order to simulate the temperature and pressure environmental effects on the PORV. In many PORV designs, testing at COLD SHUTDOWN is not considered to be a representative test for assessing PORV performance under normal plant operating conditions.

The Modified Standard Technical Specification (STS) requirements include the following changes from prior STS guidance:

1. ACTION statement a. includes the requirement to maintain power to the closed block valve, because removal of power would render the block valve inoperable and the requirements of ACTION statement c. would apply. Power is maintained to the block valve so that it is operable and may be subsequently opened to allow the PORV to be used to control reactor pressure. Closure of the block valve establishes reactor coolant pressure boundary integrity for a PORV that has excessive seat leakage. (Reactor coolant pressure boundary integrity takes priority over the capability of the PORV to mitigate an overpressure event.) However, the APPLICABILITY requirement of the LCO to operate with the block valve closed with power maintained to the block valve is only intended to permit operation of the plant for a limited period of time not to exceed the next refueling outage (MODE 6) so that maintenance can be performed on the PORV to eliminate the seat leakage condition. The PORV should normally be available for automatic mitigation of overpressure events and should be returned to OPERABLE status prior to entering STARTUP (MODE 2).

2. ACTION statement b. includes the removal of power from the closed block valve as additional assurance to preclude any inadvertent opening of the block valve at a time in which the PORV may not be closed due to maintenance to restore it to OPERABLE status. (In contrast, ACTION statement a. is intended to permit continued plant operation for a limited period of time with the block valve closed, i.e., continued operation is not dependent on maintenance at power to eliminate excessive PORV leakage, and, therefore, ACTION statement a. does not require removal of power from the block valve.)

3. ACTION statements a. and b. have been changed to terminate the forced shutdown requirements with the plant being in HOT SHUTDOWN rather than COLD SHUTDOWN because the APPLICABILITY requirements of the LCO do not extend past the HOT STANDBY mode.

4. ACTION statement c. has been modified to establish remedial measures that are consistent with the function of the block valves. The prime importance for the capability to close the block valve is to isolate a stuck-open PORV. Therefore, if the block valve cannot be restored to operable status within 1 hour, the remedial action is to place the PORV in manual control to preclude its opening for an overpressure event and to avoid the potential for a stuck-open PORV at a time that the block valve is inoperable. The time allowed to restore the block valve to operable status is based upon the remedial action time limits for an inoperable PORV per ACTION statement b. since the PORV is not capable of mitigating an overpressure event when placed in manual control. This action is also consistent with the use of the PORV to control reactor coolant system pressure if the block valve is inoperable at a time when it was

closed to isolate a PORV that has excessive seat leakage. The modified ACTION statement does not specify closure of the block valve because such action would not likely be possible when the block valve is inoperable. Likewise, it does not specify either the closure of the PORV, because it would not likely be open, or the removal of power from the PORV. When the block valve is inoperable, placing the PORV in manual control is sufficient to preclude the potential for having a stuck-open PORV that could not be isolated because of an inoperable block valve. For the same reasons, reference is not made to ACTION statement b. for the required remedial action.

5. Surveillance requirement 4.4.4.2 has been modified to remove the exception for testing the block valve when it is closed to isolate an inoperable PORV. If the block valve is closed to isolate a PORV with excessive seat leakage, the operability of the block valve is of importance, because opening the block valve is necessary to permit the PORV to be used for manual control of reactor pressure. If the block valve is closed to isolate an otherwise inoperable PORV, the maximum allowable outage time is 72 hours, which is well within the allowable limits (25 percent) to extend the block valve surveillance interval (92 days). Furthermore, these test requirements would be completed by the reopening of a recently closed block valve upon restoration of the PORV to operable status, i.e., completion of the ACTION statement fulfills the required surveillance requirement.

