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TECHNICAL EVALUATION OF THE ELECTRICAL, INSTRUMENTATION, AND CONTROL DESIGN
ASPECTS OF THE LOW TEMPERATURE OVERPRESSURE PROTECTION SYSTEM FOR
THE POINT BEACH NUCLEAR POWER PLANT, UNITS 1 AND 2

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March, 1979

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Discussion

The present study is the first to examine the effects of a single session of a group-based, self-paced, and self-monitored exercise program on the health-related quality of life of sedentary, middle-aged women. The results of this study suggest that a single session of a group-based, self-paced, and self-monitored exercise program can improve the health-related quality of life of sedentary, middle-aged women. The results of this study also suggest that a single session of a group-based, self-paced, and self-monitored exercise program can improve the physical, mental, and social functioning of sedentary, middle-aged women. The results of this study also suggest that a single session of a group-based, self-paced, and self-monitored exercise program can improve the overall health-related quality of life of sedentary, middle-aged women. The results of this study also suggest that a single session of a group-based, self-paced, and self-monitored exercise program can improve the physical, mental, and social functioning of sedentary, middle-aged women. The results of this study also suggest that a single session of a group-based, self-paced, and self-monitored exercise program can improve the overall health-related quality of life of sedentary, middle-aged women.

ABSTRACT

This report documents the technical evaluation of the electrical, instrumentation, and control design aspects of the low temperature over-pressure protection system for the Point Beach nuclear power plant, Units 1 and 2. Design basis criteria used to evaluate the acceptability of the system included operator action, system testability, single failure criterion, and seismic Category I and IEEE Std-279-1971 criteria. This report is supplied as part of the Selected Electrical, Instrumentation, and Control Systems Issues Support Program being conducted for the U. S. Nuclear Regulatory Commission by Lawrence Livermore Laboratory.

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FOR THE POINT BEACH NUCLEAR POWER PLANT, UNITS 1 AND 2

1. INTRODUCTION

By letter to the Wisconsin Electric Power Company (WEPCO) dated 11 August 1976, the U. S. Nuclear Regulatory Commission (NRC) requested an evaluation of system designs to determine susceptibility to overpressurization events and an analysis of these possible events, and proposed interim and permanent modifications to the systems and procedures to reduce the likelihood and consequences of such events. By letter dated 3 September 1976 and subsequent letters (refer to the Appendix), the Wisconsin Electric Power Company submitted the additional information requested by the NRC staff, including the administrative operating procedures and the proposed low temperature overpressure protection mitigating system. The system hardware includes sensors, actuating mechanisms, alarms, and valves to prevent a reactor coolant system transient from exceeding the pressure and temperature limits of the Technical Specifications for Point Beach Units 1 and 2, as required by the Code of Federal Regulations, Title 10, Part 50 (10 CFR 50), Appendix G.

The purpose of this report is to evaluate the Licensee's equipment and procedures based on the information provided (refer to the Appendix), and to define how well they meet the criteria established by NRC as necessary to prevent unacceptable overpressurization events.

2. EVALUATION OF POINT BEACH UNITS 1 AND 2

2.1 INTRODUCTION

Review of the Point Beach Units 1 and 2 low temperature overpressure protection system design by WEPCO was begun in 1976 at NRC's request. The overall approach to eliminating overpressure events incorporates administrative, procedural, and hardware controls with reliance upon the plant operator as the principal line of defense. Preventive administrative/procedural measures include:

- (1) Procedural precautions.
- (2) Deenergization (power removed) of nonessential and essential components which are not required to be operable during the cold shutdown mode of operation.
- (3) Maintenance of a non-water-solid reactor coolant system condition whenever possible.
- (4) Incorporation of a low pressure relief setpoint for the existing power operated relief valve (PORV) control logic.

The design basis criteria that were applied in evaluating the acceptability of the electrical, instrumentation, and control aspects of the low temperature overpressure protection system (OPS) are:

- (1) Operator Action. No assumption of operator action is made until ten minutes after the operator is aware, through an action alarm, that a pressure transient is in progress.
- (2) Single Failure Criterion. The OPS shall be designed to protect the reactor vessel given a single failure which is in addition to the failure that initiated the pressure transient.
- (3) System Testability. The OPS must be testable on a periodic basis prior to dependence on the OPS to perform its function.

- (4) Seismic Category I and IEEE Std-279-1971 Criteria. The OPS should satisfy both the seismic Category I and IEEE Std-279-1971 criteria. The basic objective is that the OPS should not be vulnerable to a failure mode that would both initiate a pressure transient and disable the low temperature overpressure mitigating system. Events such as loss of instrument air and loss of offsite power must be considered.

2.2 WEPCO OVERPRESSURIZATION MITIGATING SYSTEM DESIGN

The WEPCO Overpressurization Mitigating System (OMS) design information detailed in this section was derived from References 1 and 5 in the Appendix. The WEPCO design for the Point Beach OMS consists of two methods of relieving pressure during periods of water-solid operation and when the reactor coolant system (RCS) temperature is below the value permitted to perform the primary system leak test. The primary method of pressure relief utilizes the pressurizer PORV's, and the diverse method utilizes the relief valves on the residual heat removal (RHR) system.

The PORV's are made operational for low-pressure relief by utilizing a dual setpoint where the low-pressure circuit is energized and deenergized, depending on plant conditions, by the operator with a keylock switch. The logic required for the low-pressure setpoint is in addition to the existing PORV actuation logic and will not interfere with existing automatic or manual actuation of the PORV's.

The relief valves on the RHR system are available for pressure relief whenever the RHR system is connected to the RCS. The RHR system is normally connected to the RCS during plant conditions when overpressurization events have been most prevalent, i.e., during low-temperature and low-pressure conditions. The RHR-system relief valves can be considered a diverse relief system at Point Beach because the RHR-system isolation valves do not automatically isolate the RHR system during a pressure transient, thereby making the relief valves available throughout the transient.

During plant cooldown and prior to the collapse of the steam bubble in the pressurizer, the operator acting under administrative procedure places the keylock switch in the "low pressure" position and connects the RHR system to the RCS. An alarm will alert the operator when the pressure is sufficiently low so that activation of the low-pressure setpoint circuit will not inadvertently open a PORV. Placing the keylock switch in the "low pressure" position blocks the alarm indicating low pressure and enables the low-temperature high-pressure alarm.

During plant heatup, the operating procedure will identify the plant conditions for which low-pressure protection is no longer needed. The operator places the keylock switch in the "normal" position, thereby returning control of the PORV's to the operating high-pressure condition and avoiding inadvertent opening of the PORV's. Placing the keylock switch in the "normal" position removes or blocks the low-temperature high-pressure alarms and enables the low-pressure operation alert alarm.

Plant operation and pressure tests utilizing the OMS during heatup and cooldown are described below:

- (1) The 425-psig setpoint is a constant value on the heatup curves for Point Beach Units 1 and 2. During heatup, the OMS will be in the enable mode until the temperature is reached at which the inservice pressure test can be performed. At that time, the RCS is separated from the RHR system, and the pressure test is performed. The pressure test is an allowed violation of the 10 CFR 50, Appendix G curve. Following the pressure test, normal pressurization and heatup will continue.
- (2) The likelihood of exceeding the 10 CFR 50, Appendix G curve above the temperature allowed for pressure testing is very small. The safety injection (SI) pumps need not be considered because their shutoff head is at slightly less than 1600 psig and the allowable pressure at 300°F is 1800 psig. The only pumps that can continue to pressurize the system are the low capacity (40 gpm) charging pumps. If this should happen, it would be similar to the pressurization allowed in the inservice pressure test and would not cause a problem.
- (4) The cooldown is the reverse of heatup, i.e., the plant is depressurized to below 425 psig and the OMS is manually enabled before the plant temperature drops below the temperature at which the inservice pressure test can be performed.

The OMS is comprised of two redundant channels, each consisting of a sensor, power supply, alarm bistable, alarm status light, PORV bistable, keylock switch, valve actuation circuitry, and PORV. The low-pressure alarm and the low-temperature high-pressure alarm of one channel each share an annunciator with the similar alarm of the other channel.

The redundant channels allow the single failure criterion to be met. Equipment quality and qualification are consistent with the classification of the system, and channel integrity will be maintained. The system is designed against multiple failures resulting from a credible single event. The system inputs are derived from signals that are a direct measure of the desired variable, and the system is capable of test and calibration.

The source of electrical power for the Channel No. 1 pressure transmitter, bistable auxiliary relays, and valve actuation circuitry will be different from those for Channel No. 2. The OMS is completely separate from the present plant protection system and, therefore, degrading of the plant protection system is completely avoided.

An enable status light, provided for each PORV, will be lighted when its respective system is activated and the valve is able to perform its intended function. In order for the enable status light to indicate an active system, the enable/disable key-operated switch must be in the "enable" position, and the isolation valve associated with the PORV must be in the "open" position.

The total OMS, i.e., both PORV channels and the RHR-system relief valves, has been examined for possible modes of failure and their resulting effect on the mitigating system pressure relief capability. The results of this evaluation reveal that under all possible failure assumptions at least one relief valve is available. The RHR-system relief valve will be adjusted to open at 500 psig, resulting in a flow rate of 990 gpm with 10% accumulation. This will handle the maximum assumed injected flow that can occur by the starting of a single SI pump. The pump injection flow rate at 500 psig is less than the RHR-system relief valve capability at that pressure.

2.3 EVALUATION OF POINT BEACH UNITS 1 AND 2 USING DESIGN BASIS CRITERIA

Point Beach Units 1 and 2 were evaluated under the guidance of the four design basis criteria stated in Section 2.1 of this evaluation, and with specific attention given to various pertinent NRC staff positions resulting from these criteria. Sections 2.3.1 through 2.3.4 are concerned with the four design basis criteria.

2.3.1 Operator Action

In each design basis transient analyzed, no credit for operator action was assumed until 10 minutes after the initiation of the RCS overpressurization transient and after the operator is made aware of the overpressure transient by the low temperature overpressure transient alarm.

2.3.2 Single Failure Criterion

WEPCO states in References 1 and 4 that the OMS is designed to protect the reactor vessel given a single failure in addition to the failure that initiated the overpressure transient. Redundant or diverse pressure protection channels are used to satisfy the single failure criterion. Redundant pressure sensors, bistables, keylock switches, two full-capacity PORV's with independent power sources, and two RHR-system relief valves are provided for the long-term OMS.

We conclude that the WEPCO Point Beach Units 1 and 2 OMS satisfies the NRC staff single failure criterion.

2.3.3 System Testability

The NRC staff position requires that the OMS control circuitry from pressure sensor to valve solenoid shall be tested prior to each heatup and cooldown. The PORV's should be tested during each refueling. Deviations from these criteria should be justified.

WEPCO states in Attachment A of Reference 1 and Appendix D of Reference 5 that the OMS system is capable of test and calibration.

The safety evaluation report (SER), dated January 1978, by the NRC Reactor Safety Branch/Division of Operating Reactors (RSB/DOR) for the Point Beach Units 1 and 2 OMS states that:

- (1) Testability will be provided.
- (2) WEPCO has stated that verification of the operability of the OMS control system will be performed prior to entering water-solid conditions.
- (3) PORV testing will be performed during each refueling outage.
- (4) Testing requirements will be incorporated in the Technical Specifications as discussed in Section 4.2 of this evaluation.

We conclude that WEPCO OMS system testability is insufficiently detailed in the reference material. It is recommended that the NRC staff ensure that thorough surveillance of the OMS from sensor to valve solenoid and proper PORV testing are adequately described in the WEPCO Point Beach Units 1 and 2 Technical Specifications.

2.3.4 Seismic Design and IEEE Std-279-1971 Criteria

WEPCO states that seismic Category I and IEEE Std-279-1971 criteria were considered in the design of the OMS. The PORV's and associated instrumentation and control hardware will serve as the long-term mitigating system for low temperature RCS overpressurization, based on the existing applicable plant criteria and the following considerations derived from References 1 and 4:

- (1) The mitigating system is designed against single failure. The system is not vulnerable to a failure mode that would both initiate a pressure transient and disable the OMS.

- (2) The PORV's are designed to withstand seismic loadings equivalent to 3.0 g in the horizontal direction and 2.0 g in the vertical direction. The PORV's are mounted on the pressurizer in a manner that maintains primary system integrity during the design seismic event. To ensure that the PORV's have an air supply for proper operation, a gas cylinder for each valve will be seismically mounted to an adjacent Category I structure and with air lines to the valves. The air lines will tolerate differential movements of the components and structures. The remainder of the system is located within Category I structures, trays, and cabinets.
- (3) The long-term OMS satisfies IEEE Std-279-1971 criteria because of the channel separation, diverse relief capability of the RHR system, and independent and redundant pressure sensors along with their associated electronics, alarms and PORV's. In addition, both channels have separate power supplies.

We conclude that the WEPCO Point Beach Units 1 and 2 OMS satisfies the NRC staff seismic design and IEEE Std-279-1971 criteria.

2.4 ALARM SYSTEMS DESIGNS AND OPERATION

Specific details concerning alarm systems designs and operation for the OMS are described below.

2.4.1 High-Pressure Alarm

The NRC staff position requires that a high-pressure audio/visual alarm shall be used during low RCS temperature operations as an effective means to provide unambiguous information and alert the operator that a pressure transient is in progress.

WEPCO states in Appendix A of Reference 5 that the high-pressure alarm system design is as follows:

- (1) A separate single output channel and a dual output channel have been added to the wide-range reactor coolant pressure indication loop. The separate single output channel enables an alarm in the control room, and each channel of the dual output channel causes a PORV to open. The three bistable setpoints are independently adjustable. The entire circuit is key-switched from the main control board. The setpoints for opening the PORV's will be approximately 25 psi apart and that the alarm setpoint will be lower than the opening setpoint of either valve. The operator will activate the system before the RCS is water-sol

when the plant is being shutdown, and will deactivate the system after a steam bubble has been drawn prior to plant heatup.

- (2) The high pressure alarm used during low RCS temperature operations is an audible signal and a flashing light. It consists of a single output channel from the wide-range reactor coolant pressure indication loop A hot leg (PT-420). The alarm bistable is set to alarm at 425 psig and resets at approximately 390-395 psig. Alarm setpoint adjustments can be made by adjusting the bistable trip point.
- (3) The high-pressure alarm availability and operability are assured by administrative controls in the plant's cooldown and heatup procedures. These procedures require that the reactor operator activate the low-temperature high-pressure relief system and its alarm prior to cooling the RCS below 300°F. These circuits are activated by means of a keyswitch on the main control board. The procedure for heatup requires this system to be locked-out when the RCS temperature is greater than 300°F. The high-pressure alarm availability and reliability will be maintained by a periodic program of checks, tests, and calibration similar to that done for other instrumentation channels.

We conclude that this design satisfies the NRC staff position.

2.4.2 Isolation Valve Alarm

The NRC staff position requires that

- (1) The upstream isolation valve shall be wired into the overpressure protection alarm in such a way that the alarm will not clear unless the system is enabled and the isolation valve is open.
- (2) The alarm shall be of the audio/visual type and provide unambiguous information to the operator.

WEPCO states in Attachment A of Reference 1 that the isolation valve alarm system design is as follows:

- (1) Each PORV has an enable status light which will be lighted when its respective system is activated and the valve is able to perform its intended function. In order for the enable status light to indicate an active system, the enable/disable key-operated switch must be in the "enable" position and the isolation valve associated with the PORV must be in the "open" position.

- (2) The indicator for the keyswitch for PORV 430 is interlocked through the indication circuit for MOV 516 in order to provide additional assurance to the operator that MOV 516 is open when the overpressurization circuit is activated. In the same manner, the keyswitch indicator for PORV valve 431C is interlocked through the indication circuit for MOV 515.
- (3) MOV 516 is the isolation valve in series with PORV 430. MOV 515 is the isolation valve for PORV 431C.

We conclude that this design does NOT fully satisfy IEEE Std-279-1971(4.20) and the NRC staff position. We recommend the installation of an audio/visual alarm that provides unambiguous information to the operator.

2.4.3 Enable Alarm

The NRC staff position requires that

- (1) An alarm shall be activated as part of the plant cooldown process to ensure that the PORV "low" setpoint is activated.
- (2) The alarm shall be of the audio/visual type and provide unambiguous information to the operator.

WEPCO states in Appendix D of Reference 5 that the enable alarm system design is as follows:

- (1) The operator, acting under administrative procedure, places the keylock switch in the "low pressure" position and connects the RHR system to the RCS during plant cooldown and prior to the collapse of the steam bubble in the pressurizer. Placing the keylock switch in the "low pressure" position blocks the alarm indicating low pressure and enables the low-temperature high-pressure alarm.
- (2) An alarm will alert the operator when the pressure is 425 psig so that activation of the low-pressure setpoint circuit will not inadvertently open a PORV.

We conclude that this design satisfies the NRC staff position.

2.4.4 Disable Alarm

The NRC staff position requires that

- (1) An alarm shall be activated as part of the plant heatup process to ensure that the PORV's are reset to the "high" setpoint.

- (2) The alarm shall be of the audio/visual type and provide unambiguous information on to the operator.

WEPCO states in Appendix D of Reference 5 that the disable alarm system design is as follows:

- (1) During plant heatup, the operating procedure will identify the plant conditions for which low-pressure protection is no longer needed.
- (2) The operator places the keylock switch in the "normal" position, thereby returning control of the PORV's to the operating high-pressure setpoint. Placing the keylock switch in the "normal" position removes or blocks the low-temperature high-pressure alarm and enables the low-pressure operation alert alarm.
- (3) An alarm will alert the operator when a high-pressure condition requires a switching action to avoid inadvertent opening of the PORV's.

We conclude that this design satisfies the NRC staff position.

2.4.5 PORV Open Alarm

The NRC staff position requires that an audio/visual alarm shall be activated to provide unambiguous information and alert the operator that a PORV is in the "open" position.

WEPCO states in Appendix B of Reference 5 that the PORV open alarm system design is as follows:

The pressurizer PORV's have an open/shut indicator on the main control board.

We conclude that this design does NOT fully satisfy IEEE STD-279-1971(4.20) and the NRC staff position. We recommend the installation of an audio/visual alarm that provides unambiguous information to the operator.

2.5 PRESSURE TRANSIENT REPORTING AND RECORDING REQUIREMENTS

The NRC staff position is that a pressure transient which causes the OMS to function, thereby indicating the occurrence of a serious pressure transient, is a 30-day reportable event. In addition, pressure-recording and temperature-recording instrumentation are required to provide a permanent record of the pressure transient. The response time of the pressure/temperature recorders shall be compatible with pressure transients that increase at a rate of approximately 100 psig per second.

WEPCO states in Attachment 1 of Reference 4 that instrumentation and recording equipment is available at Point Beach Units 1 and 2 and is operational during shutdowns to record both pressure and temperature in the event of an overpressurization transient. This equipment is located on the main control board in the control room and is listed below:

- (1) Wide Range Reactor Coolant Pressure Recorder
 - (a) Range: 0-3000 psig.
 - (b) Paper recording width: 4 inches.
 - (c) Paper speed: 3/4 inch per hour.
- (2) Cold Leg Temperature Recorder
 - (a) Range: 50°F-650°F.
 - (b) Paper recording width: 4 inches.
 - (c) Paper speed: 3/4 inch per hour.
 - (d) Two pens: One records loop A temperature and the other loop B temperature.
- (3) Residual Heat Removal Temperature Recorder
 - (a) Range: 100°F-400°F.
 - (b) Paper recording width: 4 inches.
 - (c) Paper speed: 3/4 inch per hour.
 - (d) Two pens: One records inlet temperature and the other outlet temperature.

We conclude that this implementation, if properly incorporated in the WEPCO Point Beach Units 1 and 2 Technical Specifications, satisfies the NRC staff position.

2.6 DISABLING OF ESSENTIAL COMPONENTS NOT REQUIRED DURING COLD SHUTDOWN

The NRC staff position requires the deenergizing of safety injection system (SIS) pumps and the closure of safety injection (SI) header/discharge valves during cold shutdown operations.

WEPCO states in Attachment A of Reference 5 that this will be accomplished by the following changes to the "low power operation to cold shutdown" operating procedures prior to reducing the temperature below the minimum required for full pressurization:

- (1) The high head SI pumps will be isolated electrically with the breaker locked in the "off" position.
- (2) The high head SIS discharge motor-operated isolation valves, MOV-866A&B, will be shut with the breakers for the motor operators locked in the "off" position.
- (3) The accumulator discharge motor-operated isolation valves will be shut with the breakers for the motor operators locked in the "off" position.

We conclude that this implementation, if properly incorporated in the WEPCO Point Beach Units 1 and 2 Technical Specifications, satisfies the NRC staff position.

3. TECHNICAL SPECIFICATIONS

The Technical Specifications information detailed in this section was derived from the RSB/DOR SER entitled, "Safety Evaluation Report of the Overpressure Mitigating System for Point Beach Units 1 and 2", dated January 1978.

To ensure operation of the OMS, the Licensee is to submit for NRC staff review its Technical Specifications for incorporation into the license for Point Beach Units 1 and 2. The Licensee should ensure that the proposed Technical Specifications are compatible with other Licensee requirements and are consistent with the intent of the statements listed below:

- (1) The Point Beach Units 1 and 2 OMS must be operable whenever the RCS temperature is below the value at which inservice pressure testing may be performed. The OMS setpoint is to be incorporated in the Technical Specifications. Operability of the OMS requires that the system is enabled, the upstream isolation valves are open, and the backup air supply is charged. In case one redundant train (control circuitry and associated relief valve) is inoperable for more than seven days, either a vapor bubble must be established in the pressurizer, or the primary system must be depressurized and vented to the atmosphere within eight hours. If both redundant trains are inoperable, either a vapor bubble must be established in the pressurizer or the primary system must be depressurized within eight hours.
- (2) Suitable surveillance requirements are to be proposed consistent with the need for use of the OMS.
- (3) Electrical deenergizing of the high pressure safety injection (HPSI) pump and appropriate isolation MOV's and reenergizing of electrical power to these components are to be incorporated in the Technical Specifications.
- (4) Surveillance requirements consistent with the assumption that the RCS/steam generator (SG) differential temperature is less than or equal to 50°F are to be proposed. It is noted that calculations based on a RCS/SG ΔT of 50°F result in ample margins to the 10

CFR 50, Appendix G curves. If WEPCO chooses to demonstrate that larger values of RCS/SG ΔT are acceptable with respect to violations of the 10 CFR 50, Appendix G curves, then corresponding relaxation of surveillance requirements will be accepted.

- (5) Operation of the OMS (i.e., PORV's and/or RHR relief valves) to relieve a pressure transient is to be reported.

We recommend that final acceptance of the WEPCO Point Beach Units 1 and 2 OMS by the NRC staff be contingent upon the Licensee's incorporating into its Technical Specifications statements which will adequately

- (1) Comply with those statements listed above.
- (2) Describe the thorough surveillance of the OMS from sensor to valve solenoid and proper PORV testing (refer to Section 2.3.3).
- (3) Describe the implementation of the pressure transient instrumentation reporting and recording equipment (refer to Section 2.5).
- (4) Describe the implementation of WEPCO's changes to its operating procedures for low power operation to cold shutdown (refer to Section 2.6).

4. CONCLUSIONS

The electrical, instrumentation, and control (EI&C) design aspects of the low temperature overpressurization mitigating system (OMS) for Point Beach Units 1 and 2 were evaluated using those design criteria originally prescribed by the NRC staff and later expanded during subsequent discussions with the Licensee.

We recommend that the NRC staff find the following EI&C aspects of the WEPCO Point Beach Units 1 and 2 OMS design acceptable:

- (1) Operator action
- (2) Single failure criterion
- (3) Seismic Category I and IEEE-279-1971
- (4) High pressure alarm
- (5) Enable alarm
- (6) Disable alarm.