Enclosure B to Generic Letter 90-06

Staff Positions Resulting from
Resolution of Generic Issue 94 -
Additional Low-Temperature Overpressure Protection
For Light-Water Reactors

1. BACKGROUND

Generic Issue 94 (GI-94), "Additional Low-Temperature Overpressure Protection for Light-Water Reactors," addresses concerns with the implementation of the requirements set forth in the resolution of Unresolved Safety Issue (USI) A-26, "Reactor Vessel Pressure Transient Protection (Overpressure Protection)." In support of GI-94, the Battelle Pacific Northwest Laboratories (PNL) performed a study based on actual operating reactor experiences to determine the risks associated with current low-temperature overpressure protection (LTOP) systems. A report, prepared by PNL, has been issued as NUREG/CR-5186, "Value/Impact Analysis of Generic Issue 94, Additional Low Temperature Overpressure Protection for Light-Water Reactors," dated November 1988. The staff has prepared a regulatory analysis for GI-94 based on the work performed by PNL and reported in NUREG-1326, "Regulatory Analysis for the Resolution of Generic Issue 94, Additional Low-Temperature Overpressure Protection for Light-Water Reactors" (Enclosure D).

Low-temperature overpressure protection (LTOP) was designated as Unresolved Safety Issue A-26 in 1978 (NUREG-0371). PWR licensees implemented procedures to reduce the potential for overpressure events and installed equipment modifications to mitigate such events based on the staff recommendations from the USI A-26 evaluations, under Multi-Plant Action Item B-04 (NUREG-0748). Current staff guidelines for LTOP are in Standard Review Plan Section 5.2.2, "Overpressure Protection," and in its attached Branch Technical Position (BTP) RSB 5-2, "Overpressure Protection of Pressurized Water Reactors While Operating at Low Temperatures" (NUREG-0800).

The administrative controls and procedures that were identified as part of Multi-Plant Action Item B-04 include the following items:

1. Minimize the time the reactor coolant system (RCS) is maintained in a water-solid condition.

2. Restrict the number of high-pressure safety injection pumps operable to no more than one when the RCS is in the LTOP condition.

3. Ensure that the steam generator to RCS temperature difference is less than 50 Deg F when a reactor coolant pump (RCP) is being started in a water-solid RCS.

4. Set the PORV setpoint (if the particular plant relies on this component for LTOP) to a plant-specific analysis supported value, and have surveillance that checks the PORV actuation electronics and setpoint.

Twelve PWR overpressure transients were reported during the period from 1981 to 1983 after completion of USI A-26. Two of these events, at Turkey Point Unit 4, exceeded the pressure/temperature limits of the technical specifications. During this same timeframe, there were 37 reported instances when at least one LTOP channel was out of service. In 12 of these cases, both LTOP channels were inoperable.

The continuation of overpressure transient events, and the unavailability of LTOP protection channels, suggested the need to reevaluate the current overpressure protection requirements, or their implementation, to determine whether additional measures are warranted.

Major overpressurization of the reactor coolant system while at low temperature, if combined with a critical crack in the reactor pressure vessel welds or plate material, could result in a brittle fracture of the pressure vessel. Failure of the pressure vessel could make it impossible to provide adequate coolant to the reactor core and result in major core damage or a core melt accident.

The safety significance of these continuing low-temperature overpressure transients was designated as Generic Issue 94, "Additional Low Temperature Overpressure Protection." The concerns of GI-94 are applicable to all PWR plants regardless of the features used to mitigate a low-temperature overpressure event or of any measures to preclude events that would challenge these features or exceed the design basis for LTOP.

The implementation of the requirement for an LTOP system (the resolution of USI A-26) has been found to be essentially uniform for the Combustion Engineering (CE) and Westinghouse (W) PWRs. With the exception of a few plants,* the LTOP protection systems consist of either redundant PORVs or redundant safety relief valves (SRVs) in the residual heat removal (RHR) system and in general meet the guidance set forth in Branch Technical Position RSB 5-2, "Overpressurization Protection of Pressurized Water Reactors While Operating at Low Temperatures."

Variability in meeting IEEE-279 requirements, equipment environmental qualification, and in meeting the guidance of Regulatory Guide 1.26, "Quality Group Classification and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," exists. As part of the NRC staff acceptance of LTOP protection system designs for the implementation of the resolution of USI A-26, it was concluded that the costs associated with upgrading existing systems to meet the guidance of Regulatory Guide 1.26 were not

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* CE - San Onofre Units 2 and 3 rely on a single RHR (SDCS) SRV for LTOP. With the SRV inoperable, depressurize and vent within 8 hours.
- Maine Yankee relies on two PORVs when pressure is above 400 psig and two RHR SRVs when pressure is below 400 psig.

W - DC Cook Units 1 and 2 rely on either two PORVs or one PORV and one RHR SRV.
- Yankee Rowe relies on one PORV and two RHR SRVs.
- Newer Westinghouse plants allow either two PORVs or two RHR SRVs.

justifiable. Further evaluations performed for GI-94 have also concluded that it is not cost beneficial to upgrade these systems to fully safety-grade standards.

2. CURRENT STANDARD TECHNICAL SPECIFICATION REQUIREMENTS

The section of the Standard Technical Specifications (STS) covering the LTOP protection system is entitled Overpressure Protection System, Section 3.4.10.3 for CE plants and Section 3.4.9.3 for W plants. The LTOP system setpoints are established based on additional restrictions for the restart of an idle reactor coolant pump and on the number of high-pressure safety injection pumps and/or coolant charging pumps allowed to be operable when LTOP is required. These additional restrictions define the initial conditions for the plant-specific transient analyses performed to establish the LTOP system setpoints. The additional restrictions are provided regarding the restart of inactive reactor coolant pumps in Sections 3.4.1.3 (Hot Shutdown) and 3.4.1.4 (Cold Shutdown). High-pressure safety injection pump operability restrictions are provided in Section 3/4.5.3 (ECCS Subsystems). In addition to these administrative restrictions, the transient analyses are based on a dual-channel system being operable to satisfy the single failure criterion of 10 CFR Part 50, Appendix A, for a system that performs a safety function. Therefore, the Overpressure Protection System TS is consistent with Criterion 2 of the Commission's Policy Statement on Technical Specification Improvements for Nuclear Power Plants. The TS also satisfied Criterion 3 of the policy statement in that the LTOP system is the primary success path for the mitigation of low-temperature overpressure transients that present a challenge to a fission product barrier, in this case, the reactor pressure vessel.

PORVs are relied on, by most Westinghouse designed plants and about one-half of the Combustion Engineering plants, to provide LTOP protection. In addition to PORVs, the RHR SRVs are also relied on to provide LTOP protection for some W plants and for the CE plants that do not have PORVs. Newer W plants have TS that require either two PORVs or two RHR SRVs for LTOP protection.

The current STS ACTION requirements for the LTOP system include a 7-day allowable outage time (AOT) to restore an inoperable LTOP channel to operable status before other remedial measures would have to be taken. In addition, ACTION d. states that the provisions of Specification 3.0.4 are not applicable. Therefore, the plant may enter the modes for which the Limiting Conditions for Operation (LCO) apply, during a plant shutdown or placement of the head on the vessel following refueling, when an LTOP channel is inoperable. In this situation, the 7-day AOT applies for restoring the channel to operable status before other remedial measures would have to be taken. This is the same manner in which the ACTION requirements apply when an LTOP channel is determined to be inoperable while the plant is in a mode for which the LTOP system is required to be operable.

Based on the NRC evaluation of the LTOP system unavailability, it is concluded that additional restrictions on operation with an inoperable LTOP channel are warranted when the potential for a low-temperature overpressure event is the highest, and especially when the plant is in a water-solid condition. Furthermore, it is concluded that the additional restrictions regarding the restart of inactive reactor coolant pumps and regarding the operability of high-pressure safety injection pumps should be implemented in the TS, as indicated in the STS, and licensees should verify that these administrative restrictions have been implemented. Finally, it is concluded that these additional measures will help to emphasize the importance of the LTOP system, especially while operating in a water-solid condition, as the primary success path for the mitigation of overpressure transients during low-temperature operation.

3. IMPROVEMENTS IN PROTECTION SYSTEM AVAILABILITY

The staff has determined that LTOP protection system unavailability is the dominant contributor to risk from low-temperature overpressure transients. The staff has further concluded that a substantial improvement in availability when the potential for an overpressure event is the highest, and especially during water-solid operations, can be achieved through improved administrative restrictions on the LTOP system.

In developing the staff position on the resolution of the low-temperature overpressure protection generic issue, a number of factors have been taken into consideration.

The staff has considered the conditions under which a low-temperature overpressure transient is most likely to occur. While LTOP protection is required for all shutdown modes, the most vulnerable period of time was found to be MODE 5 (Cold Shutdown) with the reactor coolant temperature less than or equal to 200 Deg F, especially when water-solid, based on the detailed evaluation of operating reactor experiences performed in support of GI-94. LTOP transients that have challenged the overpressure protection system have occurred with reactor coolant temperatures in the range of 80 Deg F to 190 Deg F. In addition, a review of the STS for containment integrity indicates that there are no specific requirements imposed during MODE 5, when the reactor coolant temperature is below 200 Deg F. Industry responses to Generic Letter 87-12, "Loss of RHR While RCS Partially Filled," dated July 9, 1987, also indicate that containment integrity during MODE 5 is often relaxed to allow for testing, maintenance, and the repair of equipment.