We recommend that the NRC staff find the following EI&C aspects of the WEPCO Point Beach Units 1 and 2 OMS design acceptable ONLY AFTER they are adequately described in the WEPCO Point Beach Units 1 and 2 Technical Specifications:

- (1) System testability
- (2) Pressure transient reporting and recording requirements
- (3) Disabling of essential components not required during cold shutdown.

We recommend that the NRC staff find the following EI&C aspects of the WEPCO Point Beach Units 1 and 2 OMS design unacceptable:

- (1) Isolation valve alarm
- (2) PORV open alarm

We recommend the installation of an audio/visual isolation valve alarm and an audio/visual PORV open alarm that provide unambiguous information to the operator.

APPENDIX

REFERENCES

- *1. WEPCO letter (Burstein) to NRC (Case) dated July 28, 1977.
- *2. WEPCO letter (Burstein) to NRC (Case) dated October 28, 1977.
- *3. WEPCO letter (Burstein) to NRC (Rusche) September 3, 1976.
- *4. WEPCO letter (Burstein) to NRC (Rusche) dated October 14, 1976.
- *5. WEPCO letter (Burstein) to NRC (Rusche) dated March 2, 1977.
- *6. WEPCO letter (Burstein) to NRC (Rusche) dated April 18, 1977.
- 7. "Staff Discussion of Fifteen Technical Issues Listed in Attachment G, 3 November 1976 Memorandum from Director NRR to NRR Staff", NUREG-0138, November 1976.
- *8. NRC letter (Lear) to WEPCO (Burstein) dated August 11, 1976.
- 9. "Pressure Mitigating System Transient Analysis Results" prepared by Westinghouse for the Westinghouse User's Group on Reactor Coolant System Overpressurization, July 1977.
- 10. "Supplement to the July 1977 Report, Pressure Mitigating Systems Transient Analysis Results," prepared by Westinghouse for the Westinghouse User's Group on Reactor Coolant System Overpressurization, September 1977.

*Copy included in this evaluation.

Wisconsin Electric POWER COMPANY
231 WEST MICHIGAN, MILWAUKEE, WISCONSIN 53201

Regulatory

File Cys

Reference 1

July 28, 1977

Mr. Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation
U. S. NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555

ATTENTION: Mr. George Lear, Chief
Operating Reactors Branch #3

Dear Mr. Case:

DOCKET NOS. 50-266 AND 50-301
PRESSURE MITIGATING SYSTEM FOR OVERPRESSURIZATION PROTECTION
POINT BEACH NUCLEAR PLANT UNIT NOS. 1 AND 2

This letter provides final details of the proposed pressure mitigating system for Point Beach Nuclear Plant Units 1 and 2 and forwards four (4) copies of the report prepared by Westinghouse Electric Corporation titled, "Pressure Mitigating Systems Transient Analysis Results".

The transient analysis report presents the results of pressure transient analyses for the reactor coolant system of a pressurized water reactor during low temperature, water solid operation when either mass or heat is injected into the primary system. The analyses were performed using conservative bounding input parameters plus parameter sensitivity studies to provide for results applicable to plant specific parameters. For the cases presented, the use of a nominal two inch air operated relief valve, such as the pressurizer power operated relief valve, is shown to mitigate the pressure transient without the need for immediate operator intervention. An algorithm is presented for selection of the relief valve setpoint to avoid violation of the 10 CFR 50 Appendix G pressure limitation for the reactor vessel.

Attachment A to this letter provides details of the proposed pressure mitigating system which is identical to that described in our March 2, 1977, letter except for the following additional information or system features:

1. The completion of the transient analysis provides information for sizing the air supply for an assumed loss of instrument air. To ensure that the valves have a sufficient air supply a gas cylinder for each valve will be utilized.

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July 28, 1977

2. The transient analysis report provides an algorithm for setpoint determination. The setpoint determination for Point Beach Units 1 and 2 is included in Appendix A to this letter.
3. At the request of the NRC, the open position of the isolation valve is needed in order to light the enable status light.

The interim mitigating system described in Appendix C of our March 2, 1977, letter has been installed and is operational on both Units 1 and 2. The final mitigating system as described in Attachment D of the March 2 letter and enclosed as Attachment A to this letter will be installed at the first possible refueling after NRC approval has been received. Our March 2 letter indicated that the final mitigating system could be installed in Unit 1 during the fall 1977 refueling if all technical issues were settled and the procurement and receipt of the necessary hardware proceeded without delay. Unfortunately this has not been the case. Since submitting the proposed mitigating system for NRC approval on March 2, 1977, we have not received any notice as to the acceptability of the proposed system or that the concept of using pressurizer power operated relief valves is acceptable. Therefore, the installation of the final mitigating system in Unit 1 this fall may no longer be possible.

The request has been made by the NRC that we submit proposed Technical Specification changes associated with the design and operation of the pressure mitigating system. It is our judgement that the design of the mitigating system, the procedure governing the operation of the mitigating system and the other plant systems when in the low temperature low pressure mode of operation, and the present reporting requirements of the Technical Specifications adequately address the NRC concern for additions to the Point Beach Technical Specification. Accordingly, no additions to the Technical Specifications are needed or are being proposed.

The design of the pressure mitigating system described in Attachment A, the enclosed transient analysis report, plus the responses to NRC questions submitted by our March 2, 1977, letter address all known NRC concerns associated with the overpressurization issue. We believe the installation of this pressure mitigating system in our facility should resolve this issue. Should you have any questions concerning the material attached hereto, or associated with previous correspondence, please contact us.

Very truly yours,

Executive Vice President

Sol Burstein

Attachments

ATTACHMENT A

OVERPRESSURIZATION MITIGATING SYSTEM DESCRIPTION

This attachment describes the system that will be utilized at Point Beach Nuclear Plant Units 1 and 2 to provide protection against exceeding 10 CFR 50 Appendix G limits during periods of water solid operation. This system will be referred to as the long term solution to pressure transients during water solid operation. This system provides significantly improved reliability over the Interim System presently installed and provides sufficient protection against overpressurization when the primary system is water solid.

Installation Schedule

- Unit -1 will be refueled in October, 1977. Assuming all technical issues can be settled and proper equipment procured by that date, the "Overpressurization Mitigating System" will be installed during this October, 1977, refueling. If delays are encountered in either of these areas, the next refueling in October, 1978, would be the installation date.
- Unit -2 will be refueled in March, 1978. Assuming all technical issues can be settled and proper equipment procured by that date, the system will be installed during this March, 1978, refueling. If delays are encountered in either of these areas, the next refueling scheduled for a year later would be the installation date.

System Description

The "Overpressurization Mitigating System" consists of two diverse means of relieving pressure during periods of water solid operation and when system temperature is below the value permitted to perform the primary system leak test. The primary method of water relief utilizes the pressurizer power operated relief valves (PORV's) and the diverse method uses the relief valves on the Residual Heat Removal System. The PORV's are made operational for low pressure relief by utilizing a dual setpoint where the low pressure circuit is energized and de-energized by the operator with a keylock switch depending on plant conditions. The logic required for the low pressure setpoint is in addition to the existing PORV actuation logic and will not interfere with existing automatic or manual actuation of the PORV's. The relief valve on the Residual Heat Removal (RHR) System is available for water relief whenever the RHR System is connected to the Reactor Coolant System. The RHR System is normally connected to the Reactor Coolant System during plant conditions when overpressurization events have been most prevalent, i.e., low temperature and low pressure conditions. The RHR relief valve can be considered a diverse relief system at Point Beach because the RHR isolation valves do not automatically isolate the RHR System during a pressure transient, thus making the valve available throughout the transient.

During plant cooldown and prior to the collapse of the steam bubble in the pressurizer, the operator under administrative procedure places the keylock switch in the "Low Pressure" position and connects the RHR System to the Reactor Coolant System. An alarm will alert the operator when the pressure is sufficiently low so that activation of the low pressure setpoint circuit will not inadvertently open a PORV. Placing the keylock switch in the "Low Pressure" position blocks the alarm indicating low pressure and enables the low temperature high pressure alarm.

During plant heatup, the operating procedure will identify the plant conditions for which low pressure protection is no longer needed. The operator places the keylock switch in the "Normal" position, returning control of the PORV's to the operating high pressure setpoint. An alarm will alert the operator to the high pressure condition requiring the switching action to avoid inadvertent opening of the PORV's. Placing the keylock switch in the "Normal" position removes or blocks the low temperature high pressure alarm and enables the low pressure operation alert alarm.

A logic diagram of the "Overpressurization Mitigating System" is shown in Figure A-1. Two redundant channels are provided with each channel consisting of a sensor, power supply, alarm bistable, alarm status light, PORV bistable, keylock switch, valve actuation circuitry and PORV. The low pressure alert and the low temperature high pressure alert of one channel share an annunciator with the similar alarm of the other channel.

Figure A-2 shows a simplified instrumentation loop diagram for the two pressure channels. The source of electrical power to channel 1's pressure transmitter, bistable auxiliary relays and valve actuation circuitry will be different from those of channel 2's. The "Overpressurization Mitigating System" is completely separate from the present plant protection system and therefore, degrading of the plant protection system is completely avoided. We have attempted to design a system that has a minimum impact on the plant but meets many of the requirements of IEEE 279, e.g.; the redundant channels allow the single failure criteria to be met, equipment quality and qualification is consistent with the classification of the system, channel integrity will be maintained, the system is designed against multiple failures resulting from a credible single event, the system inputs are derived from signals that are a direct measure of the desired variable, and the system is capable of test and calibration. Complete separation and independence of the two channels are not considered necessary because of the diverse relief capability on the RHR System. Because the existing pressure channel is treated as "control grade", it would be very difficult and detrimental to overall plant safety to make the system "safety grade". The resulting design of the "Overpressurization Mitigating System" considers the system to be treated as "control grade" because the increase in safety risk (decrease in safety) of installing a protection grade system in the plant is greater than any increases in safety a protection grade system would provide over and above the proposed "Overpressurization Mitigating System". It is also our belief that any overpressurization mitigating system should be treated as a control grade system due to the safety margins inherent in the Appendix G limits and the availability of the diverse relief system on the RHR system.

The circuit for the PORV's is shown in Figure A-3. Separate DC power sources are associated with each valve. The valves are designed to withstand seismic loading equivalent to 3.0g in the horizontal direction and 2.0g in the vertical direction and they are mounted on the pressurizer in a manner to maintain primary system integrity during the design seismic event. To ensure that the valves have an air supply for proper operation, a gas cylinder as shown in Figure A-4 for each valve will be seismically mounted to an adjacent category I structure with air lines to the valves that will tolerate differential movements of components and structures. The remainder of the system is located within category I structures, trays and cabinets.

An enable status light is provided for each PORV which will be lighted when the system for that valve is activated and the valve is able to perform its intended function. In order for the status light to indicate an active system, the enable/disable key operated switch must be in the enable position and the isolation valve associated with the PORV must be in the open position. The circuit for this status light is shown in Figure A-5.

The total system, i.e., both channels of PORV plus the RHR relief valve, has been examined for possible modes of failure and the resulting effect on the mitigating system pressure relief capability. The results of this evaluation are given in Table A-1 and show that under all possible failure assumptions at least one relief valve is available. The RHR relief capacity versus opening set pressure is shown in Figure A-6. The RHR relief valve will be adjusted to open at 500 psig, resulting in a flow rate of 990 gpm with 10% accumulation. This will handle the maximum assumed injected flow which can occur by the starting of a single safety injection pump. The flow delivery characteristics of a single safety injection pump injecting water into the primary system is shown in Figure A-7. The pumped injection flow rate at 500 psig. is less than the RHR relief valve capability at that pressure.

The "Overpressurization Mitigating System" described herein provides adequate and sufficient protection to the Reactor Coolant System against overpressurization transients. When coupled with the diverse relief capability on the RHR System triple protection is provided. The design of this system is selected to be compatible with present design of the plant, thus minimizing modification and backfit to existing equipment. It is our belief that overall safety can best be provided by minimizing modifications to and backfitting of existing plant equipment and systems. Major modifications to nuclear plants have the potential to degrade the existing plant design and whenever construction work is required the potential for a serious fire is greatly increased. Any system that is more complex than the simple and straight forward system described herein, and requires additional changes to the plant, should be examined closely to determine if the added features are worth the cost and risk of installing those features. This backfit consideration applies more to operating plants because of the difficulty in making complex modifications to existing systems than to plants under construction or in the design phase. The "Overpressurization Mitigating System" provides the proper balance between these competing concerns at the Point Beach Nuclear Plant.

Air Requirements For PORV Operation

The volume of air required to operate power operated relief valves 430 and 431C in their present configuration is approximately 450 cubic inches per valve per cycle. This can be reduced to approximately 150 cubic inches if a spacer plate for each valve is installed. The spacer plate can be ordered from Copes-Vulcan. The portable gas cylinders used at Point Beach have a volume of one cubic foot and are normally charged to 2200 psi. A larger cylinder of 1.66 cubic feet capacity is available on lease. If the pressure regulator for the backup air supply is set to 90 psi the number of times a valve could be cycled with one charge of the cylinder is:

$$\text{No. of Cycles} = \frac{2200 - 90}{90 \left(\frac{\text{Volume of valve}}{\text{Volume of cylinder}} \right)} = \frac{2110}{90} \frac{V_c}{V_v}$$

This equation can be rearranged to give the cylinder pressure required to operate the valve through a given number of cycles.

$$\text{Pressure} = \text{No. of Cycles} \times 90 \left(\frac{V_v}{V_c} \right) + 90$$

A study performed by Westinghouse simulated a number of valve and pump configurations and among other things calculated the number of times the power operated relief valves would cycle in ten minutes. The largest number found in this study was 86. Although it does not represent Point Beach setpoints, it is used as a conservative upper boundary.

<u>Configuration</u>	<u>No. of Cycles</u>	<u>Cyl. Press. For 86 Cycles</u>
Present valve (450 cu. in.) Common gas cylinder (1 cu. ft.)	90	2106 psi
Valve with spacer (150 cu. in.) Common gas cylinder (1 cu. ft.)	270	762 psi
Present valve (450 cu. in.) Larger gas cylinder (1.66 cu. ft.)	149	1304 psi
Valve with spacer (150 cu. in.) Larger gas cylinder (1.66 cu. ft.)	448	495 psi

Use of the present valve configuration with the one cubic foot cylinder is not recommended because it does not provide enough reserve for valve testing. Any of the other three options are satisfactory and one of the three will be installed in the plant for each PORV.

PORV Setpoint Determination

The final measure of the acceptability of the proposed pressure mitigating system is the determination of a PORV setpoint that provides reasonable assurance that the plant specific Appendix G curve will not be violated during a pressure transient and still have a setpoint high enough in pressure to provide operating margin when the plant is in the low pressure and temperature mode of operation. In order to determine the acceptability of the pressure mitigating system for the life of the plant, the Appendix G curves applicable for operation at the end of the plant's life must be calculated. The method used in the determination of the end of life curves is described in Section 6 of WCAP-8743 and WCAP-8738 for Units 1 and 2 respectively. These reports were submitted to the NRC under cover of a Sol Burstein to B. Rusche letter dated March 4, 1977. The resulting data points, which include instrumentation errors as noted, for the 32 EFY curves are given in Table A-2 for Units 1 and 2. Using the data of Table A-2 the values are plotted as typical heatup curves for Unit-1 in Figure A-8 and for Unit-2 in Figure A-9.

Mass Input Transient

The mass input transient setpoint determination will follow the methods described in Section 4 of the attached report titled, "Pressure Mitigating Systems Transient Analysis Results". Since all of the transient analyses for the mass input cases were run at 100°F primary system temperature, a pressure limit corresponding to this temperature at end of life conditions must be selected for Units 1 and 2. For purposes of the mass input calculation, a pressure limit of 520 psig. for both units will be used. The desired setpoint for the PORV's is 425 psig. The criteria for showing that the 425 psig. setpoint is acceptable will be to determine the setpoint overshoot (ΔP) and add it to the setpoint, if the sum is less than 520 psig. the setpoint is considered to be acceptable.

The basic equation to use in the determination of setpoint overshoot for the mass input transient is as follows:

$$\Delta P(V, S, Z, X) = \Delta P_{REF}(X) * F_V * F_S * F_Z$$

where $\Delta P(V, S, Z, X)$ = setpoint overshoot, psi
 V = total RCS volume, cu. ft.
 S = relief valve setpoint, psig
 Z = relief valve opening time, seconds
 X = mass input rate, lb/seconds
 $\Delta P_{REF}(X)$ = reference overshoot at mass input rate X, psi
 F_V = RCS volume factor
 F_S = relief valve setpoint factor
 F_Z = relief valve opening time factor

The method described in Section 4 was developed from a reference set of parameters which are as follows:

- X = mass input rate from the reference safety injection pump
- V = 6000 cubic foot primary system volume
- S = relief valve setpoint at 600 psig
- Z = reference 3 second opening valve

From these reference parameters and the results of the various transient analyses the factors F_y , F_s and F_z were developed as described in Section 4.3 of the attached report. The report states that the development of these factors is conservative and plant specific analysis would result in peak values less than the peak values calculated using the algorithm outlined in Section 4. The Point Beach plant specific parameters are the same for both units and have the following values:

- X = mass input rate is shown on Figure A-7 which is identical to the reference SI pump used in the analyses (curve C of Figure 2.3.2). This means that results of the analyses can be used directly because the Point Beach SI pump characteristic was used as the reference SI pump.
- V = 6900 cubic feet for total RCS volume
- S = 425 psig. for relief valve setpoint
- Z = 2 seconds for relief valve open time

The two factors ΔP_{REF} (X) and F_s can be considered to determine the overshoot at a specific setpoint for a given mass input characteristics. Since the Point Beach pump characteristic was the one used in the analyses, the results of the analyses can be used directly. Therefore, the appropriate values for setpoint overshoot are:

<u>Setpoint (psig)</u>	<u>Overshoot (psi)</u>	<u>Reference</u>
600	155	Line 1 page A-2
400	192	Line 1 page A-3

The linear interpolation for a 425 psig. results in:

$$\Delta P_{REF}(X) * F_s = 187.38 \text{ psig.}$$

From Figure 4.2.3 the F_z for a 2 second valve is:

$$F_z = 0.733 \text{ at 2 seconds}$$

From Figure 4.2.2 the F_y for a 6900 cubic foot RCS volume is:

$$F_y = 0.93 \text{ at 6900 cu.ft.}$$

The resulting overshoot is:

$$\Delta P = 187.38 * 0.733 * 0.93$$

$$\Delta P = 127.7 \text{ psi}$$

Adding this to the setpoint results in:

$$P_{MAX} = 425 + 127.7 = 552.7 \text{ psig.}$$

The maximum pressure exceeds the target value of 520 psig and therefore is not considered acceptable. In order to arrive at an acceptable setpoint, either the setpoint has to be lowered or conservatism in the calculations have to be identified and removed to reduce the calculated overshoot. One conservatism that was evaluated and reported in Section 5.2 of the report is the effect of metal expansion. One case of the effect of metal expansion was examined which can be applied to Point Beach and the results are shown in Figure 5.2. The effect on overshoot is related to the ratio of the value in peak pressure when metal expansion is assumed in the analysis to the value without metal expansion. Using the maximum values from Figure 5.2, the ratio is:

$$\frac{\text{Maximum overshoot with metal expansion}}{\text{Maximum overshoot without metal expansion}} = \frac{115}{155} = 0.74$$

Applying this factor to the overshoot calculated above, the maximum pressure becomes:

$$P_{MAX} = 425 + 127.7 * 0.74 = 519.5 \text{ psig.}$$

This value is less than the target value of 520 psig. and the setpoint of 425 psig. for the mass input transient is acceptable.

Heat Input Transient

The heat input transient setpoint determination will follow the methods described in Section 4 of the attached report. The setpoint of 500 psig will be examined because the algorithm provided in Section 4.2.2 uses analyses at that setpoint pressure. The Point Beach plant specific parameters for the heat input transient are the same for both units and have the following values:

Steam generator heat transfer area	= 44,000 sq.ft.
RCS volume	= 6,900 cu.ft.
RCS/SG ΔT	= 50°F
Initial RCS pressure	= 300 psig.
Relief valve setpoint	= 500 psig.

The resulting steam generator heat transfer area fraction, f , equals:

$$f = \frac{44,000}{58,000} = 0.759$$

Table A-3 gives the results of each step in the determination of the heat input transient setpoint. The step numbers correspond to those given on pages 4-16 and 4-20 of the attached report. The resulting setpoint overshoot appears in step 6. The maximum pressure during the transient for the various temperatures is determined and listed as step 7 in Table A-3 as P_{MAX} .

In order to determine if, or to what extent, the 500 psig. setpoint heat input transient violates the Appendix G curves for Point Beach, a table is constructed to compare the P_{MAX} values at various temperatures to the pressure limit at each temperature. This is accomplished in Table A-4 by comparing the results in Table A-3 to the pressure limit values in Table A-2. The maximum violation of the Appendix G pressure limit curve occurs with Unit-2 at a RCS temperature of 177°F by an amount of 25 psi. This means that the setpoint of 500 psig. is not acceptable for the heat input case. Data is not available for lower setpoint determination at 50° ΔT but judgment can be used to show that the heat input transient is less limiting than the mass input transient. For a 5000 cubic foot RCS plant the reduction in P_{MAX} is at least equal to the reduction in setpoint as shown on Figure H37. A similar relationship exists at other initial RCS temperatures. Therefore, one can conclude that if we lower the setpoint to 475 psig. the peak pressure that could occur during a heat input transient pressure will remain below the Appendix G curve for both Point Beach Units. This allowable setpoint is higher than the 425 psig. value determined for the mass input transient and supports the statement that the mass input transient is the limiting transient. Using realistic model assumptions, more precise lower setpoint analysis results at 425 psig. and a faster opening time, the initial ΔT between the RCS and the steam generator that would be allowed and still not exceed the Appendix G limit curves is estimated to be greater than 100°F.

Plant Operation Utilizing the Pressure Mitigating System

The 425 psig. setpoint is shown as a constant value on the heatup curves for Units 1 and 2, Figures A-8 and A-9 respectively. During heatup the mitigating system will be in the ENABLE mode until the temperature at which the inservice pressure test can be performed is achieved. At this time the RCS is separated from the RHR system and pressure test is performed. The pressure test is an allowed violation of the Appendix G curve. Following the pressure test, normal pressurization and heatup would proceed. The likelihood of exceeding the Appendix G curve above the temperature allowed for pressure testing is very small because very little of the curve remains at a pressure that would present a problem. The safety injection pumps need not be considered because their shutoff head is slightly less than 1600 psig and the allowable pressure at this temperature is 1800 psig. The only pumps that can continue to

pressurize the system are the low capacity charging pumps (40 gpm). If this should happen, the event would not cause a problem because it would be similar in nature to the pressurization which is allowed for by the inservice pressure test.

The cooldown is the reverse of heatup, i.e., the plant is depressurized to below 425 psig. and the pressure mitigating system ENABLED before the plant temperature goes below the temperature at which the inservice pressure test can be performed. For the end of plant life curves shown on Figures A-8 and A-9, there is a narrow temperature window between the NPSH curve and the inservice pressure test temperature at 425 psig. Appendix G curves at earlier times in the life of the plant will allow a larger temperature window but the relationship to the allowable pressure will be the same. This mode of operation is achievable at Point Beach and the mitigating system, as well as the 425 psig. setpoint, is considered acceptable for plant operation.