In addition, the staff takes note of the fact that, in all instances when pressure/temperatures limits in the TS have been exceeded, one LTOP protection channel was removed from service for maintenance-related activities. During these events the redundant LTOP protection channel failed to mitigate the overpressure transient as a result of a system/component failure that had not been detected.

The reported LTOP transients have occurred in MODE 5 with RCS temperatures ranging from 80 Deg F to 190 Deg F. Since this temperature range includes MODE 6, RCS temperature less than 140 Deg F but with k-eff less than 0.95 as compared to k-eff less than 0.99 for MODE 5, the staff concludes that the additional administrative restriction for the single channel AOT is applicable to MODE 5 and MODE 6 (with the reactor pressure vessel head on).

The staff concludes that the LTOP system performs a safety-related function and inoperable LTOP equipment should be restored to an operable status in a shorter period of time. The current 7-day AOT for a single channel is considered to be too long under certain conditions. The staff has concluded that the AOT for a single channel should be reduced to 24 hours when operating in MODE 5 or 6 when the potential for an overpressure transient is highest. The operating reactor experiences indicate that these events occur during planned heatup (restart of an idle reactor coolant pump) or as a result of maintenance and testing errors while in MODE 5. The reduced AOT for a single channel in MODES 5 and 6 will help to emphasize the importance of the LTOP system in mitigating overpressure transients and provide additional assurance that plant operation is consistent with the design basis transient analyses.

Based on the foregoing concerns, added assurance of LTOP availability is to be provided by revising the current Technical Specification for Overpressure Protection to reduce the AOT for a single channel from 7 days to 24 hours when the plant is operating in MODES 5 or 6. Attachment B-1 is provided for guidance for Westinghouse and CE plants. The guidance provided is also applicable to plants that rely on both PORVs and RHR SRVs or that rely on RHR SRVs only. Attachment B-2 provides the staff bases for the Overpressure Protection Technical Specification.

In performing the studies for GI-94, the staff has assumed that the administrative controls and procedures identified in Section 1 have been implemented to ensure that the plant is being operated within the design base. If it is determined that the design base was developed based on restricted safety injection pump operability and/or differential temperature restrictions for RCP restart and that these restrictions have not been implemented as part of USI A-26, then these restrictions should be implemented now. This is not a new requirement. Attachment B-3 is provided for guidance.

Enclosure B to Generic Letter 90-06

Attachment B-1

Modified Technical Specifications
for Combustion Engineering and Westinghouse Plants

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.3 Two power-operated relief valves (PORVs) shall be OPERABLE with a lift setting of less than or equal to [450] psig.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal to [275]°F, MODE 5, and MODE 6 when the head is on the reactor vessel and the RCS is not vented through a ___ square inch or larger vent.

ACTION:

a. With one PORV inoperable in MODE 4, restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through at least a ____ square inch vent within the next 8 hours.

b. With one PORV inoperable in MODES 5 or 6, either (1) restore the inoperable PORV to OPERABLE status within 24 hours, or (2) complete depressurization and venting of the RCS through at least a ____ square inch vent within a total of 32 hours.

c. With both PORVs inoperable, complete depressurization and venting of the RCS through at least a ____ square inch vent within 8 hours.

d. With the RCS vented per ACTIONS a, b, or c, verify the vent pathway at least once per 31 days when the pathway is provided by a valve(s) that is locked, sealed, or otherwise secured in the open position; otherwise, verify the vent pathway every 12 hours.

e. In the event either the PORVs or the RCS vent(s) are used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence.

f. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9.3 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST, but excluding valve operation, at least once per 31 days; and

b. Performance of a CHANNEL CALIBRATION at least once per 18 months; and

c. Verifying the PORV isolation valve is open at least once per 72 hours.

Enclosure B to Generic Letter 90-

Attachment B-2

3/4.4.9.3 OVERPRESSURE PROTECTION SYSTEM

Bases of the Limiting Condition for Operation and Surveillance Requirements:

The OPERABILITY of the PORVs is determined on the basis of their being capable of performing the function to mitigate an overpressure event during low-temperature operation.

The Modified Standard Technical Specification (STS) requirements include the following changes from prior STS guidance:

1. The depressurizing and venting of the RCS is not classified as an overpressure protection system. However, the APPLICABILITY of the LCO excludes MODE 6 when the RCS is adequately vented. This avoids any possible question on Specification 3.0.4 being applied to preclude placement of the head on the vessel if any part of the LCO is not met when the RCS is vented.

2. The APPLICABILITY for MODE 6 is clarified as "when the head is on the reactor vessel" rather than as "MODE 6 with the reactor vessel head on."

3. ACTION a. is revised to clarify that it is only applicable in MODE 4.

4. ACTION b. was added to reduce the allowed outage time for an inoperable PORV to 24 hours in MODES 5 or 6. Because this LCO does not apply under certain conditions specified under the APPLICABILITY for this specification, the ACTION statements likewise do not apply under those conditions. ACTIONS a. and b. do not repeat those qualifying conditions that apply for these modes since the actions only apply when the unit is under those conditions.

5. ACTION d. includes the requirements to verify that ACTIONS a., b., or c. continue to be met on an ongoing basis when the unit would be in MODES 4, 5, or 6.

6. The Surveillance Requirements were simplified by removing requirements that exist because of the general requirements applicable to all surveillance requirements as specified in Section 4.0 of the TS.

7. Surveillance Requirement 4.4.9.3.2 was removed since it is addressed by ACTION d.

For plants with existing TS for PORVs used for LTOP, the only required change is that indicated to restrict the applicability of ACTION a. to MODE 4 and for incorporating ACTION b. Any other changes that are proposed consistent with the above guidance are voluntary. For a plant without existing TS for PORVs that are used for LTOP, a TS should be proposed that conforms to the above guidance.

Because some plants use residual heat removal (RHR) safety relief valves for LTOP, either in addition to or in lieu of PORVs, similar requirements are included in TS as noted above for PORVs. The same changes in ACTION requirements a. and b. are required, as noted above, for these plants. Likewise, any plant without existing TS for RHR suction relief valves that are used for LTOP should propose TS that are consistent with the above guidance. When only RHR safety relief valves are used for LTOP, the Surveillance Requirements would state: "No additional requirements other than those required by Specification 4.0.5."

Enclosure B to Generic Letter 90-

Attachment B-3

Technical Specifications Guidance
for Combustion Engineering and Westinghouse Plants

Operational Limitations Consistent With the Design Basis Assumptions for the Low-temperature Overpressure Protection (LTOP) System

The TS requirements for LTOP typically apply in MODE 4 when the temperature of any cold leg is below 275°F, MODE 5, and MODE 6 when the head is on the reactor vessel. During these conditions, one train (or channel) of the LTOP system is capable of mitigating an LTOP event that is bounded by the largest mass addition to the RCS or by the largest increase in RCS temperature that can occur. The largest mass addition to the RCS is limited based upon the assumption that no more than a fixed number of pumps are capable of providing makeup or injection into the RCS. Hence, this is a matter important to safety that pumps in excess of this design basis assumption for LTOP not be capable of providing makeup or injection to the RCS.

The capability for makeup and injection to the RCS is also a safety concern for normal makeup to the reactor coolant system for reactivity control as well as for events that could result in a loss of coolant from the RCS. The former are covered by Technical Specifications (TS) under Reactivity Control Systems, Charging Pump - Shutdown (MODES 5 and 6); Charging Pumps - Operating (MODES 1 through 4); and Flow Paths - Operating (MODES 1 through 4). The latter is covered by TS under Emergency Core Cooling Systems, ECCS Subsystems - T_{cold} Less Than 350°F (MODE 4).

The manner in which restrictions, consistent with the design basis assumptions of the LTOP system, have been incorporated in TS that require the operability of makeup or injection pumps has varied depending upon plant-specific considerations for the LTOP design and plant-specific designs for the use of pumps for makeup and ECCS functions. A common method has been the use of footnotes to the pump operability requirements to note that:

> A maximum of one Safety Injection [and/or] one charging pump shall be OPERABLE when the temperature of one or more of the RCS cold legs is less than 275°F.

This footnote is used for each specification that requires the operability of a safety injection and/or charging pump in MODES 4 or 5.