TABLE A-1

FAILURE MODE AND EFFECTS ANALYSIS

Failure Mode	Resultant Effect on Operability			Comment
	PORV 1	PORV 2	RHR Relief	
Failure of channel 1 component	Not Op*	Op*	Op	Note 1
Failure of channel 2 component	Op	Not Op	Op	Note 2
Loss of channel 1 instrument power	Not Op	Op	Op	Note 2
Loss of channel 2 instrument power	Op	Not Op	Op	Note 2
Loss of PORV 1 battery	Not Op	Op	Op	
Loss of PORV 2 battery	Op	Not Op	Op	
Loss of instrument air	Op	Op	Op	Note 3
Loss of off-site power	Op	Op	Op	
Loss of off-site power plus loss of one diesel	Not Op	Op	Op	Note 2
Loss of off-site power plus loss of other diesel	Op	Op	Op	Note 2
Loss of letdown	Op	Op	Op	Note 3
Isolation of RHR	Op	Op	Not Op	Note 3
Closure of PORV 1 isolation valve	Not Op	Op	Op	Note 6
Closure of PORV 2 isolation valve	Op	Not Op	Op	Note 6
Inadvertent SI	Op	Op	Op	Note 3
Manual starting of SIS pump	Op	Op	Op	Note 3
Manual starting of additional charging pump	Op	Op	Op	Note 3
Manual starting RCP	Op	Op	Op	Note 3

Failure Mode	Resultant Effect on Operability			Comment
	PORV 1	PORV 2	RHR Relief	
Failure to activate low pressure setpoint channel 1	Not Op	Op	Op	Note 4, 6
Failure to activate low pressure setpoint channel 2	Op	Not Op	Op	Note 4, 6
Problems with operation of pressurizer relief tank	Op	Op	Op	
Undefined common mode failure of channels 1 and 2	Not Op	Not Op	Op	Note 5

NOTES

*Not Op denotes that this valve is not operable and not capable of performing its pressure relieving function.

* Op denotes that this valve is operable and can perform its pressure relieving function.

Note 1 Channel component consists of one of the following: sensor, channel power supply, bistable to valve, or PORV.

Note 2 For the purpose of this analysis channel 1 power is supplied from a buss which gets power from one of the emergency diesels and channel 2 power is supplied from an inverter on a station battery.

Note 3 For all possible causes of overpressurization, no failure of PORV 1 or 2 results.

Note 4 Alarm will alert the operator to this condition.

Note 5 Common mode failure of channels 1 and 2 is considered incredible at the time of an overpressurization event, but this shows that pressure relief is still available via the RHR relief valves.

Note 6 ENABLE status light for associated PORV will not be lighted.

TABLE A-2

APPENDIX G Reactor Vessel Pressure Limit Curves
32 Effective Full Power Years

Unit-1 Steady State (0°F/hr)

<u>Temp.*</u>	<u>Pressure*</u>	<u>Temp.*</u>	<u>Pressure*</u>
81	510	271	782
91	513	291	880
121	525	306	973
151	543	336	1233
181	571	356	1480
211	614	386	2005
241	680	401	2359

Unit-2 Steady State (0°F/hr)

<u>Temp.*</u>	<u>Pressure*</u>	<u>Temp.*</u>	<u>Pressure*</u>
77	504	297	779
117	513	317	875
147	524	332	968
177	542	362	1225
207	570	382	1469
237	612	412	1988
267	678	427	2338

* Instrumentation errors of 10°F and 30 psi are included in the calculated values.

TABLE A-3

Heat Input Setpoint Determination

Step	Reactor Coolant System Initial Temperature			
	100°F	140°F	180°F	250°F
1	Figure (4.2.6) $\Delta P_{6K} = 31 \text{ PSI}$ $\Delta P_{13K} = 27 \text{ PSI}$	Figure (4.2.6) $\Delta P_{6K} = 62 \text{ PSI}$ $\Delta P_{13K} = 49 \text{ PSI}$	Figure (4.2.6) $\Delta P_{6K} = 98 \text{ PSI}$ $\Delta P_{13K} = 68 \text{ PSI}$	Figure (4.2.6) $\Delta P_{6K} = 156 \text{ PSI}$ $\Delta P_{13K} = 117 \text{ PSI}$
2	Figure (4.2.7) $\Delta P = 31 \text{ PSI}$ $UA_{6K} = .083$ Figure (4.2.8) $\Delta P = 27 \text{ PSI}$ $UA_{13K} = 0.13$	Figure (4.2.7) $\Delta P = 62 \text{ PSI}$ $UA_{6K} = .098$ Figure (4.2.8) $\Delta P = 49 \text{ PSI}$ $UA_{13K} = .16$	Figure (4.2.7) $\Delta P = 98 \text{ PSI}$ $UA_{6K} = .185$ Figure (4.2.8) $\Delta P = 68 \text{ PSI}$ $UA_{13K} = .185$	Figure (4.2.7) $\Delta P = 156 \text{ PSI}$ $UA_{6K} = .22$ Figure (4.2.8) $\Delta P = 117 \text{ PSI}$ $UA_{13K} = .22$
3	$F = \frac{44,000}{58,000} = .759$	$F = .759$	$F = .759$	$F = .759$
4	$UA'_{6K} = (.083)(.759)$ $= .063$ $UA'_{13K} = (0.13)(.759)$ $= .099$	$UA'_{6K} = (.098)(.759)$ $= .074$ $UA'_{13K} = (.16)(.759)$ $= .121$	$UA'_{6K} = (.115)(.759)$ $= .087$ $UA'_{13K} = (.185)(.759)$ $= .140$	$UA'_{6K} = (.138)(.759)$ $= .105$ $UA'_{13K} = (.22)(.759)$ $= .167$
5	Figure (4.2.7) $UA_{6K} = .063$ $\Delta P'_{6K} = 21 \text{ PSI}$ Figure (4.2.8) $UA_{13K} = .099$ $\Delta P'_{13K} = 20 \text{ PSI}$	Figure (4.2.7) $UA_{6K} = .074$ $\Delta P'_{6K} = 44 \text{ PSI}$ Figure (4.2.8) $UA_{13K} = .121$ $\Delta P'_{13K} = 37 \text{ PSI}$	Figure (4.2.7) $UA_{6K} = .087$ $\Delta P'_{6K} = 72 \text{ PSI}$ Figure (4.2.8) $UA_{13K} = .140$ $\Delta P'_{13K} = 53 \text{ PSI}$	Figure (4.2.7) $UA_{6K} = .105$ $\Delta P'_{6K} = 118 \text{ PSI}$ Figure (4.2.8) $UA_{13K} = .167$ $\Delta P'_{13K} = 89 \text{ PSI}$

TABLE A-3 (CONTINUED)

Step	Reactor Coolant System Initial Temperature			
	100°F	140°F	180°F	250°F
6	$V_{RCS} = 6900 \text{ ft}^3$ $\Delta P'_{6K} = 21 \text{ PSI}$ $\Delta P'_{13K} = 20 \text{ PSI}$ $\Delta P'_{VRCS} =$ $21 - \frac{(6900-6000)(21-20)}{7000}$ $= 20.87 \text{ PSI}$	$V_{RCS} = 6900 \text{ ft}^3$ $\Delta P'_{6K} = 44 \text{ PSI}$ $\Delta P'_{13K} = 37 \text{ PSI}$ $\Delta P'_{VRCS} =$ $44 - \frac{(6900-6000)(44-37)}{7000}$ $= 43.1 \text{ PSI}$	$V_{RCS} = 6900 \text{ ft}^3$ $\Delta P'_{6K} = 72 \text{ PSI}$ $\Delta P'_{13K} = 53 \text{ PSI}$ $\Delta P'_{VRCS} =$ $72 - \frac{(6900-6000)(72-53)}{7000}$ $= 69.56 \text{ PSI}$	$V_{RCS} = 6900 \text{ ft}^3$ $\Delta P'_{6K} = 118 \text{ PSI}$ $\Delta P'_{13K} = 89 \text{ PSI}$ $\Delta P'_{VRCS} =$ $118 - \frac{(6900-6000)(118-89)}{7000}$ $= 114.3 \text{ PSI}$
7	$P_{MAX} = P_{set} + \Delta P'_{VRCS}$ $= 500 + 20.87$ $P_{MAX} = 520.87 \text{ psig}$	$P_{MAX} = P_{set} + \Delta P'_{VRCS}$ $= 500 + 43.1$ $P_{MAX} = 543.1 \text{ psig}$	$P_{MAX} = P_{set} + \Delta P'_{VRCS}$ $= 500 + 69.56$ $P_{MAX} = 569.56 \text{ psig}$	$P_{MAX} = P_{set} + \Delta P'_{VRCS}$ $= 500 + 114.3$ $P_{MAX} = 614.3 \text{ psig}$

TABLE A-4

Heat Input Transient Violation of Appendix G Curves

POINT BEACH UNIT 1

Temperature Degrees F	Appendix G Pressure Limit PSIG	Heat Input Pressure (MAX) PSIG	Temperature Difference PSIG
81	510	510*	0
91	513	516*	3
100	517	521	4
121	525	533*	8
140	536*	543	7
151	543	550*	7
181	571	570	-1
211	614	590*	-24
241	680	609*	-71
250	710*	615	-95
271	782	635*	-147

POINT BEACH UNIT 2

77	504	509*	5
100	509*	521	12
117	513	530*	17
140	521*	543	22
147	524	548*	24
177	542	567*	25
181	546*	570	24
207	570	587*	17
237	612	607*	-5
250	641*	615	-26
267	678	625*	-53

* interpolated data

○ maximum ΔP for respective unit

**FIGURE A-1
LOGIC DIAGRAM
OVERPRESSURIZATION MITIGATING SYSTEM**

1 and 2 denote sharing of annunciator
s indicates status light

**FIGURE A-2
WIDE RANGE PRESSURE SIGNAL**

FIGURE A-3

TYPICAL POWER OPERATED RELIEF VALVE CIRCUIT

FIGURE A-4
BACKUP GAS/AIR SUPPLY

(ENABLE STATUS LIGHT CIRCUIT)

The indicator for the keyswitch for valve 430 is interlocked through the indication circuit for valve 516 in order to provide additional assurance to the operator that valve 516 is open when the overpressurization circuit is activated. In the same manner the keyswitch indicator for valve 431C is interlocked through the indication circuit for valve 515.

Motor operated valve 516 is in series with and is an isolation valve for power operated relief valve 430. In the same manner valve 515 is the isolation valve for valve 431C.

The graph shows a linear relationship between Capacity (GPM) and Set Pressure (PSIG). The Y-axis (Capacity - GPM) ranges from 850 to 1100 in increments of 50. The X-axis (Set Pressure - PSIG) ranges from 350 to 600 in increments of 50. A single line represents the data for a 1 1/2 inch diameter, 100 feet long, 1 inch thick wall pipe.

Set Pressure - PSIG	Capacity - GPM
380	875
400	890
450	940
500	990
550	1040
600	1090

20 X 20 TO THE INCH • 7 X 10 INCHES
KEUFFEL & ESSER CO. MANHATTAN

W
I

46 1240

Figure A-7
Point Beach Nuclear Plant
Safety Injection System
Delivery Characteristics
(Single Pump)

Figure A-8
Point Beach Nuclear Plant Unit No. 1
Heatup Limitations Applicable To
32 Effective Full Power Years
(10°F and 30 psi Instrument Error Included)

Reactor Coolant System Pressure, psig

2400
2200
2000
1800
1600
1400
1200
1000
800
600
400
200

Temperature °F

Wisconsin Electric POWER COMPANY
231 WEST MICHIGAN, MILWAUKEE, WISCONSIN 53201

October 28, 1977

Mr. Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation
U. S. NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555

Attention: Mr. George Lear, Chief
Operating Reactors Branch 3

Gentlemen:

DOCKETS 50-266 AND 50-301
ADDITIONAL INFORMATION
OVER-PRESSURIZATION PROTECTION
POINT BEACH NUCLEAR PLANT
UNITS 1 AND 2

Our letter dated July 28, 1977 provided final details of the proposed pressure mitigating system for Point Beach Nuclear Plant Units 1 and 2 and forwarded four copies of the report, "Pressure Mitigating Systems Transient Analysis Results". Attachment A to that letter presented the details of the pressure mitigating system at Point Beach and the results of the setpoint determination calculations.

The purpose of this letter is to forward a supplement to the "Pressure Mitigating Systems Transient Analysis Results" report which has been prepared by the Westinghouse Electric Corporation. This supplement provides additional Loftran heat input results, and provides further instructions and guidance for using this supplemental information to determine pressure overshoots for the heat input transient. As you are aware, the heat input transients presented in Attachment A to the July 28, 1977 letter utilized a relief setpoint of 500 psig because the algorithm provided in Section 4.2.2 of the original report used analyses at that setpoint. The information provided in the enclosed supplement provides information at other relief setpoints. By use of interpolation, plant specific parameters may be applied to the heat input transient and a more accurate resulting overshoot pressure determined.

773040032

October 28, 1977

Attachment 1 to this letter provides the results of our revised heat input transient overshoot analyses. This determination supersedes the information presented in Tables A-3 and A-4 of Attachment A to our July 28, 1977 letter. The transient pressure overshoots for the heat input condition at the Point Beach over-pressure relief setpoint of 425 psig are well under the allowable Appendix G pressure limit. This is as predicted in our July 28, 1977 letter.

You will also find as part of Attachment 1 certain additional information regarding the specific over-pressure mitigating installation at the Point Beach Nuclear Plant. This information includes revised Figures A-1 and A-4 from our July 28 submittal. These figures have been modified to show the circuits and piping as they are being installed at Point Beach. Additionally, we have specified the gas cylinder size to be used at Point Beach, and have provided an elaboration on the determination of metal expansion effects when considering coolant pressure transients.

We are providing four copies of both the supplement to the July 1977 report and Attachment 1 to this letter. Installation of the over-pressurization mitigating system at Point Beach Nuclear Plant Unit 1 is presently in progress. The installation on Unit 2 is still scheduled for the spring 1978 refueling.

Very truly yours,

Executive Vice President

Sol Burstein

Enclosures

ATTACHMENT 1

I. Heat Input Transient

The information provided in the Supplement to the July 1977 report (the supplement) has been utilized to determine the setpoint overshoot for the heat input transient given the following Point Beach Nuclear Plant specific parameters:

Steam generator heat transfer area	=	44,000 sq. ft.
RCS volume	=	6,900 cu. ft.
RCS/SG ΔT	=	50°F
Initial RCS pressure	=	300 psig
Relief valve setpoint	=	425 psig
Relief valve opening time	=	2 seconds

The results of this determination are presented in Table 1-1 which compares the maximum pressure overshoot values at various initial RCS temperatures to the end of life Appendix G pressure limit at those temperatures. Maximum pressure overshoot values were obtained at 100°F and 250°F using the information found on Figures 14 to 17 and Figures 20 to 23 of the supplement and the linear interpolation process described in section C of the supplement. A sample calculation for the 250°F initial RCS temperature is shown in Table 1-2 to clarify the technique used.

II. Overpressurization Mitigating System Description

- A. Attached figures 1-1 and 1-2 are revisions of figure A-1 and A-4 as found in Attachment A to our July 28, 1975 letter. These revised figures show the overpressurization mitigating system logic diagram and backup gas/air supply as they are being installed at the Point Beach Nuclear Plant. We also wish to advise that the size of the gas cylinder to be used as part of the Point Beach installation is 1.55 cubic feet. Using the present valves without a valve spacer, this provides a conservative allowance of 139 valve cycles on one cylinder charge.
- B. The analyses presented in the July 1977 report and the enclosed September 1977 supplement were conducted such that certain parameters provided a degree of conservatism in the peak pressure reached during a transient. By selecting more realistic values of these parameters, the peak pressures would be reduced. Section 5 of the July report described five of these particular parameters.

One of these items, the effects of metal expansion, is discussed in Section 5.2. Figure 5.2 presents the results of material expansion in reducing pressure overshoot. The curves of this figure were produced using the Loftran program with and without a safety injection flow rate correction factor based on metal expansion. The safety injection flow rate factor was based on computing the increase in reactor coolant system volume as a result of increased reactor coolant pressure and ratioing the actual safety injection flow at the system pressure to the flow necessary to achieve the same pressure rise when including the flow needed to fill and pressurize the expansion volume.

The information presented in Table 5.2 is the result of hand calculations on the reference 4-loop and 2-loop plants to determine the increase in volume for 1000 psi pressure change. In this simplified calculation only radial and axial expansion in cylindrical and hemispherical sections were considered. These included straight runs of piping, the steam generator tubes, the pressurizer, and the reactor vessel below the nozzle course. No expansion volumes were calculated for more complicated geometries nor was any credit taken for compression of the zircalloy fuel pins.

In the specific case graphed in Figure 5.2, a transient from 500 to 600 psi was assumed. For this 100 psi increase, the calculation described above resulted in an increase in reactor coolant system volume of 2.25 gallons. For this pressure (assuming 0.75 seconds to increase the pressure to 600 psi) and based on the safety injection pump flow from Figure M35 of 850 gpm, the flow into the system would be 10.6 gallons. However, to get a 100 psi increase in 0.75 seconds when including the effects of metal expansion, $10.6 + 2.25 = 12.85$ gallons flow into the reactor coolant system would be necessary. The ratio of actual (10.6 gallons) flow to required (12.85 gallons) flow, or 0.825, is then applied as a correction factor to the safety injection flow rate in the Loftran code.

The application of this factor in the Loftran code demonstrates that the pressure overshoot above setpoint calculated for the inelastic case is at least 35% higher than the realistic pressure overshoot for the actual elastic system. In the Point Beach setpoint determination for the mass input case, this factor, actually 0.74 based on the ratio of peak pressures from Figure 5.2, was applied to the peak pressure to confirm the acceptability of the 425 psig relief valve setpoint. Although application of this correction eliminates one of the conservatisms assumed in the analysis, it is apparent that the application of this correction has been, in turn, conservative; since the simplistic volume expansion calculation did not take credit for all the possible elastic expansion nor the fuel pin compression in the actual reactor coolant system.

It must also be pointed out again that in addition to the two power-operated relief valves, there is also the diverse relief capability of the RHR system relief valve. Thus triple protection against the overpressurization situation is provided.

TABLE 1-1

POINT BEACH UNIT 1

<u>Initial RCS Temperature °F</u>	<u>Appendix G Pressure Limit psig</u>	<u>Heat Input Transient Max. Pressure Overshoot psig</u>
100	517	441
140	536	454
180	571	465
250	700	490

POINT BEACH UNIT 2

100	509	441
140	521	454
180	546	465
250	641	490

TABLE 1-2

HEAT INPUT OVERSHOOT DETERMINATION
SAMPLE CALCULATION
REACTOR COOLANT SYSTEM INITIAL TEMPERATURE 250°F

1. From Figure 16 $UA_{6K} = 0.138$ at 250°F
From Figure 17 $UA_{13K} = 0.223$ at 250°F
2. Multiply by $\frac{44,000 \text{ sq. ft.}}{58,000 \text{ sq. ft.}} = 0.759$ to normalize for the Point Beach
Steam Generators

then $UA_{6K} \times 0.759 = 0.105$
 $UA_{13K} \times 0.759 = 0.169$
3. Enter Figure 16 with $UA_{6K} = 0.105$ gives $P_{MAX} - P_{SETPOINT} = \Delta P_{6K} = 118$ psi
Enter Figure 17 with $UA_{13K} = 0.169$ gives $\Delta P_{13K} = 90$ psi
4.
$$\Delta P_{6.9K} = \Delta P_{6K} - \frac{6900 - 6000}{13000 - 6000} (\Delta P_{6K} - P_{13K})$$
$$= 118 - \frac{900}{7000} (118 - 90)$$

 $\Delta P_{6.9K} = 114.4$ psi for 500 psig setpoint and 3 sec. opening time
5. Using techniques identical to steps 1-4 but substituting Figures 14 and 15
$$\Delta P_{6.9K} = 92 - \frac{900}{7000} (92 - 71)$$

 $\Delta P_{6.9K} = 89.3$ psi for 400 psig setpoint and 3 sec. opening time
6. Interpolating the results from steps 4 and 5
$$\Delta P_{425 \text{ psi}} = \Delta P_{500} - \frac{500 - 425}{500 - 400} (\Delta P_{500} - \Delta P_{400})$$
$$= 114.4 - \frac{75}{100} (114.4 - 89.3)$$

 $\Delta P_{425} = 95.6$ psi for 3 sec. opening time
7. Steps 1-6 are then repeated using Figures 22 and 23 and then 20 and 21 to
obtain a ΔP_{425} at 1.5 sec. opening time.

 $\Delta P_{425} = 48.8$ psi at 1.5 sec. opening time

TABLE 1-2 (Continued)

8. As a final step, the results of steps 6 and 7 may be interpolated to obtain ΔP for 425 psig setpoint, 2 sec. opening time, 6900 cu. ft. RCS volume and 44,000 sq. ft. s.g. area.

$$\begin{aligned}\Delta P_{2 \text{ sec.}} &= \Delta P_{425/3} - \left(\frac{3.0 - 2.0}{3.0 - 1.5} \right) (\Delta P_{425/3} - \Delta P_{425/1.5}) \\ &= 95.6 - \left(\frac{1}{1.5} \right) (95.6 - 48.8)\end{aligned}$$

$$\Delta P = 64.4 \text{ psi}$$

**FIGURE 1-1
LOGIC DIAGRAM
OVERPRESSURIZATION MITIGATING SYSTEM**

Revised October 1977

FIGURE 1-2
OVERPRESSURIZATION MITIGATING SYSTEM
BACKUP GAS/AIR SUPPLY

Revised October 1977

Wisconsin Electric POWER COMPANY

231 WEST MICHIGAN, MILWAUKEE, WISCONSIN 53201

Regulatory

File Cys

Reference 3

September 3, 1976

Mr. Benard C. Rusche, Director
Office of Nuclear Reactor Regulation
U. S. NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555

Attention: Mr. George Lear, Chief
Operating Reactors Branch 3

Dear Mr. Rusche:

DOCKETS 50-266 AND 50-307
REACTOR VESSEL OVERPRESSURIZATION
UNITS 1 AND 2, POINT BEACH NUCLEAR PLANT

Wisconsin Electric Power Company and Wisconsin Michigan Power Company were requested by your letter dated August 11, 1976 to evaluate system designs to determine the susceptibility to overpressurization events and to notify you with 20 days that we will provide all the information requested within 60 days or explain why we cannot meet this schedule and provide the schedule that will be met.

This letter is to notify you that a task group of utilities with Westinghouse plants has been formed, of which Wisconsin Electric Power Company and Wisconsin Michigan Power Company are participants, to examine the complexity of the overpressurization events and to identify similarities between Westinghouse plants for the purpose of determining a consistent solution to this issue. The task group will meet with Westinghouse for the purposes of establishing plant groupings and to evaluate the means by which the likelihood of additional occurrences of reactor vessel overpressurization events can be minimized.

Each utility will examine its plant's operating procedures and administrative controls during the 60-day period referenced in your August 11, 1976 letter. The purpose of this examination is to evaluate procedural changes that will minimize the likelihood of overpressurization events. As an interim measure, the plant operating personnel are being informed of the potential for overpressurization and are being cautioned to minimize conditions which could cause such an event.

9113

September 3, 1976

The results of the task group meetings will be reported to the NRC by each utility at the end of the 60 day period. Recommendations and schedules will be given at that time as well as the results of the plant procedure review and resulting procedure revisions, if appropriate.

Should you have any questions concerning this program or the preliminary schedule described herein, please contact us.

Very truly yours,

A handwritten signature in dark ink, appearing to read "Sol Burstein", with a long horizontal flourish extending to the right.