The Surveillance Requirements typically include the following:

> All Safety Injection [and/or] charging pumps, except the above required OPERABLE pump[s], shall be demonstrated to be inoperable by verifying that the motor circuit breakers are secured in the open position at least

once per 12 hours whenever the temperature of one or more of the RCS cold legs is less than or equal to 275°F.

Generally, it is preferable to include requirements for implementing the intent of an LCO as part of an LCO rather than to only define requirements, such as securing motor circuit breakers in the open position, in a Surveillance Requirement. Furthermore, the requirements for operable pumps could be stated in terms of requiring one pump to be operable rather in terms of "at least one pump shall be operable" and then including a footnote requiring that, in fact, no more than one pump shall be operable. The preferred alternative would be an LCO which stated:

> One Safety Injection [and/or] charging pump shall be operable and all other Safety Injection [and/or] charging pumps shall be secured with their motor circuit breakers in the open position.

The form of the above requirements for any given specification would be dependent upon which pumps are addressed by that specification, e.g., charging or injection pumps or both.

The surveillance requirements would be similar to that noted above with the following substitution:

> . . .except the above required OPERABLE pump(s), shall be demonstrated to be secured by verifying that the motor circuit breakers are in the open position. . . .

Changes to plant TS should be proposed to incorporate one of the above methods, to ensure that pumps are not capable of initiating a mass addition to the RCS that exceeds the design basis assumptions for the LTOP system, for plants that do not currently include such requirements.

The largest temperature increase in the RCS that could result in a challenge to the LTOP system is dependent upon the differential temperature between the RCS and the secondary system when starting a reactor coolant pump. Hence, this is also a matter important to safety when reactor coolant pumps are started and the resulting increase in RCS temperature is in excess of the design basis assumption for the LTOP system to mitigate the resulting increase in RCS pressure. The manner in which this design basis assumption of the LTOP system is reflected in TS has been the use of a footnote to the reactor coolant pump operability requirements to note that:

> A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to 275°F unless the secondary water temperature of each steam generator is less than ____°F above each of the RCS cold leg temperatures.

The above footnote has been included in the TS for residual heat removal under title of the Reactor Coolant System, Hot Shutdown.

Changes to plant TS should be proposed to incorporate the above method, to ensure that the starting of RCS pumps are not capable of initiating a pressure transient that exceeds the design basis assumptions for the LTOP system, for plants that do not currently have this requirement.

For GI-94, the actions proposed by the NRC staff to improve the availability of the low-temperature overpressure protection (LTOP) system, as identified in Section 3 of Enclosure B, represent a new interpretation of existing requirements for some licensees and CP holders, and this request is considered a backfit in accordance with NRC procedures. This backfit is a cost-justified safety enhancement. Therefore, an analysis of the type described in 10 CFR 50.109(a)(3) and 50.109(c) was performed, and a determination was made that there will be a substantial increase in overall protection of the public health and safety and that the attendant costs are justified in view of this increased protection. The analysis and determination will be made available in the Public Document Room with the minutes of the 167th and 168th meetings of the Committee to Review Generic Requirements.

This request is covered by Office of Management and Budget Clearance Number 3150-0011, which expires January 31, 1991. The estimated average burden hours is 320 person-hours per licensee response, including assessment of the new recommendations, searching data sources, gathering and analyzing the data, and preparing the required reports. Comments on the accuracy of this estimate and suggestions to reduce the burden may be directed to the Office of Management and Budget, Room 3208, New Executive Office Building, Washington, D.C. 20503, and the U.S. Nuclear Regulatory Commission, Information and Records Management Branch, Office of Information Resources Management, Washington, D.C. 20555.

Sincerely,
Original signed by
James G. Partlow
James G. Partlow
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Office of Nuclear Reactor Regulation

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Enclosures:
A. Staff Positions Resulting from Resolution of Generic Issue 70
B. Staff Positions Resulting from Resolution of Generic Issue 94
C. NUREG-1316, "Technical Findings and Regulatory Analysis Related to Generic Issue 70--Evaluation of Power-Operated Relief Valve and Block Valve Reliability in PWR Nuclear Power Plants"
D. NUREG-1326, "Regulatory Analysis for the Resolution of Generic Issue 94, Additional Low-Temperature Overpressure Protection for Light-Water Reactors"

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