Executive Vice President

Sol Burstein

Wisconsin Electric POWER COMPANY
231 WEST MICHIGAN, MILWAUKEE, WISCONSIN 53201

Reference 4

REGULATORY DOCKET FILE COPY

Mr. Benard C. Rusche, Director
Office of Nuclear Reactor Regulation
U. S. NUCLEAR REGULATORY COMMISSION
Washington, D.C. 20555

Attention: Mr. George Lear, Chief
Operating Reactors Branch #3

Dear Mr. Rusche:

DOCKETS 50-266 AND 50-301
REACTOR VESSEL OVERPRESSURIZATION
UNITS 1 AND 2, POINT BEACH NUCLEAR PLANT

Wisconsin Electric Power Company and Wisconsin Michigan Power Company were requested by your letter of August 11, 1976 to evaluate system designs to determine susceptibility to overpressurization events, analyze the possible events and propose both interim and permanent modifications to systems and procedures to reduce the likelihood and consequences of such events. In our letter of September 3, 1976, we advised the Commission that a task group of utilities owning Westinghouse pressurized water reactor plants had been formed to evaluate this problem and that we would review our operating procedures to minimize the likelihood of overpressurization events. We also stated that at the end of the 60-day period noted in your letter of August 11, a progress report would be provided. This letter provides that report.

A meeting was held by the utility group on September 23, 1976, to review and discuss actions performed by the utilities and Westinghouse. The major items reviewed and the conclusions of that meeting are summarized herein.

October 14, 1976

- A. The overpressurization events which have occurred on Westinghouse plants were discussed and the cause of each of the events was noted. In addition, the effectiveness of assumed mitigating systems, such as a relief valve, was considered. The review of these occurrences indicated that single equipment failure or operator error caused each event and some form of pressure relief would reduce the consequences of such events.
- B. The groupings of various plants was considered as a means to reduce the amount of analysis necessary to evaluate the effectiveness of power operated relief valves to preclude overpressurization. The review of parameters which would affect analysis results indicated that plant grouping was not necessary because Westinghouse plants are sufficiently similar to envelop the plants by use of a bounding analysis.
- C. The pressurizer power operated relief valves were found to have significant water relief capability and relatively rapid response time, i.e., approximately 2-second opening time.
- D. The preliminary evaluation of the pressure transients attributable to mass injection from possible dynamic sources indicates that the pressurizer power operated relief valves are of the proper mass flow characteristics to limit the pressure surges resulting from such events.
- E. The preliminary evaluation of pressure transients attributable to component temperature difference following a reactor coolant pump start indicates that the pressurizer power operated relief valve may be capable of mitigating the pressure surge of such an event. The pressure transient resulting from equipment temperature difference in conjunction with a pump start will require a detailed transient analysis to assure acceptability of the system modification selected.
- F. Due to the preliminary indications that the pressurizer power operated relief valves may be capable of providing overpressure protection during solid system operation, the overpressurization transients will be analyzed assuming a mitigating system employing the pressurizer power operated relief valves.
- G. A working subcommittee was formed to identify the possible overpressurization events to be evaluated and to define the conditions and parameters to be included in the transient analysis. This subcommittee was directed to meet with Westinghouse and reach concurrence as to the number of events and conditions for each event to be analyzed. This meeting was held on September 28, 1976.

October 14, 1976

- H. A second working subcommittee was also formed to review operating parameters, such as chemistry requirements and temperature difference limits which affect implementing procedures, to minimize solid system operation by use of a steam bubble at low reactor system average temperatures. This second working subcommittee was directed also to consider an action plan should Appendix G limits be exceeded. This committee met on September 29, 1976.

The meetings which have been held by the utilities and the working sub-committees have resulted in agreement as to a decisive course of action to resolve this overpressurization problem. This course of action includes transient analysis. The transient analysis will include consideration of mass input induced overpressurization and heat input induced overpressurization. The range of system and component physical parameters, performance characteristics and operating limits applicable to Westinghouse designed plants will be used to bound the analysis. Conservative assumptions will be employed to characterize the relief valve performance.

The single failure criteria presented in your letter of August 11, 1976, will be applied. That is, no single event, whether equipment failure or operator error, will result in Appendix G limitations being exceeded. If the overpressurization transient is caused by an equipment failure or operator error, that failure or error will be considered the single failure event and subsequent actions resulting from the failure or error which could reduce the effectiveness of the mitigating system will be considered and included in the analysis.

The preliminary evaluations performed indicate that protection of a solid reactor coolant system by use of a relief system from an inadvertently opened safety injection accumulator, which is charged to its design pressure, is not practical. It is our opinion that administrative controls similar to the controls employed for assuring that the accumulator valves are open during normal operation, which satisfy the single failure criteria, will provide adequate protection against overpressurization from inadvertent operation of an accumulator during solid system operation. We are revising our procedures to incorporate such controls.

It should be noted that, with the administrative controls and "single failure" criteria applied, 100% assurance of remaining within Appendix G limits still cannot be provided. However, when the modifications are installed, the consequences of an overpressurization transient will be significantly less severe. Since the remote possibility of exceeding Appendix G limits by a small amount will still exist following installation of any mitigating system, the action plan following such an event should also be presented. It is our position that following an overpressurization transient where Appendix G limits are exceeded, an analysis

October 14, 1976

of the event is required to determine the long term consequences of the event and the impact upon plant safety before plant operation is resumed. Such an analysis would include the detailed information applicable to the specific reactor vessel at the fluence accumulated at the time of the event. The analysis would be similar to the analysis presented by Virginia Electric Power Company in Abnormal Occurrence Report AO-SI-73-01-10 dated February 13, 1973, following the overpressurization due to opening an accumulator isolation valve with a solid reactor coolant system. Following completion of such analysis and assurance that continued operation would not compromise health and safety of the public, we would proceed with normal operations and provide a detailed report of the incident and analysis to the Commission.

The analysis of the overpressurization transients will be a major activity with an estimated duration of six months. Following completion of the analysis, modification to the Point Beach Nuclear Plant will be initiated consistent with the analysis results. The schedule for the modification activity will be provided following completion of the analysis and detailed identification of the modification performance requirements.

We have reviewed our operating procedures to determine if changes to the procedures are necessary to conform to the objective of limiting solid system operation to the minimum time practical and to increase the awareness of the operators to plant conditions that have the potential of initiating overpressurization transients. The existing plant equipment has been examined for the purpose of providing relief capability during an overpressurization event and a modification utilizing the pressurizer power operated relief valves is planned to be implemented during each unit's next refueling. Attachment 1 includes a description of the procedure review that has been performed, the instrumentation installed at the Point Beach Nuclear Plant to provide a record of both pressure and temperature in the event of an overpressurization transient, and a description of the modification utilizing the pressurizer power operated relief valves.

Should you or your staff have any questions concerning the material presented herein, please contact me.

Very truly yours,

Executive Vice President

Sol Burstein

Attachment

ATTACHMENT 1

I. Procedure Review

The below listed operating procedures were reviewed to determine if changes could be made to promote the objectives of limiting solid system operating time and to increase operator awareness of plant conditions and evolutions which have the potential of initiating overpressurization transients. A summary of the changes made to the procedures as a result of this review is listed below.

1. Cold Shutdown to Low Power Operation

- a. The operator has been directed to start the reactor coolant pumps near the minimum allowable pressure for pump operation.
- b. During the primary system leak test the reactor coolant pump is run as long as possible. The only reason for securing the pump is if the temperature rise is too great.
- c. An operator caution note has been added to check the letdown path via the letdown orifices prior to securing the residual heat removal isolation valves.
- d. A step is included to return the accumulator isolation valves from the locked closed to the locked open position when reactor coolant system pressure is greater than 1000 psig.

2. Hot Shutdown to Cold Shutdown

- a. Precautions were added to keep a reactor coolant pump operating when on residual heat removal (RHR) with a solid reactor coolant system and little decay heat. A reactor coolant pump should not be restarted under these conditions unless absolutely necessary. If a pump start is necessary, operator is now directed to open all letdown orifice isolation valves and maximize letdown from the RHR system.
- b. Procedure includes a special caution to red tag shut the accumulator isolation valves and red tag off the safety injection pumps prior to cooling down below 300°F or going solid.
- c. Operator is directed to have RHR on line before taking the plant solid.

3. Reactor Coolant Pump Operation

- a. Precautions added as in 2a. above.

II. Plant Instrumentation

The list below identifies the instrumentation that is available and operation during shutdowns to record both pressure and temperature in the event of an overpressurization transient. The following recorders are located on the main control board in the control room.

1. Wide Range Reactor Coolant Pressure Recorder

Range is 0-3000 psig.
Paper recording width is 4 inches.
Paper speed is 3/4 inch per hour.

2. Cold Leg Temperature Recorder

Range is 50°F-650°F.
Paper recording width is 4 inches.
Paper speed is 3/4 inch per hour.
The recorder has two pens one records loop A temperature and the other loop B temperature.

3. Residual Heat Removal Temperature Recorder

Range is 100°F-400°F.
Paper recording width is 4 inches.
Paper speed is 3/4 inch per hour.
The recorder has two pens one records inlet temperature and the other outlet temperature

III. Modification Description

A dual output channel and a separate single output channel will be added to the Wide Range Reactor Coolant Pressure Indication Loop. One channel enables an alarm in the control, the second channel causes one power operated relief valve to open and the third channel causes the second power operated relief valve to open. The three bistable setpoints are independently adjustable. The entire circuit is keyswitched from the main control board. It is anticipated that the setpoints for opening the relief valves will be approximately 25 psig apart and that the alarm setpoint will be lower than the opening setpoint of either valve. The operator will activate the system before the plant is taken solid when the plant is being shutdown and he will deactivate the system after a steam bubble has been drawn prior to plant heatup.

Wisconsin Electric POWER COMPANY
231 WEST MICHIGAN, MILWAUKEE, WISCONSIN 53201

March 2, 1977

Mr. Benard C. Rusche, Director
Office of Nuclear Reactor Regulation
U. S. NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555

Attention: Mr. George Lear, Chief
Operating Reactors Branch #3

Dear Mr. Rusche:

DOCKET NOS. 50-266 AND 50-301
ADDITIONAL INFORMATION - OVERPRESSURIZATION PROTECTION
POINT BEACH NUCLEAR PLANT UNIT NOS. 1 AND 2

This letter is in reply to your two letters of January 11, 1977 and February 9, 1977 regarding the subject of overpressure protection of the reactor vessels at PWR facilities. The first of these letters requested that we commit to one of two overpressure protection implementation options and provide responses to certain staff positions and requests for additional information. The second letter identified specific design features of the "Reference Mitigating System", described in our submittal of December 20, 1976, which did not meet your design criteria regarding long term modifications for overpressurization protection. The letter requested our response to those items identified and requested additional information. This letter is in response to your requests.

Regarding your January 11, 1977 letter, we have enclosed herewith as Appendix A our response to your request for additional information. You are also advised that we have selected the second of the two implementation options offered in the letter. That is, we shall commit to the installation of interim hardware improvement before December 31, 1977 and long term improvements during the first scheduled shutdown after December 31, 1977.

2431

March 2, 1977

Our response to your February 9, 1977 letter may be found in Appendix B. That submittal consists of resolution of those specific design features of the Reference Mitigating System which you felt did not meet the acceptance criteria and our reply to the additional information requested. You will note that a number of the latter responses reference certain studies which have not yet been completed. These studies shall be completed, reviewed and forwarded to you by late April or early May 1977.

The interim improvements, which are described more fully in Appendix C, have already been installed and are operational on Point Beach Unit 1. They will be installed on Point Beach Unit 2 during the spring refueling which is scheduled to commence on March 5, 1977. The long term improvements, which are discussed fully in Appendix D, are scheduled to be installed on Point Beach Unit 1 during the fall 1977 refueling and on Point Beach Unit 2 during the spring 1978 refueling. This assumes that all technical issues can be settled and the procurement and receipt of the necessary hardware proceeds with no foreseeable delays.

We regret the delay in responding to your January 11 letter. This was done to allow us to respond to both information requests in the same letter. We trust this has caused you no inconvenience and may actually aid your review.

Very truly yours,

E. R. Halson
Sol Burstein
Executive Vice President

Sol Burstein

APPENDIX A

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION DATED JANUARY 11, 1977

The following are the responses of the Wisconsin Electric Power Company and Wisconsin Michigan Power Company Licensees to the Nuclear Regulatory Commission's request for additional information submitted by letter dated January 11, 1977.

1. The staff considers it essential that all plant operators (i.e., reactor operators, equipment operators, Instrument & Control personnel) be made aware of the details of the pressure transients which have taken place at all PWR facilities. POSITION: Formal discussions should be held with the operator to review the causes of past pressure transients that have occurred at other operating PWR facilities. Your discussions should include the plant conditions at the time, the mitigating action that could have been taken to avoid the event and the steps taken to prevent similar, further occurrences. Plant similarities and distinctions should be identified along with how these relate to plant startup, shutdown, and testing operations. With regard to this position, you are requested to provide the following information:

- a. If you have not already completed the required formal discussion, when will you do so?
- b. How will the discussions be held?

Response: A discussion of operator and plant support staff training with regard to consideration of past occurrences and means to avoid recurrence of past incidents has been provided with Licensees' response to I & E circular 76-07, which was dated February 25, 1977.

A training session has been conducted by the Operation Superintendent for all Operations Group Supervisors concerning the overpressurization occurrences at Point Beach Nuclear Plant. The Point Beach specific events were covered in detail and the general principles of mass input or heat input as the cause for overpressurization when water solid were discussed. In addition, the plant Training Supervisor is presently cooperating with other PWR facilities in organizing a PWR overpressurization occurrence training package. As soon as this package is available, further training will be conducted with all plant operators.

- c. Of the past PWR Appendix G violations that have occurred at PWR facilities and which are described in License Event Reports, identify which are not credible in your plant due to equipment differences. Provide a description of the distinctions.

Response: Since Licensees do not routinely receive License Event Reports (LER) from all PWR facilities and many of the events predated LER's, we are not aware of all Appendix G violations as described in these reports. As discussed in the response to I & E circular 76-07, Licensees do receive some correspondence concerning occurrences at other facilities. This information is routed to cognizant plant personnel. No program exists, nor is one contemplated, to formally discuss the difference and similarities between each of these many event descriptions and the conditions at the Point Beach Nuclear Plant. Such a program would be extremely time consuming and in many cases very difficult because of the lack of detailed and specific information supplied in the abbreviated LER reporting format.

We believe it is more worthwhile to understand the basic principles of the causes for overpressurization events and study their relationship to our plant and procedures. The list of events provided in NUREG-0138 is a good reference for the causes of past events at other plants.

- d. Describe, in detail, how you are reducing the likelihood of the other remaining credible events. Furnish schematics, diagrams or procedural summaries necessary to support the effectiveness and reliability of these measures.

Response: Licensees have previously provided a summary of procedural changes made to reduce the likelihood and consequences of potential Appendix G violations with our October 4, 1976 letter. The response to question 3 in this Appendix provides additional information.

2. The majority of the reported pressure transients events have occurred while the plants were operating in a water solid condition. POSITION: The staff will require that operations during which the plant is maintained in a water solid condition be minimized or if possible eliminated. These operations in which the plant is in a water solid condition must be fully justified. Accordingly, please provide the following information:

- a. Describe the procedures, evolutions or situations that require the plant be maintained in a water solid condition. Also provide reasons why a nitrogen, air or steam bubble cannot be maintained in these situations.

Response: Solid water operation is required in two modes of plant operation; during the plant cooldown just prior to depressurization and draindown, and during the initial phases of recovery from a primary system drained condition.

1. Cooldown Condition (Procedure OP-3C, Hot Shutdown to Cold Shutdown; Step 4.10 and on)

During the cooldown the primary objective is to reach a drained condition in a safe and expeditious manner in order that refueling or shutdown maintenance can proceed.

- a. The use of a steam bubble in place of solid water operation is not feasible as the pressurizer also needs to be cooled down. It is impossible to cool down the pressurizer, maintain a steam bubble and achieve the required overpressure for main coolant pump operation simultaneously.
- b. The use of air or nitrogen is not considered a feasible solution for the following reasons:
 - (1) The pressurizer must be cooled down and in order to cool it, water must be flushed through it. If air or nitrogen is being used to maintain a bubble, the pressurized dissolving of gas will cause a resultant increase in released gaseous activity when the primary system is opened for maintenance. This is not in conformance with the intent of ALARA. The gaseous level in the coolant system must be low at the time of securing reactor coolant pumps as the water in the steam generators will be stagnant from this period on. Depressurization and draindown is commenced directly after securing reactor coolant pumps.

(2) Chemistry Effect:

The use of highly pressurized nitrogen or air in a Point Beach-type pressurizer to avoid solid water operation for such activities as cooldown, heatup or leak testing is an unreviewed safety problem, and can have complicating corrosion consequences. These corrosion consequences may include oxygen effects in the pressurizer interior, the reactor coolant system boundary, the control rod drive stepping action, the reactor coolant seals, and other second order effects from possible crud releases or bursts. The effects of pressurizing nitrogen into solution at modest temperature levels can also be complicating since nitrogen, unlike oxygen, cannot be chemically and almost completely removed from the reactor coolant with accepted techniques. Therefore, residual nitrogen can remain in solution and depending upon subsequent operational steps, this can result in the formation of nitric acid in the presence of nuclear flux and water dissociation. The potential could exist with pressurized nitrogen use to cause a nitric acid incident similar to the one which occurred in a BWR some years ago when nitrogen (as a breakdown product from introduced hydrazine) was introduced into the nuclear system.

2. Recovery Condition (Procedures OP-4A, Filling and Venting the Reactor Coolant System, and OP-1A, Cold Shutdown to Low Power Operation)

The primary objective of filling and venting is to remove all of the gas from the fluid systems so the oxygen in the coolant can be scavenged prior to going above 250°F. This especially includes the pressurizer as the potential for oxygen corrosion is highest in the boiling conditions which exist in the pressurizer.

The initial objectives of the heatup procedure, OP-1A, are to establish proper chemical conditions prior to exceeding 250°F and then to establish the integrity of the various welds and flanges which were opened during the outage. The present limitation on pressurization requires a reactor coolant system temperature of about 350°F prior to performing the leak test. Since the pressurizer flanges and manway are part of the test, the use of a steam bubble makes leak discovery more difficult and introduces a considerable time delay into the leak testing activity. As soon as the leak test is complete and system integrity is proven, a bubble is drawn in the pressurizer.

- b. Include sufficient background or supplementary information such as system diagrams, procedure summaries and descriptions of equipment operation to justify your need for operating the plant in a water solid condition.

Response: Attached to this Appendix are copies of the procedures discussed in items 2a. System schematics, including the reactor coolant system drawing, may be found in the Point Beach Nuclear Plant Unit Nos. 1 and 2 Final Facility Description and Safety Analysis Report.

3. The inadvertent operation of SIS components during cold shutdown conditions has been responsible for a major portion of the overpressure incidents. POSITION: Based on the Licensee submittals, the recent November 3-5, 1976 meetings, and discussions with NSSS vendors, the staff will require the deenergizing of SIS pumps and closure of SI header/discharge valves during cold shutdown operations. Those situations during which this position cannot be met must be described and be fully justified. Accordingly, please provide the following information:

- a. A schematic diagram of the SIS showing the flowpaths into the RCS.
- b. The head-flow characteristics of each of the SIS pumps.
- c. Identify on the schematic diagrams the pumps and the valves to be closed and deenergized.

- d. Your time schedule for implementing these administrative and operating procedural changes to meet this position.
- e. Indicate all circumstances for which the SIS pumps and valves may not be isolated and deenergized and for those situations, describe the manner in which SIS injection would be prevented.
- f. The location of the breakers that will be opened, and the places from which they can be controlled.
- g. Describe the position indication and status signals which will be lost as a result of deenergizing these components.
- h. Describe in detail the administrative procedures which will be used to assure proper equipment alignment and the supervisory personnel responsible for maintaining control.
- i. Paragraph 12b in Attachment 1 to your October 14, 1976 letter to the staff says that the accumulator MOV's will be shut and red tagged. Provide justification for not removing power from these valves since an inadvertent SIAS would cause them to open.

Response: By changes to the Low Power Operation to Cold Shutdown and Cold Shutdown to Low Power Operation procedures, the requirements set forth in Item 3 will be met. This will be accomplished as follows:

1. Prior to reducing temperature below minimum required for full pressurization:
 - a. The high head safety injection pumps will be isolated electrically with the breaker locked in the off position.
 - b. The high head safety injection system discharge motor-operated isolation valves, MOV-B66A3B, will be shut with the breakers for the motor operators locked in the off position.
 - c. The accumulator discharge motor-operated isolation valves will be shut with the breakers for the motor operators locked in the off position. (This should clarify 3.i.)
2. Low pressure relief protection is required by procedure anytime the primary system temperature is below 300°F.
3. ORT #1, Flow Test of the High Head Safety Injection Pumps, is accomplished at each refueling outage. The test is done only when the reactor vessel head is removed and the cavity is partially flooded. Overpressurization cannot occur during this test.

Answers to specific information inquiries are as follows:

- 3.a Normal safety injection lineup (see attached drawings 110E017 Sheets 1 of 3 and 2 of 3 and 110E018 Sheet 1 of 4 and procedure ORT #1)
 - 3.b A general safety injection pump performance characteristic can be found as figure 6.2-7 of the FFDSAR. Specific pump flow characteristics are attached.
 - 3.c Normal lineup (see attached drawing).
 - 3.d N/A.
 - 3.e ORT #1 (reactor vessel head removed)
 - 3.f N/A.
 - 3.g N/A.
 - 3.h See ORT #1 attached
 - 3.i N/A.
4. ORT #3, Loss of Engineered Safeguards AC Simultaneous with Safety Injection. This test is accomplished at each refueling outage. The circumstances of each refueling outage determine the best time to accomplish the test; generally the test is done near the end of the refueling outage after the majority of maintenance is completed. During this test the high head safety injection pumps are started; however the pump discharge isolation valves, MOV-866A&B, are shut. To insure that an overpressurization does not occur, the pump discharge valves, MOV-866A&B, will be checked shut with the motor-operated valve operator breaker locked in the off position.

Answers to specific questions are as follows:

- 3.a Safety injection lined up per CL-ORT #3 attached.
- 3.b See under item 3 immediately above.
- 3.c See ORT #3 attached.
- 3.d Effective immediately
- 3.e ORT #3, MOV-866A&B locked shut.
- 3.f N/A.
- 3.g N/A
- 3.h ORT #3 attached.
- 3.i N/A.

- J. Describe the impact on overall plant operations if you routinely lowered accumulator nitrogen pressure when in a cold shutdown condition.

Response: The reduction of the accumulator gas pressure will vent about 24,320 ft³ of gas per accumulator to the containment atmosphere. This will constitute a low level radiation release however the primary problem will be the atmospheric problems if the particular conditions at the time did not allow the operation of the containment purge system. The venting of about 50,000 ft³ of nitrogen to the containment atmosphere would be unacceptable unless done over a long period of time with proper ventilation.

The problems of repressurizing gas into the accumulator are two in nature. First, the cost and second, the time. The time is estimated at about eight to 16 hours per accumulator or a total time of about one day. The cost would be about \$1200 for the gas each time the two accumulators are vented and refilled.

4. The staff has noted that several Appendix G violations have occurred during component or system tests while in cold and shutdown conditions. In this regard, please address the following questions.

- a. What components or systems that could cause overpressure transients, are routinely tested while in a cold shutdown condition?

Response: It has been the Point Beach philosophy to test safety-related equipment, i.e., safety injection pumps, if possible, during cold shutdown periods. In fact, an interpretation of our Technical Specifications would require a test be made; or if not, a report made when the requirements of the surveillance section of the Technical Specifications cannot be followed. It appears this conflict will be eliminated upon the adoption of the ASME Code Section XI requirement. Since the discharge valves of the safety injection pumps will be locked closed during the period involved, except for special tests, there does not appear to be a potential for overpressurization in this area in the future.

- b. What extra measures are taken to prevent an overpressure event during these tests?

Response: The discharge valves, MOV-866A&B, are verified closed prior to each test.

5. The staff believes that a high pressure alarm used during low RCS temperature operations is an effective means to attract the operator's attention to a transient in progress. POSITION: The staff will require that an alarm be installed as soon as possible. The alarm you intend to utilize is partially described in section III, Attachment 1 to your October 14, 1976 letter. Please provide the following information:

- a. The alarm setpoint, reset pressure and means to adjust the setpoint.
- b. The alarm annunciators and system sensors.
- c. Discuss your means to assure the alarm's availability and operability during all water-solid operations, and to minimize its downtime for all other cold shutdown conditions.

Response a: The high pressure alarm used during low reactor coolant system temperature operations consists of a single output channel sensed off of the Wide Range Reactor Coolant Pressure Indication Loop (PT-420). The alarm bistable is set to alarm at 400 psig. The alarm bistable resets at approximately 390-395 psig. Alarm setpoint adjustments can be made by adjusting the bistable trip point.

Response b: The high pressure at low reactor coolant system temperature alarm annunciation consists of an audible signal and a flashing light located on the alarm panel of C04. (C04 is the primary reactor systems control panel and contains most of the reactor coolant system alarms). The system sensor is the Wide Range Reactor Coolant Pressure Indication Loop (PT-420) which sense reactor coolant system pressure off the loop A hot leg.

Response c: The alarm's availability and operability is assured by administrative controls in the plant's cooldown and heatup procedures. These procedures require that the reactor operator activate the low temperature high pressure relief system and the system alarm prior to cooling the reactor coolant system below 300°F. These circuits are activated by means of a keyswitch on the main control board, C04. The procedure for heatup requires this system to be locked out when primary temperature is greater than 300°F. The alarm's availability and reliability will be maintained by a periodic program of checks, tests, and calibration similar to that done for other instrumentation channels.

6. The RHR (or SCS) is normally connected to the RCS and operating when the plant is in a cold shutdown condition. The inadvertent isolation of the RHR system while water solid has caused a number of overpressure transients, and the RHR safety valve has actually terminated others. The RHR (or SCS) therefore plays an important part in the initiation and possible mitigation of the PWR overpressurizations. Accordingly, we request the following additional information:

- a. RHR (or SCS) design pressure.
- b. A description of the system isolation valves and their arrangement (e.g., number and configuration of valves installed, and pneumatic or motor operated).

- c. Interlocks, interlock setpoints, and alarms associated with each isolation valve.
- d. Nominal stroke time of isolation valves.
- e. The setpoint and capacity of RHR (or SCS) relief and safety valves.
- f. All pressure alarms, setpoints and associated annunciation for the system.

Response a: The design pressure of the Point Beach residual heat removal (RHR) system is 600 psig.

Response b: The RHR isolation valves consist of two 10-inch motor operated stop valves (700 and 701) located in series on the suction and one motor operated 10-inch stop valve (720) followed by a 10-inch swing check valve on the RHR discharge to the reactor coolant system.

Response c: The first valve off the reactor coolant system loop A RHR suction (700) and the discharge valve (720) are interlocked with the Wide Range Reactor Coolant Pressure Indication Loop so that they cannot open unless reactor coolant system pressure is less than 425 psig.

Response d: The first off isolation valve (700) has a nominal stroke time of ten seconds. The remaining motor operated isolation valves (701 and 720) have a nominal stroke time of two minutes.

Response e: The RHR system is protected by three relief valves. Two are located on the RHR suction line and one on the discharge line. The setpoints of these relief valves are set at 600 psig. One of the suction reliefs and the discharge relief are 3/4-inch valves with a rated capacity of 15 gpm each. The other RHR suction relief is a 3-inch valve with a rated capacity 1200 gpm. This larger 3-inch valve was added to the Point Beach units after one of the early pressure transients for the purpose of adding enough relief capability to deal with expected pressure transients to the Residual Heat Removal System.

Response f: There are no pressure alarms associated with the Residual Heat Removal System.

- 7. Several provisions are mentioned in Attachment 1 to your October 14 letter to reduce the likelihood of an overpressure event due to improper RCP operation while shutdown. Please address the following with regard to these measures:
 - a. When the RCS is in a cold condition, what is the Appendix G pressure limit and the minimum pressure for RCP operation? Furnish any curves that help explain these limits.

- b. What measures do you take to prevent RCS overpressure events when water-solid due to the addition of the RCP discharge pressure?
- c. Describe the techniques you use to reduce or eliminate the RCS temperature imbalance should you have to start a RCP in a solid and idle flow system.
- d. Describe the instrumentation used to determine if the RCS is isothermal, and the criteria for defining an isothermal system.

Response a: The Figure 1 of OP-1A, attached, includes the Point Beach Appendix G pressure limits plus it has the required reactor coolant pump pressures.

Responses b, c and d: There are several measures which have been successful in minimizing any pressure transient upon starting a reactor coolant pump in a solid water condition. The additional pressure addition of the reactor coolant pump start is less than 150 psig. This added to the normal starting pressure under these conditions (about 250 psig) is below the Appendix G pressure limits. The administrative steps which are presently taken are per OP-3C, Step 2.17.1 as follows:

"If a pump needs to be started under this condition, the following recovery procedure should be followed:

- (7) a. Minimize the charging flow rate.
- b. Open all letdown orifices, isolation valves, and maximize letdown from residual heat removal just prior to starting a reactor coolant pump.
- c. Reduce pressure to the minimum required to start a pump.
- d. Restart a reactor coolant pump and observe plant pressure."

During the Unit 1 refueling cooldown completed in the fall of 1976, another technique was tried and is now being evaluated for a possible procedure change. Prior to Step (d) in the procedure above, residual heat removal cooling was secured long enough to allow the reactor vessel temperatures, as measured by the core thermocouples, to increase to about 10°F above the estimated steam generator temperature. The resulting pressure transient on starting the reactor coolant pump was less than 15 psig. It is anticipated that this procedure change will be effective for any future cooldowns.

LIST OF ATTACHMENT TO APPENDIX A

The following attachments to Appendix A were provided with the original NRC copy only.

NO.

- 1 OP-3C, Hot Shutdown to Cold Shutdown
- 2 OP-4A, Filling and Venting Reactor Coolant System
- 3 OP-1A, Cold Shutdown to Low Power Operation
- 4 110E017 Sheet 1 of 3, SIS System
- 5 110E017 Sheet 2 of 3, SIS System
- 6 110E018 Sheet 1 of 4, Auxiliary Coolant System
- 7 ORT #1, Flow Test of High Head Safety Injection Pumps
- 8 ORT #3, Loss of Engineered Safeguards AC Simultaneous with Safety Injection Unit 1
- 9 Flow Characteristic Pump 1P15A
- 10 Flow Characteristic Pump 1P15B
- 11 Flow Characteristic Pump 1P10A
- 12 Flow Characteristic Pump 1P10B
- 13 Flow Characteristic Pump 2P15A
- 14 Flow Characteristic Pump 2P15B
- 15 Flow Characteristic Pump 2P10A
- 16 Flow Characteristic Pump 2P10B

APPENDIX B

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION DATED FEBRUARY 9, 1977

Your February 9, 1977 letter contained specific comments on design features of our proposed "Reference Mitigating System" (RMS) submitted to you on December 20, 1976, and eight additional questions which you requested a response to. This Appendix provides a response to each of the items listed in your February 9 letter.

The specific design features of the RMS which you identified as not meeting the acceptance criteria are listed below. The responses listed identify how the "Overpressurization Mitigating System" discussed in Appendix D will alleviate these concerns.

- Paragraph a.: The RMS uses a single pressure transmitter thus not providing a system that can accept a single failure and still function. If no acceptable alternate protection is available, two power operated relief valve systems are required, that are separate and independent throughout the system from pressure sensor to relief valve.
- Response: As described in Appendix D, the "Overpressurization Mitigating System" will provide two pressure transmitters and a system that is redundant from pressure sensor to relief valve. This system will be able to accept a single failure and still function.
- Paragraph b.: The single failure criteria requires a separate air accumulator for each relief valve operator.
- Response: As described in Appendix D, separate air accumulators for each relief valve operator will be provided.
- Paragraph c.: Provide the basis for any deviations from IEEE279 requirements for electrical components.
- Response: The bases for deviations from IEEE279 are provided in Appendix D.
- Paragraph d.: An annunciator must be provided to insure that the keylock switch is in the proper position when shutdown.
- Response: An annunciator will be provided to alert the operator to low pressure operation. This will insure that the keylock switch is in the proper position. Appendix D provides a more complete description of this annunciator.

Paragraph e.: Provide the basis for any deviations from Seismic Category I design requirements for system components.

Response: The seismic considerations of the "Overpressurization Mitigating System" components are described in Appendix D.

Paragraph f.: The accumulators should also be seismically qualified.

Response: The accumulators will be seismically mounted to Category I structures to provide assurance that an air supply will be available.

Paragraph g.: Loss of air supply to the relief valve accumulators should be alarmed during shutdown.

Response: Loss of air supply to the relief valve accumulators will be alarmed at all times by low air pressure alarm on the instrument air system. Normal system pressure is between 100 and 110 psig and the alarm setpoint is 92 psig. Individual alarms on accumulator low pressure will not be provided.

In addition to providing resolution of the design criteria given above, responses to the requested additional information is as follows:

1. Provide schematic piping and instrumentation diagrams of all systems which are utilized during plant shutdown and startup operations, indicate primary and alternate flow paths, fluid and heat sources, pressure and flow controllers, RCS pressure protection systems, ECCS and make up systems.

Response: It is recommended that the Point Beach Nuclear Plant Unit Nos. 1 and 2 Final Facility Description and Safety Analysis Report (FFDSAR) be consulted for piping and instrumentation diagrams of systems used during plant operations. Responses to questions in Appendix A provide additional information.

2. Provide the failure modes and effects analysis of the overpressure protection system for startup, shutdown and testing operations which defines the limiting combination of initiating event and additional single failure or operator error subsequent to initiation of the pressure transient.

Response: A failure modes and effects analysis of the "Overpressurization Mitigating System" is provided in Appendix D.

3. Discuss the basis for determining the most limiting initial conditions for analysis of the pressure transient. Items that must be considered include but should not be limited to: RCS pressure, reactor coolant temperature, pressurizer temperature, pressurizer level and accumulator level.

Response: Initial conditions for the analysis will be provided with the report describing the analysis assumptions, models and results. It is expected that this report can be filed by the end of April or early May.

4. Were approved systems computer codes used to model the overpressure transients as presented in the generic report? If so, were any major modifications required? If the methods utilized have not been previously approved for system transient analyses, provide model descriptions, flow charts, program listing, and sensitivity studies to verify the performance of the codes.

Response: Computer codes used to model the overpressure transients will be described in the generic report which will be submitted by the end of April or early May.

5. Provide the value of the backpressures assumed in the relief valve discharge pipes when calculating relief capacity. Discuss relief capacity sensitivity to this parameter.

Response: Relief valve backpressure considerations will be discussed in the generic report.

6. Provide the degree of relief capacity installed in the RHR system that could provide additional protection in the event of an overpressure transient. Provide the water relief capacity of the valve. Is the RHR system automatically isolated on RCS high pressure? Provide the pressure setpoints for the RHR system relief valve opening and its automatic isolation.

Response: The relief capacity installed in the RHR system is discussed in the response to item 6c of Appendix A. The RHR system is not automatically isolated on reactor coolant system high pressure.

7. Describe the instrumentation and alarms available to enable the operator to detect and quickly terminate an overpressure transient. Describe the indication of pressurizer relief valve or RHR system relief valve operation.

Response: The alarm functions at low pressure are discussed in the answer to Item 5 of Appendix A.

1. The pressurizer power-operated relief valves have open/shut indication on the main control board.

2. The pressurizer safety valves and the power-operated relief valves discharge to the pressurizer relief tank. This tank has level, temperature and pressure indication on the main control board. In addition, these three signals have a common alarm, also on the main control board. Any lifting of a relief or safety valve will be seen in the pressurizer relief tank immediately.
3. The residual heat removal relief valves discharge to the containment sump which is normally isolated. If the level increases, it alarms in the main control room.
8. When power is removed from valve motor operators under administrative control provisions, describe the status lights and indicators available to verify their proper alignment. When administrative controls call for removing power from a valve motor operator or a pump, is this accomplished from the control room or from a motor control center?

Response:

1. In general, when power is removed from a valve motor operator, the indication of valve position is lost. There are specific valves where there is backup indication in the control room on one of the ready status panels. Specifically, with respect to the valves in the safety injection system which might contribute to an overpressurization event, the accumulator isolation valves, 841A&B, the safety injection isolation valves, 866A&B, and the cold leg discharge valves, 878B&C do have this backup indication.
2. Power is always removed from a valve motor operator by opening the breaker at the motor control center.

7.

APPENDIX C

INTERIM MITIGATING SYSTEM DESCRIPTION

An "Interim Mitigating System" will be installed and utilized at Point Beach Nuclear Plant Units 1 and 2 until the final system, referred to as the "Overpressurization Mitigating System", has been installed. This Appendix describes the "Interim Mitigating System".

Installation Schedule

Unit -1 had the "Interim Mitigating System" installed during the fall 1976 refueling and has been in service since the Cycle 5 startup in November.

Unit -2 will install the "Interim Mitigating System" during the spring 1977 refueling and it will be in service during the startup of Cycle 4 which is scheduled for April.

System Description

The pressurizer power operated relief valves (PORV's) are utilized to provide protection against overpressurization events during periods of water solid operation. Inadvertent reactor coolant pressure increases are limited by opening the PORV's through the addition of a low pressure setpoint interlocked with a keylock switch located on the main control board. The logic required is in addition to the existing PORV actuation logic and will not interfere with existing automatic or manual actuation of the PORV's.

Since this overpressure protection is required only during periods of low pressure and water solid operation, the low pressure setpoint is energized and de-energized by the operator depending on plant conditions. He does so by use of an administratively controlled keylock switch located on the main control board.

During plant cooldown and prior to the collapse of the steam bubble in the pressurizer, the operator places the switch in the "Low Pressure" position. This enables the PORV's to be automatically actuated as required to mitigate an overpressurization transient. An alarm will alert the operator to the high pressure condition.

During plant heatup, the operating procedure will identify the plant conditions for which low pressure protection is no longer needed. The operator will place the keylock switch in the "Normal" position, returning control of the PORV's to the operating high pressure setpoint. Should the operator forget to return the switch to "Normal", an alarm will alert the operator to the high pressure condition. Placing the keylock switch in the "Normal" position removes or blocks the alarm.

A logic diagram of the interim mitigating system is shown in Figure C-1. A single keylock switch enables both valves low pressure logic and the pressure alarm. The instrumentation loop of one pressure transmitter includes bistables for each valve and the alarm. Figure C-2 shows a simplified instrumentation loop diagram for the pressure channel. The circuit for the PORV's is shown in Figure C-3. Because of the single channel, a loss of power to the associated instrumentation buss will prevent both valves from operating. As an interim system, this lack of redundancy is considered acceptable because pressure relief is available through diverse means, namely, relief valves on the PWR system, the time period of dependency on the interim system is short compared to the life of the plant, the likelihood of overpressure transients has been greatly reduced due to changes in procedures and special operator training, and the vessels have increased strength due to the short length of time they have been in service.

FIGURE C-1
 LOGIC DIAGRAM
 INTERIM MITIGATING SYSTEM

FIGURE C-2
WIDE RANGE PRESSURE SIGNAL

FIGURE C-3

TYPICAL POWER OPERATED RELIEF VALVE CIRCUIT

APPENDIX D

OVERPRESSURIZATION MITIGATING SYSTEM DESCRIPTION

This Appendix describes the system that will be utilized at Point Beach Nuclear Plant Units 1 and 2 to provide protection against exceeding 10 CFR 50 Appendix G limits during periods of water solid operation. This system will be referred to as the "Overpressurization Mitigating System" and will be installed in the plant as the long term solution to pressure transients during water solid operation. This system provides significantly improved reliability over the Interim System described in Appendix C and provides sufficient protection against overpressurization when the primary system is water solid.

Installation Schedule

- Unit -1 will be refueled in October, 1977. Assuming all technical issues can be settled and proper equipment procured by that date, the "Overpressurization Mitigating System" will be installed during this October, 1977, refueling. If delays are encountered in either of these areas, the next refueling in October, 1978, would be the installation date.
- Unit -2 will be refueled in March, 1978. Assuming all technical issues can be settled and proper equipment procured by that date, the system will be installed during this March, 1978, refueling. If delays are encountered in either of these areas, the next refueling scheduled for a year later would be the installation date.

System Description

The "Overpressurization Mitigating System" consists of two diverse means of relieving pressure during periods of water solid operation and when system temperature is below the value permitted to perform the primary system leak test. The primary method of water relief utilizes the pressurizer power operated relief valves (PORV's) and the diverse method uses the relief valves on the Residual Heat Removal System. The PORV's are made operational for low pressure relief by utilizing a dual setpoint where the low pressure circuit is energized and de-energized by the operator with a keylock switch depending on plant conditions. The logic required for the low pressure setpoint is in addition to the existing PORV actuation logic and will not interfere with existing automatic or manual actuation of the PORV's. The relief valves on the Residual Heat Removal (RHR) System are available for water relief whenever the RHR System is connected to the Reactor Coolant System. The RHR System is normally connected to the Reactor Coolant System during plant conditions when overpressurization events have been most prevalent, i.e., low temperature and low pressure conditions. The RHR relief valves can be considered a diverse relief system at Point Beach because the RHR isolation valves do not automatically isolate the RHR System during a pressure transient, thus making the valves available throughout the transient.

During plant cooldown and prior to the collapse of the steam bubble in the pressurizer, the operator under administrative procedure places the keylock switch in the "Low Pressure" position and connects the RHR System to the Reactor Coolant System. An alarm will alert the operator when the pressure is sufficiently low so that activation of the low pressure setpoint circuit will not inadvertently open a PORV. Placing the keylock switch in the "Low Pressure" position blocks the alarm indicating low pressure and enables the low temperature high pressure alarm.

During plant heatup, the operating procedure will identify the plant conditions for which low pressure protection is no longer needed. The operator places the keylock switch in the "Normal" position, returning control of the PORV's to the operating high pressure setpoint. An alarm will alert the operator to the high pressure condition requiring the switching action to avoid inadvertent opening of the PORV's. Placing the keylock switch in the "Normal" position removes or blocks the low temperature high pressure alarm and enables the low pressure operation alert alarm.

A logic diagram of the "Overpressurization Mitigating System" is shown in Figure D-1. Two redundant channels are provided with each channel consisting of a sensor, power supply, alarm bistable, alarm status light, PORV bistable, keylock switch, valve actuation circuitry and PORV. The low pressure alert and the low temperature high pressure alert of one channel share an annunciator with the similar alarm of the other channel.

Figure D-2 shows a simplified instrumentation loop diagram for the two pressure channels. The source of electrical power to channel 1's pressure transmitter, bistable auxiliary relays and valve actuation circuitry will be different from those of channel 2's. The "Overpressurization Mitigating System" is completely separate from the present plant protection system and therefore, degrading of the plant protection system is completely avoided. We have attempted to design a system that has a minimum impact on the plant but meets many of the requirements of IEEE 279, e.g.; the redundant channels allow the single failure criteria to be met, equipment quality and qualification is consistent with the classification of the system, channel integrity will be maintained, the system is designed against multiple failures resulting from a credible single event, the system inputs are derived from signals that are a direct measure of the desired variable, and the system is capable of test and calibration. Complete separation and independence of the two channels are not considered necessary because of the diverse relief capability on the RHR System. Because the existing pressure channel is treated as "control grade", it would be very difficult and detrimental to overall plant safety to make the system "safety grade". The resulting design of the "Overpressurization Mitigating System" considers the system to be treated as "control grade" because the increase in safety risk (decrease in safety) of installing a protection grade system in the plant is greater than any increases in safety a protection grade system would provide over and above the proposed "Overpressurization Mitigating System". It is also our belief that any overpressurization mitigating system should be treated as a control grade system due to the safety margins inherent in the Appendix G limits and the availability of the diverse relief system on the RHR system.

The circuit for the PORV's is shown in Figure D-3. Separate DC power sources are associated with each valve. The valves are designed to withstand seismic loading equivalent to 3.0g in the horizontal direction and 2.0g in the vertical direction and they are mounted on the pressurizer in a manner to maintain primary system integrity during the design seismic event. To ensure that the valves have an air supply for proper operation, an accumulator for each valve will be seismically qualified and mounted to an adjacent category I structure with air lines to the valves that will tolerate differential movements of components and structures. The remainder of the system is located within category I structures, trays and cabinets.

The "Overpressurization Mitigating System" described herein provides adequate and sufficient protection to the Reactor Coolant System against overpressurization transients. When coupled with the diverse relief capability on the RHR System, triple protection is provided. The design of this system is selected to be compatible with present design of the plant, thus minimizing modification and backfit to existing equipment. It is our belief that overall safety can best be provided by minimizing modifications to and backfitting of existing plant equipment and systems. Major modifications to nuclear plants have the potential to degrade the existing plant design and whenever construction work is required the potential for a serious fire is greatly increased. Any system that is more complex than the simple and straight forward system described herein, and requires additional changes to the plant, should be examined closely to determine if the added features are worth the cost and risk of installing those features. This backfit consideration applies more to operating plants because of the difficulty in making complex modifications to existing systems than to plants under construction or in the design phase. The "Overpressurization Mitigating System" provides the proper balance between these competing concerns at the Point Beach Nuclear Plant.

TABLE D-1

FAILURE MODE AND EFFECTS ANALYSIS

Failure Mode	Potential Effect on Availability			Comment
	PORV 1	PORV 2	RHR Relief	
Failure of channel 1 component	Not Op*	Op*	Op	Note 1
Failure of channel 2 component	Op	Not Op	Op	Note 2
Loss of channel 1 instrument power	Not Op	Op	Op	Note 2
Loss of channel 2 instrument power	Op	Not Op	Op	Note 2
Loss of PORV 1 battery	Not Op	Op	Op	
Loss of PORV 2 battery	Op	Not Op	Op	
Loss of instrument air	Op	Op	Op	Note 3
Loss of off-site power	Op	Op	Op	
Loss of off-site power plus loss of one diesel	Not Op	Op	Op	Note 2
Loss of off-site power plus loss of other diesel	Op	Op	Op	Note 2
Loss of letdown	Op	Op	Op	Note 3
Isolation of RHR	Op	Op	Not Op	Note 3
Closure of PORV 1 isolation valve	Not Op	Op	Op	
Closure of PORV 2 isolation valve	Op	Not Op	Op	
Inadvertent SI	Op	Op	Op	Note 3
Manual starting of SIS pump	Op	Op	Op	Note 3
Manual starting of additional charging pump	Op	Op	Op	Note 3
Manual starting RCP	Op	Op	Op	Note 3

Failure Mode	Resultant Effect on Operability			Comment
	PORV 1	PORV 2	Relief	
Failure to activate low pressure setpoint channel 1	Not Op	Op	Op	Note 4
Failure to activate low pressure setpoint channel 2	Op	Not Op	Op	Note 4
Problems with operation of pressurizer relief tank	Op	Op	Op	
Undefined common mode failure of channels 1 and 2	Not Op	Not Op	Op	Note 5

NOTES

*Not Op denotes that this valve is not operable and not capable of performing its pressure relieving function.

* Op denotes that this valve is operable and can perform its pressure relieving function.

Note 1 Channel component consists of one of the following: sensor, channel power supply, bistable to valve, or PORV.

Note 2 For the purpose of this analysis channel 1 power is supplied from a buss which gets power from one of the emergency diesels and channel 2 power is supplied from an inverter on a station battery.

Note 3 For all possible causes of overpressurization, no failure of PORV 1 or 2 results.

Note 4 Alarm will alert the operator to this condition.

Note 5 Common mode failure of channels 1 and 2 is considered incredible at the time of an overpressurization event, but this shows that pressure relief is still available via the RHR relief valves.

**FIGURE D-1
LOGIC DIAGRAM
OVERPRESSURIZATION MITIGATING SYSTEM**

1 and 2 denote sharing of annunciator
s indicates status light

FIGURE D-2
WIDE RANGE PRESSURE SIGNAL

FIGURE D-3

TYPICAL POWER OPERATED RELIEF VALVE CIRCUIT

DATE _____

UNIT # _____

GP-3C
MAJOR
Revision 14
12-21-76

HOT SHUTDOWN TO COLD SHUTDOWN

1.0 PURPOSE

This procedure describes the cooldown and depressurization of a unit from a hot shutdown condition to either a cold shutdown condition or a refueling shutdown condition.

2.0 PRECAUTION AND LIMITATION

- 2.1 The shutdown banks will be at the fully withdrawn position or at least one bank of rods with a total worth of 21% $\Delta k/k$ will be withdrawn prior to any planned positive reactivity changes. The above requirement may be waived under the following conditions:
 - 2.1.1 The reactor coolant system has been borated to at least the hot xenon-free boron concentration and is being maintained at hot shutdown.
 - 2.1.2 The reactor coolant system has been borated to the cold shutdown boron concentration.
- 2.2 Whenever possible following a change in reactor coolant boron concentration of >100 ppm, the pressurizer spray valves should be opened to adjust the pressurizer liquid boron concentration to within 100 ppm of the new reactor coolant concentration.
- 2.3 The maximum allowable cooldown rate for the reactor coolant system is 100°F/hour.
- 2.4 At least one reactor coolant pump or residual heat removal system must be in service during boron concentration changes or degassing operations.
- 2.5 Do not exceed 320°F ΔT across the pressurizer spray nozzles.
- 2.6 The reactor coolant system pressure must be reduced below 1765 psig, and safety injection normally blocked before the secondary side steam pressure is reduced below 530 psig.
- 2.7 The maximum cooldown rate for the pressurizer is 200°F/hr.
- 2.8 The reactor coolant system should be borated to the proper shutdown boron concentration before the cooldown is initiated and the blender auto mode concentration should be set to the same level.

- 2.9 The hydrogen concentration and the radioactivity in the reactor coolant must be reduced to as low as practicable if the system is to be opened for refueling or maintenance.
- 2.10 Reactor coolant pumps should be operated periodically to prevent the temperature difference between loops from exceeding about 250°F.
- 2.11 During cooldown, both steam generators should be used if possible to assure a more uniform cooldown of the loops and the steam generator secondary side.
- 2.12 The effect of boric acid addition should be observed in terms of the source range count rate and prompt corrective action taken if the count rate increases for unknown reasons.
- 2.13 The residual heat removal system must not be placed in service until the reactor coolant temperature is 3350°F and reactor coolant pressure is about 400 psig. Flow through the residual heat removal system must be initiated slowly to avoid thermal shocks.

NOTE: THE UPPER PRESSURE LIMIT FOR RESIDUAL HEAT REMOVAL OPERATION IS 450 PSIG. THE SYSTEM MUST BE BELOW 425 PSIG TO ALLOW THE VALVES TO BE OPENED.

- 2.14 The component cooling system temperature may be allowed to rise to 125°F for a maximum of two hours if necessary during a cooldown while using the residual heat removal system.
- 2.15 All reactor coolant pump operations must be performed within the requirements of OP-4B (Reactor Coolant Pump Operation).
- 2.16 The reactor coolant pressure and cooldown rate must be maintained in accordance with the pressure-temperature curve, Figure 1, OP-1A.
- 2.17 At least one reactor coolant pump should be kept in service during operation of the residual heat removal system during periods of little decay heat with a solid reactor coolant system. If pump operation is secured, colder water will accumulate in the loop and on restarting a pump, a transient will occur. The pump should not be restarted in this condition unless absolutely necessary.
 - 2.17.1 If a pump needs to be started under this condition, the following recovery procedure should be followed:
 - A. Minimize the charging flow rate.
 - B. Open all letdown orifice isolation valves and maximum letdown from residual heat removal system just prior to starting the reactor coolant pump.

- c. Reduce pressure to minimum required to start a pump.
 - d. Restart a reactor coolant pump and observe pressure.
- 2.18 Do not secure control rod drive mechanism ventilation until the reactor coolant temperature is below 200°F (or any time the mechanism coils are energized).
- 2.19 If the cooldown is for a reason which will require the reactor coolant being borated to the refueling concentration, insure that sufficient boric acid solution is available when necessary.
- 2.20 The charging line flow control valve must be adjusted as necessary to maintain the reactor coolant pump seal flow to about 8 gpm.
- 2.21 Insure water is available in the condensate storage tanks to provide for feed in case the cooldown is performed on the atmospheric steam dump valves. If alternate sources of water are required and the water treatment plant is not available, the hotwells should be pumped to the condensate storage tank. As a last resort, the service water supplies to the auxiliary feed-water pumps should be used.
- 2.22 The reactor coolant system should be sampled before and after the cooldown for gaseous radioactivity levels, dissolved hydrogen concentrations and boron level.

3.0 INITIAL CONDITIONS

Initials

- 3.1 The reactor is in the hot shutdown condition. _____
- 3.2 The reactor coolant is greater than or equal to the cold shutdown boron concentration. _____
- 3.3 At least one reactor coolant pump is in service. _____

4.0 PROCEDURE

- 4.1 Initiate degassing of the reactor coolant system if required in accordance with OP-5D. _____
- 4.2 Adjust the letdown flow rate to obtain the purification flow required for the subsequent cooldown. Maintain the required flow in accordance with OP-5A and chemistry requirements. _____

- 4.3 De-energize all pressurizer heaters after the pressurizer boron concentration is within approximately 50 ppm of the rest of the system. _____
- 4.4 Place the steam dump controller in the pressure control mode and adjust the setpoint as necessary to establish the desired cooldown rate. _____
- 4.5 Place the pressurizer spray valves in manual and commence cooling down the pressurizer.

NOTE: A. MAINTAIN APPROXIMATELY 100°F BETWEEN THE PRESSURIZER LIQUID TEMPERATURE AND THE REACTOR COOLANT TEMPERATURE DURING THE COOLDOWN AND THE REACTOR COOLANT PRESSURE-TEMPERATURE RELATIONSHIP WITHIN THE LIMITS DEFINED BY FIGURE ONE OF OP-1A.

B. AS THE COOLDOWN PROGRESSES, CONTROL CHARGING FLOW TO MAINTAIN A HIGH PRESSURIZER WATER LEVEL.

- 4.6 Manually block the automatic safety injection injection actuation circuit when the reactor coolant system pressure has decreased below 1765 psig. _____

CAUTION: THIS CIRCUIT WILL BE AUTOMATICALLY UNBLOCKED IF PRESSURE INCREASES ABOVE 1775 PSIG.

- 4.7 Unlock and close MOV-841A & B breakers. Then shut the accumulator isolation valves and return reactor coolant pump seal water differential pressure transmitter to service. _____

NOTE: THIS SHOULD BE DONE WITH THE REACTOR COOLANT PRESSURE ABOUT 1000 PSIG.

- 4.8 Open the reactor coolant pump No. 1 seal water bypass valve (RC-386), when the seal leakoff from either reactor coolant pump No. 1 seal decreases to 1 gpm. _____

NOTE: PRIOR TO COOLING DOWN BELOW 300°F AND BEFORE GOING SOLID THE SAFETY INJECTION PUMPS SHOULD BE RED TAGGED OFF AND THE SAFETY INJECTION ACCUMULATOR ISOLATION VALVES SHOULD BE RED TAGGED SHUT.

- 4.9 Align the residual heat removal loop for operation and place it in service in accordance with OP-7A. _____
- 4.10 When the reactor coolant system pressure decreases below 425 psig and the reactor coolant loop temperature is $\leq 350^{\circ}\text{F}$, arm the low temperature overpressurization protection and take the pressurizer solid and continue cooling down the pressurizer using the reactor coolant pumps. _____
- 4.11 Secure the secondary steam systems when no longer required in accordance with OP-13A and commence to place the steam generators in wet layup if required. _____
- 4.12 When the reactor coolant loop temperature is less than about 160°F , secure the reactor coolant pumps if desired. _____
- 4.13 Continue to cool down by residual heat removal system operation to the desired system temperature. _____

NOTE: MAINTAIN PURIFICATION LETDOWNS FLOW BY OPENING ADDITIONAL LETDOWN ORIFICES OR ESTABLISHING RESIDUAL HEAT REMOVAL PURIFICATION FLOW PER OP-7A, 4.14.

- 4.14 Prior to depressurizing the reactor coolant system, shut SI-894, safety injection pumps suction bypass valve. _____
- 4.15 After the reactor coolant pump is secured and prior to reducing the No. 1 seal differential to less than 100 psi, shut the No. 1 seal bypass valve CV-386, and shut the No. 1 seal leakoff isolation valves 270A and 270B and CV-313. Maintain the labyrinth seal differential pressure positive. During periods when the pump is disconnected from the motor, it may be necessary to secure seal water to the affected pump if leakage from the No. 3 seal overflows to the coupling cavity. _____

OP-4A
MAJOR
Revision 10
12-21-76

FILLING AND VENTING REACTOR COOLANT SYSTEM

1.0 PURPOSE

To define the operations necessary for filling and venting the reactor coolant system.

2.0 PRECAUTIONS AND LIMITATIONS

- 2.1 Do not pressurize the system to greater than 450 psig if the residual heat removal system is in operation.
- 2.2 Maintain reactor coolant system pressure in accordance with Figure 1 of OP-1A.
- 2.3 Do not vent the system if any portion of the reactor coolant system is greater than 200°F.
- 2.4 Do not allow pressure in the reactor coolant system to decrease below atmospheric.
- 2.5 Do not operate a reactor coolant pump except in accordance with OP-4B.
- 2.6 Observe all Health Physics procedures when venting.
- 2.7 Vent the control rod drive mechanisms as necessary.

NOTE: CONTROL ROD DRIVE MECHANISMS MUST BE VENTED IF EXERCISING IS TO BE DONE WITH THE REACTOR COOLANT PRESSURE BELOW 50 PSIG AS TO INSURE THAT THE WATER LEVEL IS ABOVE THE GRIPPER MECHANISMS.
- 2.8 Do not allow coolant to wet the mechanisms' coil stacks when venting the reactor vessel head or control rod drive mechanisms.
- 2.9 One safety valve must be operational when the reactor vessel head is in place.
- 2.10 Maximum operating pressure and temperature for the high pressure vent hose is 1000 psig and 225°F.

3.0 INITIAL CONDITIONS

- 3.1 The required equipment and valve checkoff sheets have been completed. (See last page of procedure.)
- 3.2 Mechanical, electrical and instrumentation checklists have been completed.

4.0 PROCEDURE

Initials

- 4.1 Install a one inch hose between the pressurizer relief tank at IVTC-1421 and the purge exhaust fan suction. (Figure 1)

- 4.2 Shut the pressurizer relief tank vent (RC-527) and nitrogen supply (RC-595). Insure a purge exhaust fan is running and open IVTC-1421 to vent the pressurizer relief tank. Insure WSA cavity exhaust fan is off and remains off.

- 4.3 Install a high pressure vent hose from the reactor vessel vent to the pressurizer vent, utilizing the venting spool piece for connection at the pressurizer. Install one inch 315 psig hose from the pressurizer spool piece to the pressurizer relief tank flanged vent. Open RC-535 and RC-500, the pressurizer relief tank flanged vent valve, and the missile shield spool piece isolation valve (Figure 1).

- 4.4 Establish communications between containment and control room.

- 4.5 Open pressurizer spray valves (RC-431A & B) and check the auxiliary spray valve (CV-296) shut.

- 4.6 Place the "blender" controls in auto and set the makeup concentration to that of the reactor coolant system if the blender and charging pumps are to be used.

- 4.7 During filling, shut the No. 1 seal outlet valves (270A & B) until pressure is established above the volume control tank pressure.

- 4.7.1 Verify the low temperature overpressurization circuit is armed.

- 4.8 Start one charging pump in manual and establish a reactor coolant pump seal injection flow of 3 to 5 gpm for each pump while the seal outlet valves are shut.

NOTE: DO NOT EXCEED LIMITS OF FIGURE 1 OF OP-14
WHILE PERFORMING THIS STEP.

- 4.9 Increase charging pump speed to force water into the system or use the residual heat removal pumps with suction from the refueling water storage tank. _____
- 4.10 Open RC-546A & C, 548A & C, 545A & C and 547A & C to vent steam generators to the pressurizer relief tank. _____
- 4.11 Vent control rod drive mechanisms if required. _____
- 4.12 Isolate the reactor vessel level transmitter by shutting the transmitter block isolation valves (2) and RC-523 and RC-524, transmitter to loop isolation valves. Isolate the reactor vessel level "sight glass" by shutting RC-525. _____
- NOTE: IF USING A CHARGING PUMP, REDUCE SPEED TO MINIMUM PRIOR TO FILLING THE PRESSURIZER SOLID.
- 4.13 When an increase in level is noted in the pressurizer relief tank, shut the valves opened in Step 4.10. _____
- 4.14 When the next increase in level in the pressurizer relief tank is noted, secure the charging pump if required and shut valve RC-535 and the missile shield spool piece isolation valve. Shift residual heat removal to circulation if this mode of fill has been used. _____
- 4.15 Shut the pressurizer relief tank flanged vent valve and disconnect the hose from pressurizer to pressurizer relief tank, and install the spool piece blank flange. Reopen RC-535 and the missile shield spool piece isolation valve. _____
- 4.16 Establish letdown flows in accordance with OP-5A. _____
- 4.17 Vent the reactor coolant pump seal water lines (three places, See Note A below). Instrumentation should be lined up per CL-4A, page 6 (required portion only). Vent the reactor coolant pumps labyrinth seal differential pressure instrumentation. _____

PT-131 _____ & PT-124 _____

NOTE A: MAXIMUM REACTOR COOLANT PRESSURE FOR VENTING REACTOR COOLANT PUMP SEAL WATER LINES IS 50 PSIG.

NOTE B: ALL OPERATIONS INVOLVING THE STARTING, STOPPING OR OPERATIONS OF THE REACTOR COOLANT PUMPS CALLED FOR BY THIS PROCEDURE MUST BE PERFORMED IN ACCORDANCE WITH OP-4B. THE MISSILE SHIELD MUST BE REMOVED FROM ABOVE THE REACTOR COOLANT PUMPS BEFORE EXTENDED LUMP OPERATION.

- 4.18 Insure I & C personnel have vented all reactor coolant pump instrumentation as per CL-4A. _____
- 4.19 Start reactor coolant pump P1A or P1B and secure it as soon as it is up to speed and the starting currents have decayed away. _____
- 4.20 Vent the reactor vessel head and pressurizer through the vapor space sample line. _____
- 4.21 Perform the following venting procedure:
 - 4.21.1 Start the reactor coolant pump not started in step 4.19 above. _____
 - 4.21.2 Secure the pump as soon as it is up to speed and the starting currents have decayed away. _____
 - 4.21.3 Vent the pressurizer and reactor vessel head as in Step 4.20. _____
- 4.22 Repeat step 4.21 alternating pumps until the very large amounts of air have ceased. Insure that pump starting limitations are not exceeded. _____
- 4.23 Start the reactor coolant pumps in sequence and allow to operate for five to ten minutes. _____
- 4.24 Stop the pumps, maintain pressure above 325 psig and vent the pressurizer, reactor vessel head as in step 4.20. _____
- NOTE: STEPS 4.23 AND 4.24 MAY BE REPEATED IF EXCESSIVE GAS IS RELEASED.
- 4.25 Insure that I & C personnel have vented all instrumentation and completed required checklists (See CL-4A, page 6). _____
- 4.26 Shut all vent valves, remove all vent apparatus and replace all flanges per CL-4A, page 7. _____
- 4.27 Shut pressurizer spray valves (RC-431A & B). _____
- 4.28 Establish proper conditions in pressurizer relief tank in accordance with OP-4C, after initially filling to high level and venting to the purge exhaust. _____

EQUIPMENT VERIFICATION CHECKOFF SHEET

1. Vessel head torqued ready for filling. _____
2. Instrumentation lined up for filling sytem (CL-4A, page 5). _____
3. The following system valve lineups completed as required:
 - a. Reactor coolant system (CL-4A) _____
 - b. Safety injection and RHR system (CL-7A) _____
 - c. CVCS (CL-5A) _____
 - d. Component cooling (CL-6A) _____
4. Venting appratus installed as required at the following points:
 - a. Pressurizer vent valve RC-535 to pressurizer relief tank. (Figure 1) _____
 - b. Pressurizer relief tank to purge exhaust (Figure 1). _____
 - c. Vessel vent RC-500 to pressurizer (Figure 1). _____
 - d. Control rod drive mechanisms, if required. _____
 - e. Reactor coolant pump PlA seal water vents (RC-302A, RC-206A, RC-305A). _____
 - f. Reactor coolant pump PlB seal water vents (RC-302B, RC-206B, RC-305B). _____
5. Reactor coolant pump standpipe level alarms checked. _____
Reactor cooalnt pump motor oil level glasses checked for proper oil level. _____

K & E 10-5545 500

COLD SHUTDOWN TO LOW POWER OPERATION

1.0 PURPOSE

This procedure describes the startup of a unit from a refueling shutdown condition or a cold shutdown condition to a condition of low power operation.

2.0 PRECAUTIONS AND LIMITATIONS

- 2.1 The reactor should not be made critical if a positive moderator temperature coefficient is anticipated except for zero power physic testing. Any time the reactor is critical, system pressure and temperature must be to the right of the criticality curve on Figure 1.
- 2.2 Criticality must be anticipated any time when control rods are being withdrawn or when boron dilution operations are in progress.
- 2.3 At least one reactor coolant pump must be operating when the plant is critical or a critical approach is being made.
- 2.4 When critical, boration must be initiated immediately if the insertion limit low low alarm is actuated.
- 2.5 With all rods and banks inserted, a shutdown margin of at least $3\frac{1}{2}$ $\Delta k/k$ must be maintained.
- 2.6 Any plant change which produces a sudden change in reactor coolant temperature (of the order of 10°F) or in reactor coolant boron concentration (of the order of 30 ppm) should be avoided while the reactor is critical at low power or approaching criticality.
- 2.7 The startup rate should not exceed one decade per minute (DPM).
- 2.8 Whenever possible following a change in reactor coolant boron concentration in excess of 100 ppm, the pressurizer spray control valves should be operated to adjust the pressurizer coolant boron concentration to within 100 ppm or less of the flowing reactor coolant.
- 2.9 Any time the plant is in the hot shutdown condition the feed rate to the steam generators must not exceed 400 gpm per generator and the level should be maintained above the feed ring (level in the narrow range instrument).

- 2.10 Control banks must always be withdrawn in the prescribed sequence.
- 2.11 The control rod drive mechanism cooling fans must be turned on before the reactor coolant temperature reaches 350°F, or anytime the mechanism coils are energized. Note the mechanisms may be energized at low temperatures for Reactor Engineering testing.
- 2.12 At least one reactor coolant pump or the residual heat removal system must be in service during boron concentration changes.
- 2.13 The shutdown banks will be at the fully withdrawn position or there will be at least one bank of rods with a worth of at least 1% $\Delta k/k$ withdrawn prior to any planned positive reactivity changes. The above requirement may be waived under the following conditions:
 - 2.13.1 The reactor coolant system has been borated to the hot xenon-free boron concentration and is being maintained at hot shutdown.
 - 2.13.2 The reactor coolant system has been borated to the cold shutdown boron concentration.
- 2.14 Insure that the water level in both steam generators is maintained within the indicating band during the heatup. It may be necessary to drain down from wet layup prior to starting the heatup. Sampling of the steam generators will be required, prior to bringing steam into the plant, if the generator were in wet layup. Dumping steam to atmosphere may be required to remove ammonia and excess hydrazine from the generators. Exercise caution when drawing off steam and adding feedwater to avoid step cooling of the reactor coolant system, particularly when the reactor is critical or approaching criticality.
- 2.15 Steam removal for increasing the main turbine speed to above 600 rpm may not be initiated until the reactor coolant temperature is above 540°F.
- 2.16 The reactor coolant pressure and heatup rate must be maintained in accordance with the pressure-temperature curve, Figure 1.
- 2.17 The reactor coolant must meet the required water chemistry specifications before the reactor coolant temperature is raised above 250°F.
- 2.18 Do not exceed 1700 psig in the primary system until the reactor coolant temperature is at least 480°F. This is necessary to avoid a safety injection signal due to steam generator pressure being below 530 psig. This minimum of 530 psig steam pressure must be maintained anytime the primary system pressure exceeds 1775 psig.

- 2.19 When scavenging oxygen from the primary system with hydrazine the primary system mixed bed demineralizers should be out of service prior to addition and for one hour following addition.
- 2.20 All primary system leak tests will be conducted at the temperature specified in Figure 1 of OP-1A.
- 2.21 The reactor coolant pressure must not be increased above 450 psig while the residual heat removal loop is in service.
- 2.22 Do not exceed a 320°F ΔT across the pressurizer spray nozzles.
- 2.23 Containment integrity must be established under the following conditions:
 - 2.23.1 Reactor coolant system temperature >200°F, or shutdown <1% $\Delta k/k$.
 - 2.23.2 Any rod motion causing positive reactivity changes except the following:
 - a) Rod drop tests and part-length rod disconnecting and reconnecting may be done without containment integrity provided the reactor is initially sub-critical by at least 10% $\Delta k/k$.
 - 2.23.3 If the reactor vessel head is removed, the reactor must be in the refueling shutdown condition.
 - 2.23.4 Positive reactivity injection by boron dilution unless the boron concentration is maintained ≥ 1800 ppm.
- 2.24 All primary system hydrostatic tests will be conducted at the temperature specified in Figure 1 of OP-1A.

3.0 INITIAL CONDITIONS

Initials

- 3.1 The following plant startup checkoff lists have been completed as required and the necessary portions of each system are in operation.

- 3.1.1 Reactor Coolant System, CL-4A
- 3.1.2 CVCS, CL-5A
- 3.1.3 Component Cooling System, CL-6A
- 3.1.4 Service Water System, CL-10B
- 3.1.5 Sampling Service, CL-10A
- 3.1.6 Waste Disposal System, CL-9A, B

3.1.7 Safety Injection System, CL-7A _____

3.1.8 Auxiliary Feedwater System, CL-13E _____

3.2 The reactor coolant system has been filled and vented as per OP-4A if required. _____

3.3 The residual heat removal system is in operation as required to remove core decay heat. _____

3.4 The steam generators are filled to the indicating level. They may be in the wet layup condition with vent valves open if nitrogen blanket is not established. _____

3.5 The necessary secondary systems are ready to receive steam and to return feedwater to the steam generator. _____

3.6 For one unit operation, insure the main feedwater pumps for the unit not on line have their operation handles in the lockout position to prevent an auto start of the motor-driven auxiliary feedwater pumps. _____

4.0 PROCEDURE

4.1 Establish charging, seal water and letdown flows in accordance with OP-5A. _____

4.2 Establish an indicating level in the secondary side of the steam generators. _____

4.3 Raise primary plant pressure by charging pump operation to the minimum required for reactor coolant pump operation per OP-4B. Do not exceed 450 psig while the RHR system is in operation. Open the No. 1 seal bypass valve CV-386 after a minimum of 100 psi differential pressure is developed across the No. 1 seal. Leave valve open throughout the conclusion of this procedure. CV-386 will be shut under administrative control after a period of hot shutdown. _____

4.4 Start the reactor coolant pumps in accordance with OP-4B. _____

CAUTION: CONTAINMENT INTEGRITY MUST BE ESTABLISHED
PRIOR TO INCREASING TEMPERATURE ABOVE 200°F.
SEE PRECAUTION AND LIMITATION 2.23.

4.5 Close reactor trip breakers and withdraw all shutdown and control rods five steps. If the part length rods are not withdrawn, withdraw them five steps also. Reference Step 4.14 and Precaution 2.11. _____

- 4.6 Close steam generator vents when primary temperature is 210°F and steam is observed to be issuing from them.
- 4.7 Energize all pressurizer heaters and start increasing the pressurizer temperature. Open the pressurizer spray valves.

NOTE: THE PRESSURIZER HEATERS ARE BEING USED AS AN AUXILIARY HEAT SOURCE. DO NOT DRAW A BUBBLE AT THIS TIME. DO NOT ALLOW THE PRESSURIZER TO GET OVER 100°F ABOVE THE REACTOR COOLANT SYSTEM TEMPERATURE.

- 4.8 As soon as practicable, after at least five minutes of reactor coolant pump operation, sample the primary system for dissolved oxygen.
- 4.9 Check SI-894, safety injection pump suction valve bypass valve open.
- 4.10 Add hydrazine as required for oxygen scavenging in accordance with OP-5D. NOTE: if hydrazine must be added, primary temperature must be <250°F. See Precaution and Limitations 2.17.
- 4.11 Recirculate the pressurizer by using all available pressurizer heaters and sprays in order to remove the dissolved oxygen present in the pressurizer water.
- 4.12 Heatup the primary system using the reactor coolant pumps, pressurizer heaters and core decay heat to a temperature greater than that specified in Figure 1 of OP-1A for leak test.

NOTE: STEP 4.13 MAY BE DELETED IF A STEAM LEAK TEST IS NOT REQUIRED. IF A LEAK TEST IS REQUIRED, THE SAFETY INJECTION SIGNAL FROM LOW STEAM GENERATOR PRESSURE MUST BE DISABLED IN ORDER TO PREVENT A SAFETY INJECTION SIGNAL FROM BEING ACTUATED WHEN THE PRESSURE IS RAISED TO ABOVE 1775 PSIG.

- 4.13 Primary system leak test if required.
- 4.13.1 Disable overpressurization protection.
- 4.13.2 Remove RHR from service and insure normal CVCS letdown operating.

- 4.13.3 Raise primary pressure slowly using the charging pumps to 1000 psig. Inspect the reactor vessel head seals and all other areas of the primary pressure boundary for leakage. _____
- 4.13.4 Deactivate both pressurizer power operated relief valves and place the spray control valves on manual and in the off position. Defeat "lo-lo" steam pressure safety injection as per ICP 10.7. _____
- 4.13.5 Raise primary pressure slowly using the charging pumps to 2000 psig. Inspect for leakage. _____
- 4.13.6 Secure the pressurizer heaters and secure the reactor coolant pumps if necessary. _____
- 4.13.7 Raise primary pressure slowly using the charging pumps to 2335 psig. Inspect for leakage. _____

Time reached 2335 _____
Temperature _____
Time leak test secured _____

Results Sat - Unsat.

COMMENTS:

- 4.13.8 Reduce system pressure to the minimum allowed for starting reactor coolant pumps, in accordance with OP-4B, start a reactor coolant pump as required. Return the power-operated relief valves to automatic. _____
- a) When pressure is 1765 psig, block safety injection. Insure the safety injection system is back in service if disabled for the leak test as per ICP 10.7. _____
- b) Return residual heat removal to service if leak test failed or secure residual heat removal in accordance with OP-7B. Disable overpressurization protection if in service. _____

- 4.13.9 Energize all pressurizer heaters and start increasing pressurizer temperature. _____
- 4.14 Insure that control rod drive mechanism cooling fans are in operation before the reactor coolant system temperature reaches 350°F. _____
- 4.15 When the pressurizer temperature reaches saturation for the pressure being maintained, draw a bubble and lower the level to the normal operating level in the pressurizer. _____
- 4.16 When the pressurizer water level approaches the no-load Tavg programmed level, switch the charging pump speed controller to automatic. _____
- 4.17 Secure the residual heat removal system in accordance with OP-7B. Align the residual heat removal pump suction to the refueling water storage tank. (If not already completed in 4.13.7 b). _____
- NOTE: LETDOWN MUST BE VIA THE LETDOWN CRIFICES
PRIOR TO SECURING THE RESIDUAL HEAT REMOVAL
ISOLATION VALVES.
- 4.18 Set the NIS recorder to record the count rate of the highest source range channel and an intermediate range channel. _____
- 4.19 If inserted, withdraw the shutdown banks to the fully withdrawn position. _____
- 4.20 Commence heating up the primary coolant system. _____

NOTE: HAVE CHEMISTRY SAMPLE THE SECONDARY SIDE OF THE STEAM GENERATORS FOR AMMONIA. BLOW OFF OF THE STEAM GENERATORS VIA THE ATMOSPHERIC STEAM DUMP MAY BE REQUIRED PRIOR TO BRINGING STEAM INTO THE SECONDARY PLANT.

- 4.21 Conduct a dilution operation in accordance with OP-5B to reduce the boron concentration to the cold shutdown or hot shutdown value whichever is the greatest. The CVCS mixed bed demineralizers should be cut in during this step to insure the resin remains in boron equilibrium with the coolant. _____
- 4.22 Form a hydrogen blanket in the volume control tank in accordance with OP-5D. _____

NOTE: THIS STEP MAY BE DONE AT ANY TIME AFTER NORMAL LETDOWN IS ESTABLISHED AND OXYGEN IN REACTOR COOLANT HAS BEEN REDUCED TO ≤ 1.1 PPM.

- 4.23 During the heatup, adjust the low pressure letdown controller and the number of letdown orifices in service as necessary to maintain the letdown flow required for heatup. The final setting of the controller should be between 225 and 275 psig.

- 4.24 Heatup the plant with reactor coolant pumps until the moderator temperature coefficient is no longer positive.

NOTE: IF DESIRED, THE REACTOR COOLANT PUMPS MAY BE USED TO HEATUP THE PLANT TO NORMAL OPERATING TEMPERATURE AND PRESSURE.

- 4.25 When the reactor coolant system pressure is >1000 psig, verify the accumulator gas pressure and levels are within the operating band then open the accumulator isolation valves MOV-841A & B, insure that the valves are placed firmly on their backseats. Then open and lock open MOV-841A & B breakers.

- 4.26 Isolate PT-173 and 174 reactor coolant pump seal differential pressure transmitters.

- 4.27 Verify proper position of safety injection valves per the ready status board on COI.

- 4.28 Commence hot primary system valve lineup, CL-4B, when primary temperature is greater than 400°F.

- 4.29 Take the reactor plant critical in accordance with OP-1B.

- 4.30 Stabilize reactor power to get a heatup rate of between 50 and 100°F/hour, if a nuclear heatup is required, to establish normal operating temperature and pressure.

- 4.31 Put the pressurizer pressure control system in automatic when the reactor coolant system is at normal operating pressure.

- 4.32 When the reactor coolant temperature approaches the normal no load value, verify the steam dump is on pressure control if it is available.

- 4.33 Stabilize the plant at no load, normal operating pressure with the steam dump or atmospheric dump controlling plant temperature at about the no load value.

NOTE: PRIOR TO ADMITTING STEAM INTO THE SECONDARY PLANT INSURE THAT AMMONIA LEVELS ARE SATISFACTORY BY CHEMISTRY ANALYSIS.

UNIT 2 - FIGURE 2, SHEET 1 of 2

UNIT 2 - FIGURE 2, SHEET 2 OF 2

REVISIONS	
NO.	DESCRIPTION
1	Issued for Unit 2 Cooldown
2	Revised by S. S.
3	Revised by S. S.
4	Revised by S. S.
5	Revised by S. S.
6	Revised by S. S.
7	Revised by S. S.
8	Revised by S. S.
9	Revised by S. S.
10	Revised by S. S.

OPERATIONS REFUELING TEST
FLOW TEST OF HIGH HEAD SAFETY INJECTION PUMPS

1.0 PURPOSE

To verify safety injection pumps P15A & B deliver rated flow.

2.0 REFERENCES

- 2.1 Safety injection system technical manual
- 2.2 Westinghouse flow diagrams, 110E017 and 110E035
- 2.3 PFDSAR, Section 6, Volume 2

3.0 PRECAUTIONS AND LIMITATIONS

- 3.1 The RWST level must be monitored closely and start and stop levels must be recorded as accurately as possible to allow repeatability of the conditions for future tests.
- 3.2 When MOV-825A or B is open a flow path will exist from the RWST to the reactor vessel.

4.0 INITIAL CONDITIONS

- 4.1 Safety injection lined up per CL-ORT #1.
- 4.2 The following instrumentation has been calibrated within the last 30 days:
 - 4.2.1 P15A discharge pressure, PT-923
 - 4.2.2 Injection flow, FI-925
 - 4.2.3 P15A, motor current on C01
 - 4.2.4 P15B, discharge pressure PT-922
 - 4.2.5 Injection flow, FI-925
 - 4.2.6 P15B, motor current on C01
 - 4.2.7 RWST level, PT-920
- 4.3 Portable vibration IFD at safety injection pumps.

- ## 5.0 TEST

- psi
gpm
amps
mils
mils

5.3 Open 878D. After one minute record:

RWST level, LT-920		
P15A discharge pressure, PT-923		psig
Injection flow, FI-925		gpm
P15A current		amps
Pump vibration	Vertical	mils
	Horizontal	mils

5.4 Open 878D. After one minute records:

RWST level, LT-920		
P15A discharge pressure, PT-923		psig
Injection flow, FI-925		gpm
P15A current		amps
Pump vibration	Vertical	mils
	Horizontal	mils

5.5 Shut 878B & D.

5.6 Secure P15A.

5.7 Start P15B. After starting current inching, run pump for one minute. Record.

RWST level		
P15B discharge pressure, PT-922		psig
Injection flow, FI-924		gpm
P15B current		amps
Pump vibration	Vertical	mils
	Horizontal	mils

5.8 Secure P15B.

5.9 Shut MOV-825A.

5.10 Open 878E.

5.11 Open MOV-825B.

5.12 Start P15B. After one minute record:

RWST level		
P15B discharge pressure, PT-922		psig
Injection flow, FI-924		gpm
P15B current		amps
Pump vibration	Vertical	mils
	Horizontal	mils

5.13 Secure P15B.

5.14 Shut MOV-825B.

5.15 Open 878F.

5.16 Open MOV-825B.

5.17 Start P15B. After one minute record:

RMST level

P15B discharge pressure, PT-922

Injection flow, FI-924

P15B current

Pump vibration

Vertical

Horizontal

psig

gpm

amps

mils

mils

5.18 Secure P15B.

5.19 Shut MOV-825B.

5.20 Return system to a status as directed by the Duty
Shift Supervisor.

Performed by _____

Duty Shift Supervisor _____

Unit _____

Date _____

CL-ORT #1

Revision 1

12-21-76

CONTROL ROOM

	<u>Desired</u>	<u>Actual</u>	<u>Initials</u>
MOV-825A RWST to safety injection pump suction stop	S	_____	_____
MOV-825B RWST to safety injection pump suction stop	S	_____	_____
MOV-826B Boric acid tank to safety injection pump			
suction stop	S	_____	_____
MOV-826C Boric acid tank to safety injection pump			
suction stop	S	_____	_____
MOV-896A P15A suction	S	_____	_____
MOV-896B P15B suction	O	_____	_____
MOV-852A Core deluge	S	_____	_____
MOV-852B Core deluge	S	_____	_____
MOV-866A P15A discharge stop	O	_____	_____
MOV-866B P15B discharge stop	O	_____	_____
MOV-841A Accumulator "A" discharge stop	S	_____	_____
MOV-841B Accumulator "B" discharge stop	S	_____	_____
MOV-878A Injection valve	S	_____	_____
MOV-878B Injection valve	S	_____	_____
MOV-878C Injection valve	S	_____	_____
MOV-878D Injection valve	S	_____	_____
AOV-897A Test line isolation	O	_____	_____
AOV-897B Test line isolation	O	_____	_____
AOV-839A Accumulator "A" test valve	S	_____	_____
AOV-839B Loop "A" Tc test valve	S	_____	_____
AOV-839C Accumulator "B" test valve	S	_____	_____
AOV-839D Loop "B" Tc test valve	S	_____	_____
AOV-835A Accumulator "A" makeup valve	S	_____	_____
AOV-835B Accumulator "B" makeup valve	S	_____	_____

CONTAINMENT - "B" REACTOR COOLANT PUMP COMPARTMENT

SI-V7	"B" accumulator discharge vent	S	_____	_____
-------	--	---	-------	-------

CONTAINMENT - 45' LEVEL

RH-V3	Low head safety injection vent	S	_____	_____
-------	--	---	-------	-------

CONTAINMENT - 26' LEVEL

SI-V9	Core deluge vent (behind "B" accumulator)	S	_____	_____
878E	Loop "A" Tc injection	S	_____	_____
878F	Loop "B" Tc injection	S	_____	_____
SJ-V10	Tc "B" injection test	S	_____	_____
SI-V11	Tc "A" injection test	S	_____	_____
SI-D18	Test line test connection - Unit 1 only	S	_____	_____

		<u>Desired</u>	<u>Actual</u>	<u>Initials</u>
<u>CONTAINMENT - B' LEVEL</u>				
SI-D19	Test line drain	S	_____	_____
1408	Safety injection test line I	S	_____	_____
879A	Safety injection test line isolation	O	_____	_____
<u>CONTAINMENT - "A" REACTOR COOLANT PUMP COMPARTMENT</u>				
SI-V6	Safety injection accumulator discharge vent	S	_____	_____
<u>AUXILIARY BUILDING</u>				
SI-884	Test line isolation	S	_____	_____
SI-879B	Test line isolation	S	_____	_____
SI-894	MOV-825A & B bypass	S	_____	_____
SI-857A	RHR to safety injection pump suction	S	_____	_____
SI-D9	Safety injection pump suction drain	S	_____	_____
SI-881B	PT-922 root isolation	O	_____	_____
SI-D9	P15B discharge drain	S	_____	_____
SI-888B	P15B discharge stop	O	_____	_____
SI-868A	P15A discharge stop	O	_____	_____
SI-829A	Safety injection discharge header cross-connection	S	_____	_____
SI-829B	Safety injection discharge header cross-connection	S	_____	_____
SI-881A	PT-923 root isolation	O	_____	_____
SI-D10	P15A discharge drain	S	_____	_____
SI-876B	P15B recirculation	O	_____	_____
SI-876A	P15A recirculation	O	_____	_____
SI-873A	Spray pump "A" recirculation	S	_____	_____
SI-864D	Spray pump "A" mini-recirculation	S	_____	_____
SI-873B	Spray pump "B" recirculation	S	_____	_____
SI-864C	Spray pump "B" mini-recirculation	S	_____	_____
SI-D13	P15A discharge header drain	S	_____	_____
SI-848A	FI-925 root isolation	O	_____	_____
SI-848B	FI-925 root isolation	O	_____	_____
SI-848C	FI-924 root isolation	O	_____	_____
SI-848D	FI-924 root isolation	O	_____	_____
SI-857B	RHR to SI pump suction	S	_____	_____
SI-864A	Spray pump "A" discharge test	S	_____	_____
SI-864B	Spray pump "B" discharge test	S	_____	_____
SI-D20	Drain by 2FE-924	S	_____	_____
SI-D21	Drain by 2FE-924	S	_____	_____
SI-V12	P15B discharge vent (Unit 2 only)	S	_____	_____

POINT BEACH NUCLEAR PLANT

WISCONSIN MICHIGAN POWER COMPANY

OPERATIONS REFUELING TEST

LOSS OF ENGINEERED SAFEGUARDS AC SIMULTANEOUS
WITH SAFETY INJECTION
UNIT 1SPECIAL PROCEDURES

Originated By

Date

Reviewed By

Date

Approved By

Date

WMTP'sRESULTComments
Attached
Yes No

Performed By

Date

Approved By

Date

Approved By

Date

ORT's

Originated By

Date

Checked By

Date

Operations Sup't.
(Approved)

Date

MAINTENANCESMP'sOPERATIONS

Originated By

Date

Checked By

Date

Approved By

Date

Originated By

Date

Checked By

Date

LOSS OF ENGINEERED SAFEGUARDS AC SIMULTANEOUS
WITH SAFETY INJECTION
UNIT 1

1.0 PURPOSE

The purpose of this test is to demonstrate proper:

- 1.1 Operation of safety injection sequencing and diesel loading under conditions of loss of engineered safeguards AC plus safety injection. (15.4.6-1) (15.4.5-1)
- 1.2 Containment isolation. (15.4.1-2)
- 1.3 Operation of service water system with safety injection and less than four out of six service water pumps. (15.4.1-2)
- 1.4 Operation of containment service water return bypass valves. (15.4.5-2)

2.0 PRECAUTIONS

- 2.1 If four service water pumps fail to start, service water isolation will occur to the affected unit.
- 2.2 All pumps with the exception of service water will be run in the recirculation mode.
- 2.3 Note all test connections and lifted wires on attached "Lifted Wire and Test Connection" sheet.
- 2.4 As soon as safety injection pump operation is verified in step 4.16.3, secure and go to pull out on safety injection pump control switch. Safety injection pumps should not be operated with both suctions shut.

3.0 INITIAL CONDITIONS

Initials

- 3.1 Unit 1 reactor shut down and in a refueling outage. _____
- 3.2 All safeguards system lineup as per attached Checklist CLORT #3-1. _____
- 3.3 Following equipment operating to prevent loss of process:
 - 3.3.1 K2B instrument air compressor _____
 - 3.3.2 D09 battery charger supplied from 2B31 and in service to "A" battery _____
 - 3.3.3 K3B service air compressor _____

- 3.3.4 KIA waste gas compressor _____
- 3.3.5 Combination of P32D, E & F service water pumps _____
- 3.4 Shut down the following equipment:
 - 3.4.1 #2 boric acid evaporator _____
 - 3.4.2 2 gpm waste evaporator _____
 - 3.4.3 Blowdown evaporator _____
 - 3.4.4 Letdown gas strippers _____
 - 3.4.5 Steam generator blowdown on Unit 2 _____
 - 3.4.6 Unit 1 air particulate detector _____
 - 3.4.7 Unit 2 air particulate detector _____
- 3.5 Plant in a condition such that loss of MCC B43, 1B31, will not cause damage. _____
- 3.6 Recorders hooked up to monitor timed sequence start of the below major components:
 - 3.6.1 Safety injection pumps A & B _____
 - 3.6.2 Residual heat removal pumps A & B _____
 - 3.6.3 Auxiliary feed pumps A & B _____
 - 3.6.4 Service water pumps A _____
 - B _____
 - C _____
 - D _____
 - E _____
 - F _____
 - 3.6.5 Containment vent coolers W1A1 _____
 - W1B1 _____
 - W1C1 _____
 - W1D1 _____

4.0 PROCEDURE

4.1 Insure accumulator discharge valves MOV-841A&B and residual heat removal to reactor vessel MOV-852A&B are shut. Open MOV-825A, safety injection pump suction from RWST, and manually go to close on valve until intermediate valve position indicated is received.

4.1.1 Lift the motor leads at the controller for these valves, then reclose breaker.

4.2 Clear trips and reset the scram breakers.

4.3 Install a jumper between the terminals 7A (TPAB-11) and 7EA (TPAD-11). Open slides for 62 (TPAC-5) and 62A (TPAC-6) in IC03. Jumper between TPAJ2 and TPAJ1 open slides on TPAJ8 and TPAJ9. Main feed pump oil pressure and suction pressure breaker trips.

4.4 Open the following valves:

4.4.1 480 feed regulator bypass
481 feed regulator bypass
4.4.2 466 main feed regulator
476 main feed regulator

4.4.3 CV-3212, containment vent exhaust isolation
CV-3213, containment vent exhaust isolation
CV-3244, containment vent supply isolation
4.4.4 CV-3245, containment vent supply isolation

4.5 Shut the following valves:

4.5.1 MOV-4020, P38B to 2B steam generator
4.5.2 MOV-4021, P38B to 1B steam generator
MOV-4022, P38A to 2A steam generator
4.5.3 MOV-4023, P38B to 1A steam generator

4.6 Open the following valves in the steam system for the unit to be tested:

4.6.1 TVSA9, radwaste system steam supply
PCVSA1, radwaste system steam reducer

4.7 Place the feed pump breakers in test position and operate the key switch in "A" pump breaker.

NOTE: THE KEY SWITCH INSTALLED IN THE FEED PUMP BREAKER CUBICLE WILL ALLOW SEQUENCING OF THE AUXILIARY FEED PUMPS.

4.8 Close the feed pump breakers.

4.9 Place the breaker for the non-operating component cooling water pump in the test position and close the breaker. _____

4.10 Place the charging pump breakers in test and close the breakers. _____

4.10.1 Lift leads in 1C158 on TB49-8, TB49-10 and TB49-12. Pressurizer low level heater cut out. _____

4.10.2 Place the following equipment breakers in test position: _____

T1C, Pressurizer heater breaker
T1D, Pressurizer heater breaker
T1E, Pressurizer heater breaker

4.11 Insure that the following equipment breakers are racked in and control power is on: _____

4.11.1 P15A, safety injection pump
P15B, safety injection pump
P10A, residual heat removal pump
P10B, residual heat removal pump
P32A, service water pump
P32B, service water pump
P32C, service water pump
P32D, service water pump
P32E, service water pump
P38A, auxiliary feed pump
P38B, auxiliary feed pump

W1A1, containment recirc fan
W1B1, containment recirc fan
W1C1, containment recirc fan
W1D1, containment recirc fan
3D diesel generator breakers to A05
4D diesel generator breakers to A06
P32F, service water pump

4.12 Shut RCDT manual drain 1704, place the control switches for RCDT pumps in pullout, place the control switches for RCDT pump suction valves in automatic and fill RCDT until the valves open. Open 1721 and place in automatic. Tape the control switch for 1698 in the open position. _____

4.12.1 Initial valve positions per Appendix A. _____

4.13 Place instrument air compressor K2A in continuous

4.14 Start the auxiliary feed pumps P38A and B.

4.15 Perform test by:

4.15.1 First open supply breakers A03 to A05 and A04 to A06. Go to unblock on the safety injection block/unblock switch.

4.16 The following events will occur:

4.16.1 Alarms

4.16.1.1 Containment vent coolers low water flow alarm on COL

4.16.1.2 Unit 1 containment vent isolation alarm on COL

4.16.1.3 Containment isolation alarm on COL

4.16.2 Reactor trip breakers open

4.16.3 Check the following loads stripped:

B03 and B04 safety injection lockout relays
stripped

CCW pump A

CCW pump B

Auxiliary feed pump P38A

Auxiliary feed pump P38B

Main feed pump P28A

Main feed pump P28B

Instrument air compressor K2A

Charging pumps T2A, B & C

4.16.4 Diesels will start and sequence on the following loads:

4.16.4.1 * Safety injection pump A

4.16.4.2 * Safety injection pump B

4.16.4.3 Residual heat removal pump A

4.16.4.4 Residual heat removal pump B

4.16.4.5 Auxiliary feed pump A

4.16.4.6 Auxiliary feed pump B

4.16.4.7 Service water pump A

4.16.4.8 Service water pump B

4.16.4.9 Service water pump C

*Go to pull out as soon as operation is verified.

Service water pump D
Service water pump E
Service water pump F
Containment fan W1A1
Containment fan W1B1
Containment fan W1C1
Containment fan W1D1

4.17 The following valves should close:

CV-3212, containment vent exhaust isolation
CV-3213, containment vent exhaust isolation
CV-3244, containment vent supply isolation
CV-3245, containment vent supply isolation
CV-480, feed regulator bypass valve
CV-466, main feed regulator valve bypass
CV-481, feed regulator valve bypass
CV-476, main feed regulator valve
TVSA9, radwaste system steam supply
PCVSA1, radwaste system steam supply
Valves listed in Appendix A attached
(PCVSA1, will "auto" reopen in step 4.24)

4.18 The following motor-operated valves will open:

MOV-878B, SI to Loop C coldleg
MOV-876D, SI to Loop A coldleg
MOV-2907, service water return from vent coolers
MOV-2908, service water return from vent coolers
MOV-826A, boric acid tank to SI pump suction
MOV-826B, boric acid tank to SI pump suction
MOV-826C, boric acid tank to SI pump suction

Observe controller picked up for the following valves:

841A, accumulator discharge valve
841B, accumulator discharge valve
MOV-852A, RHR to reactor vessel
MOV-852B, RHR to reactor vessel

4.19 The following motor breakers will operate as indicated:

SI pump A	Close	_____
SI pump B	Close	_____
Auxiliary feed pump A	Close	_____
Auxiliary feed pump B	Close	_____
Service water pump A	Close	_____
Service water pump E	Close	_____

Service water pump C	Close	_____
Service water pump D	Close	_____
Service water pump E	Close	_____
Service water pump F	Close	_____
Containment fan cooler WIA1	Close	_____
Containment fan cooler WIB1	Close	_____
Containment fan cooler WIC1	Close	_____
Containment fan cooler WID1	Close	_____
RHR pump A	Close	_____
RHR pump B	Close	_____

4.20 Observe change over from AC to DC lights by pushing test button on front of emergency lighting change over panel.

4.21 Check the safety injection signal has tripped the following relays:

CV- Panel C-153, SI-11X
CV- Panel C-163, SI-11X
TV- Panel C-154, SI-21X
PCV- Panel C-164, SI-21X
VBB

The relay is tripped if the little buttons in the center of the relay are sticking out.

4.22 Check all valves listed in Appendix "A" have cycled shut.

4.22.1 Place switch for V-2042 or V-2045 in the open position. Only the valve for the switch which was operated should open.

4.23 Observe six service water pumps in operation, then proceed as follows:

NOTE: SECURE BLOWDOWN TO UNIT 2 DURING TESTS.

4.23.1 Secure 3 out of 6 service water pumps.

NOTE: THIS WILL CAUSE A LESS THAN 4/6 LOGIC ON SERVICE WATER AND THE FOLLOWING MOTOR OPERATED VALVES WILL OPERATE CLOSED.

4.23.2 The following:

MOV-2880	Unit 1 turbine hall supply	_____
MOV-2817		_____
MOV-2816		_____
MOV-2930		_____
TVLN61	service water to blowdown evap.	_____
TVLN62	service water from blowdown evap.	_____

- 4.23.2 Observe 2MOV-2880 not closed. Hold switch open if closed.
- 4.23.3 When the above valves have operated, restart secured service water pumps.
- 4.24 90 seconds after initiation block and reset safety injection and containment isolation signals.
- 4.24.1 Check safety injection relays in Step 4.21 reset.
- 4.24.2 Reset BO3 and BO4 safety injection lockout relays.
- 4.24.3 Close pressurizer heater breakers for groups T-1C, T-1D, and T-1E.
- 4.24.4 Start K-3A service air compressor and electric fire pump P35A.
- 4.24.5 Reclose MCC breakers for 1B-31 and B-43.
- 4.24.6 Reclose 1P2A, B & C charging pump breakers.
- 4.25 Initiate manual safety injection and containment isolation. Check manual safety injection alarm and containment isolation alarm received.
- 4.25.1 Check 1P-2A, 2B, and C breakers not tripped.
- 4.25.2 Check BO3 and BO4 safety injection lockout relays tripped.
- 4.25.3 Check following breakers tripped:
- T-1C, pressurizer heater breaker
 - T-1D, pressurizer heater breaker
 - T-1E, pressurizer heater breaker
 - P35A, electric fire pump
 - K-3A, service air compressor
 - 1B-31, feeder breaker
 - B-43, feeder breaker

- 4.26 Ninety (90) seconds after initiation reset safety injection and containment isolation signals. _____
- 4.27 Inject lo-lo signal from boric acid tank level controller and check safety injection pump suction changes from boric acid tank to RWST. _____
4.27.1	*825A	Open	_____
	825B	Open	_____
	826C	Shut	_____
	826B	Shut	_____
- *Check 825A "open" contactor on breaker picked up.
- 4.28 Return system to normal as directed by Duty Shift Supervisor. _____
- 4.29 Restore electrical power as directed. _____
- 4.30 Restart loads supplied from stripped buses as necessary. _____
- 4.31 Restore battery chargers and batteries to normal lineup. _____
- 4.32 Remove key from "A" steam generator feed pump breaker and return to Operations Superintendent. _____
- 4.33 Disconnect all temporary jumper and restore all lifted wires per attached "Lifted Wire and Test Connection" sheet. _____
- 4.34 Restore all equipment which is in an abnormal condition, to desired condition as per Duty Shift Supervisor. _____

CONTAINMENT ISOLATION VALVES' OPERABILITY TEST UNIT 1

DATE 4.11.76

Technical Specifications 15.4.4.III.D.2: Remotely operated isolation valves shall be tested for operability at a frequency of at least eighteen months.

NOTE: ANY VALVE WHICH HAS BEEN CYCLED IN THE MONTH PREVIOUS TO THE TLST NEED NOT BE RECYCLED PROVIDED FULL DETAILS IDENTIFYING THE TEST IN WHICH IT WAS CYCLED ARE PLACED IN THE REMARKS COLUMN.

The following valves may be cycled at plant full power or at any normal plant condition utilizing the following special conditions prior to the refueling shutdown test (ORT #3). Each should be evaluated and open prior to the start of the test.

VALVE NO.	FUNCTION	SPECIAL INITIAL CONDITIONS	INITIALS	
			OPEN	CLOSED
508	Pzr relief tank makeup water	None		
538	Pzr relief tank gas sample	None		
539	Pzr relief tank gas sample	None		
1296	Aux charging line isolation	None		
371	Letdown isolation	None		
769	CCW return from excess letdown heat exchanger	None		
3200A	Containment air sample return	Shutdown sample pump		
3200B	Containment air sample from cont.	Shutdown sample pump		

VALVE NO.	FUNCTION	SPECIAL INITIAL CONDITIONS	INITIALS	
			OPEN	CLOSED
3200C	Containment air sample from cont.	Shutdown sample pump	_____	_____
3244	Purge supply	Vent containment prior to cycling this valve	_____	_____
3245	Purge supply	Vent containment prior to cycling this valve	_____	_____
3212	Purge exhaust	Vent containment prior to cycling this valve	_____	_____
3213	Purge exhaust	Vent containment prior to cycling this valve	_____	_____
3047	Instrument air to the cont.	None	_____	_____
3048	Instrument air to the cont.	None	_____	_____
966A	Pzr vapor sample	None	_____	_____
966B	Pzr liquid sample	None	_____	_____
966C	Hot leg sample	None	_____	_____
2083	SG "A" sample	None	_____	_____
2084	SG "B" sample	None	_____	_____
846	Accumulators N2 supply	None	_____	_____
1788	RCDT gas sample	None	_____	_____
1789	RCDT gas sample	None	_____	_____
1723	Sump "A" drain	Tape open	_____	_____
1728	Sump "A" drain	None	_____	_____
1721	RCDT outlet	None	_____	_____
1698	RCDT drain	Tape Open	_____	_____
1003A	RCDT pump suction	None	_____	_____

VALVE NO.	FUNCTION	SPECIAL INITIAL CONDITIONS	INITIALS	
			OPEN	CLOSED
1003B	RCDT pump suction	None		
1706	RCDT vent	None		
1787	RCDT vent	None		
2042	SG "A" blowdown	None		
2045	SG "B" blowdown	None		
313	RCP seal leakoff isolation	NOTE: IF IT IS NOT POSSIBLE TO CYCLE THIS VALVE AT THIS TIME, PLACE MARKER ON THE MANAGER'S TECHNICAL ASSISTANTS BOARD TO FLAG THE END OF THE TECH. SPEC. GRACE PERIOD.		
			INITIALS _____	
			DATE _____	
REMARKS:				

UNIT 1

Date _____

C 3
A 10
10-31-75LOSS OF AC WITH SAFETY INJECTION
CHECKOFF LIST

VALVE POSITION

FRONT OF COL

Desired Actual Initials

NOTE BRIGHT LIGHTS ON SI AND SPRAY
READY STATUS PANEL IN REMARKS.

LI-920	RWST level	>10%	_____	_____
ILI-106	Boric acid tank "A" level	>20	_____	_____
ILI-172	Boric acid tank "A" level	>20	_____	_____
W1A1	Containment accident fan control switch .	Auto	_____	_____
W1B1	Containment accident fan control switch .	Auto	_____	_____
W1C1	Containment accident fan control switch .	Auto	_____	_____
W1D1	Containment accident fan control switch .	Auto	_____	_____
MOV-851A	Containment sump isolation valve	S	_____	_____
MOV-851B	Containment sump isolation valve	S	_____	_____
MOV-825A	RWST to SI pump suction stop	Auto+S	_____	_____
MOV-825B	RWST to SI pump suction stop	Auto+S	_____	_____
MOV-826A	BAT to SI pump suction series stop	Auto+S	_____	_____
MOV-826B	BAT to SI pump suction parallel stop . . .	Auto+S	_____	_____
MOV-826C	BAT to SI pump suction parallel stop . . .	Auto+S	_____	_____
	Boric acid tank selector switch	T6A/T6C	_____	_____

MOV-871A	Spray pump suction from RHR	S	_____	_____
MOV-871B	Spray pump suction from RHR	S	_____	_____
MOV-841A	Accumulator "A" discharge stop	S	_____	_____
MOV-878D	Loop "A" SI isolation valve	S	_____	_____
MOV-878C	Vessel SI isolation valve	S	_____	_____
MOV-841B	Accumulator "B" discharge stop	S	_____	_____
MOV-878B	Loop "B" SI isolation valve	S	_____	_____
MOV-878A	Vessel SI isolation valve	S	_____	_____
MOV-866A	SI pump P15A discharge stop	S	_____	_____
MOV-866B	SI pump P15B discharge stop	S	_____	_____
P15A	SI pump "A" control switch	Auto	_____	_____
P15B	SI pump "B" control switch	Auto	_____	_____
MOV-852A	Core deluge stop valve	Auto+S	_____	_____
MOV-852B	Core deluge stop valve	Auto+S	_____	_____
MOV-896B	SI pump P15B isolation valve	0 - LH	_____	_____
MOV-896A	SI pump P15A isolation valve	0 - LH	_____	_____

REMARKS:

BACK OF CO1

		<u>Desired</u>	<u>Actual</u>	<u>Initials</u>
AOV-897A	RWST test line inlet stop valve	0		
AOV-897B	RWST test line inlet stop valve	0		
4100	Auxiliary building sump drain valve	0		
4101	Auxiliary building sump drain valve	0		
1LT-190	Level in boric acid tank "A"	>20		

CO3

P10A	Residual heat removal pump	Auto		
P10B	Residual heat removal pump	Auto		
MOV-700	RHR loop "A" stop valve	DSS		
MOV-701	RHR loop "A" stop valve	DSS		
MOV-720	RHR loop "B" stop valve	DSS		
HCV-625	RHR heat exchanger outlet	DSS		
HCV-624	RHR heat exchanger outlet	DSS		
FCV-626	RHR heat exchanger bypass	DSS		

"B" RCP COMPARTMENT

SI-V7	SI accumulator "B" discharge vent	S		
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ELEVATION 46'

MOV-120	RHR return to reactor coolant system	DSS		
RH-V3	Low head safety injection vent	S		
MOV-852A	Low head SI isolation core deluge	S		

ELEVATION 26'

MOV-701	Supply to RHR system	DSS		
RH-D9	RHR suction line drain	S		
1407	RHR suction line vent (Not in Unit 2)	S		
MOV-841A	SI accumulator "A" isolation	S		
SI-V9	Core deluge SI vent (Behind "B" acc.)	S		
MOV-878D	Loop "A" cold leg SI isolation	S		
MOV-878B	Loop "B" cold leg SI isolation	S		
MOV-878C	Core deluge SI isolation	S		
MOV-878A	Core deluge SI isolation	S		
MOV-841B	SI accumulator "B" isolation	S		
703	RHR header relief protection to letdown	0		
MOV-852B	Low head SI injection core deluge	S		

REMARKS:

<u>ELEVATION 8'</u>		<u>Desired</u>	<u>Actual</u>	<u>Initials</u>
RH-D10	Low head SI core deluge drain (not in Unit 2)	S & F		
1408	SI test line LVTC	S		
879A	SI test line isolation	S		

"A" STEAM GENERATOR COMPARTMENT

MOV-700	RHR system supply master	DSS		
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AUXILIARY BUILDING

SI-D1	RWST drain	S		
SI-893A	LI-921 root isolation	0		
SI-893B	LI-920 root isolation	0		
SI-896	RWST local sample	S		
SI-893C	RWST equalizing valve	S		
SI-897A	Test line return AOV	0		
SI-897B	Test line return AOV	0		
SI-884	Test line manual isolation	S		
RH-V4	Vent RHR "B" supply to SI and spray	S		
RH-711C	RHR pump "A" emergency seal water	S		
RH-711E	RHR pump "B" emergency seal water	S		
RH-74?	RHR return to RWST	S		
RH-890C	Disc relief for 742	0		
SI-856A	RHR pump "A" suction MOV from RWST	S		
SI-856B	RHR pump "B" suction MOV from RWST	S		
RH-714B	PCV-626 inlet isolation	0		
RH-715C	RHR to letdown crossconnect	DSS		
RH-712	RHR local sample	S		
RH-V1	RHR pump "A" recirc vent	S		
RH-704A	RHR pump "A" suction	0		
RH-704E	RHR pump "B" suction	0		
RH-V2	RHR pump "B" recirc vent	S		
SI-848E	FI-928 root isolation	0		
SI-848F	FI-928 root isolation	0		
SI-717A	FI-626 root isolation	0		
SI-717B	FI-626 root isolation	0		
RH-733B	RHR pump "B" recirc	0		
RH-733A	RHR pump "A" recirc	0		
RH-709A	RHR pump "A" discharge	0		
RH-713A	RHR pump discharge crossconnect	DSS		
RH-713B	RHR pump discharge crossconnect	DSS		
RH-709B	RHR pump "B" discharge	0		
RH-624	RHR heat exchanger "A" flow control valve	DSS		
RH-716A	PCV-624 outlet	0		
RH-D8	RHR heat exchanger discharge crossconnect drain	S		

REMARKS:

		Desired	Actual	Initials
RH-716D	RHR heat exchanger discharge crossconnect	DSS		
RH-716C	RHR heat exchanger discharge crossconnect	DSS		
RH-716B	FCV-625 outlet	0		
RH-625	RHR heat exchanger "B" flow control valve	DSS		
SI-851B	RHR pump "B" suction from Sump "B" . . .	S		
SI-851A	RHR pump "A" suction from Sump "B" . . .	S		
RH-D1	Discharge crossconnect drain	S		
RH-715A	RHR "A" heat exchanger inlet	0		
RH-D2	RHR "A" heat exchanger drain	S		
RH-D3	RHR "A" heat exchanger drain	S		
RH-D4	RHR "A" heat exchanger drain	S		
RH-715B	RHR "B" heat exchanger inlet	0		
RH-D5	RHR "B" heat exchanger drain	S		
RH-D6	RHR "B" heat exchanger drain	S		
RH-D7	RHR "B" heat exchanger drain	S		
RH-711A	PT-628 root isolation	0		
SI-823A	RHR pump "A" suction line drain	S		
RH-711D	RHR pump "A" seal leak tel-tale drain . .	S		
	Actuate "A" RHR pump compartment Hi			
	Level alarm	Alarm		
	Actuate "B" RHR pump compartment Hi			
	Level alarm	Alarm		
RH-711B	PT-629 root isolation	0		
SI-823B	RHR pump "B" suction line drain	S		
RH-711F	RHR pump "B" Seal leak tel-tale drain . .	S		
SI-826B	SI redundant boric acid suction MOV . . .	S		
SI-826C	SI redundant boric acid suction MOV . . .	S		
SI-826A	SI common boric acid suction MOV	S		
SI-894	MOV-825A & B Bypass	0		
SI-825A	SI redundant RWST suction MOV	S		
SI-825B	SI redundant RWST suction MOV	S		
SI-896A	SI pump "A" suction motor operated valve	0		
SI-896B	SI pump "B" suction motor operated valve	0		
SI-857A	RHR to SI pump "A" suction	S		
SI-857B	RHR to SI pump "B" suction	S		
SI-D8	SI pump common suction drain	S		
SI-881B	PT-922 root isolation	0		
SI-D9	SI pump "E" discharge drain	S		
SI-881A	PT-923 root isolation	0		
SI-D10	SI pump "A" discharge drain	S		
SI-876B	SI pump "B" recirc	0		
SI-876A	SI pump "A" recirc	0		

REMARKS:

		<u>Desired</u>	<u>Actual</u>	<u>Initials</u>
SI-831R	Eductor test valve from RMST	IC	_____	_____
SI-848A	FI-925 root isolation	0	_____	_____
SI-848B	FI-925 root isolation	0	_____	_____
SI-866A	SI pump "A" discharge header MOV	S - LH	_____	_____
SI-848C	FI-924 root isolation	0	_____	_____
SI-848D	FI-924 root isolation	0	_____	_____
SI-866B	SI pump "B" discharge header MOV	S - LH	_____	_____
SI-866A	Spray header isolation	S	_____	_____
SI-868B	Spray header isolation	S	_____	_____
SI-V4	Vent boric acid supply to SI pump suction	S	_____	_____
1SI-827A	Normal boric acid supply	S	_____	_____
1SI-827B	Alternate boric acid supply	S	_____	_____

REMARKS:

Check power on for the following equipment:

Initial

1.	P15A	A-05	_____
2.	P15B	A-05	_____
3.	MOV-825A	B-32	_____
4.	MOV-825B	B-42	_____
5.	MOV-826A	B-42	_____
6.	MOV-826B	B-42	_____
7.	MOV-826C	B-32	_____
8.	MOV-896A	B-32	_____
9.	MOV-896B	B-42	_____
10.	MOV-866A	B-32	_____
11.	MOV-866B	B-42	_____
12.	MOV-878B	B-42	_____
13.	MOV-878D	B-32	_____
14.	MOV-841A	B-32	_____
15.	MOV-841B	B-42	_____
16.	MOV-851A	B-32	_____
17.	MOV-851B	B-42	_____
18.	MOV-856A	B-32	_____
19.	MOV-856B	B-42	_____
20.	MOV-871A	B-32	_____
21.	MOV-871B	B-42	_____
22.	MOV-852A	B-32	_____
23.	MOV-852B	B-42	_____
24.	MOV-700	B-32	_____
25.	MOV-701	B-42	_____
26.	MOV-720	B-32	_____
27.	P10A	B-03	_____
28.	P10B	B-04	_____

Shift Supervisor _____ Time/Date _____

UNIT #1 HHSL
IPISA

100

25-2

[illegible]

0.5120

THE JACOB J. SHIMMING CO.

T: 30042

10-20-68 - (10)

671-12-0907

BYRON JACKSON

Unit #1
1P15B

HHST

ATTACHMENT 10 TO APPENDIX A

4-16-72 8th DMX

RPM

3570

NO. OF TESTS

1000000

DATE

10-15-72

TIME

15.5

FILED TO

D.M.

BYRON JACKSON TEST

1-30282

BYRON J. GIBSON

- PUMP TEST DATA

Unit 1
IPISA[illegible]

10-3-68

BYRON, JAC.

SKIPPING N

621-N-04-75

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Unit #,
IPISA

671-N-0476

 

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DATE: 10/10/2019

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UP 28
CV 44

FILED IN

I-79934-2

RHR IPIDA

UNIT 1

CONTRACTOR _____
 CUSTOMER WESTINGHOUSE ATOMIC POWER DIV _____
 ITEM NO _____ P.O. 54-E-70504-B _____
 IMPELLER PATTERN 5-2582 _____
 MAXIMUM DIAMETER 17 3/4" _____
 RATED DIAMETER 17 3/4" _____
 MINIMUM DIAMETER 14" _____

TEST PERFORMANCE CURVE NO. 33250C)

SIZE 6" L TYPE SVG STAGES 1
 RPM 1770 DATE 3-8-68
 PUMP NUMBER 43515
 PERFORMANCE ALSO APPLIES TO PUMP
 NUMBER _____

ATTACHMENT II, B. APPENDIX A

WEP ACAPRH

PNR
1P10B

WEP *unit* *Grainier Heat*
TEST PERFORMANCE CURVE NO 32250 D

CUSTOMER WESTINGHOUSE ATOMIC PWR DN
ITEM NO P.O. 54-E-70504-B
IMPELLER PATTERN S-2530
MAXIMUM DIAMETER 17 3/16
RATED DIAMETER 17 3/16
MINIMUM DIAMETER 17

SIZE 62 TYPE SVC STAGES 1
R.P.M. 1770 DATE 3-14-68
PUMP NUMBER 43516
PERFORMANCE ALSO APPLIES TO PUMP
NUMBER _____

PS

ATTACHMENT 12 TO APPENDIX 1

BYRON JACKSON HHSI

Unit #2
2PISA

PUMP NO. 4X6 C 8 ²⁷⁶ DVMX RP 1 3510	ASSEMBLY NO. FACTORY NO. 471-11-10-118	IMPL. DATE 10/15/64 DATE 10/15/64 DESIGNED BY JAK	UNLESS FILED TO 776	BYRON JACKSON TEST 1- 154
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- PUMP TEST DATA

Unit #2
2P15A[illegible]

BYRON JAC. & SHIPPING N.

71-N-0445

10254

BYR

JACKSON HST

Unit #2
LP15B

Hydrographer: M. J. J.
 Engineer: J. J. J.

PUMP SIZE AND TYPE

4 x 6 x 20 8000 DVMX

RPM

3510

PUMP MODEL NO.

1500102

PUMP SERIAL NO.

1500102

PUMP CURVE

1500102

DATE

1-12-99

SIGNATURE

S. P. S.

DATE

1-12-99

PUMP NO.

1500102

PUMP NAME

1500102

PUMP SERIAL NO.

1500102

PUMP CURVE

1500102

DATE

1-12-99

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S. P. S.

PUMP NO.

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SIGNATURE

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PUMP NO.

1500102

PUMP NAME

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PUMP SERIAL NO.

TESTED BY. S. P. S. D. K. AIA FOR D J

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CONTRACT NO.

CUSTOMER WESTINGHOUSE ATOMIC PWR DIVITEM NO. P.O. 54-5-70504-BIMPELLER PATTERN S-2580MAXIMUM DIAMETER 17 3/16MATED DIAMETER 17 3/16MINIMUM DIAMETER 14RHR
2 P10A

WIL - MAP 100

Residual Hgt. Pump
Unit 2TEST PERFORMANCE CURVE NO. 23200ESIZE 6" L TYPE SYC STAGES 1RPM 1770 DATE 3-20-68PUMP NUMBER A3517

PERFORMANCE ALSO APPLIES TO PUMP

NUMBER

PS

CONTRACT NO.

CUSTOMER WESTINGHOUSE ATOMIC POWER DIV

DESIGN NO. 54-E-7050A-B

IMPELLER PATTERN S-2580

MAXIMUM DIAMETER 17 3/16

RATED DIAMETER 17 3/16

MINIMUM DIAMETER 14

RHR
2P10B

WIS ALUMINUM
WIS RESIDUAL MOTOR
Unit 2
TEST PERFORMANCE CURVE NO. 33250 F

SIZE 6" L TYPE SVC STAGES 1

RPM 1770 DATE 3-14-68

PUMP NUMBER 13518

PERFORMANCE ALSO APPLIES TO PUMP

NUMBER

PS

PACIFIC PUMPS

STOCK PARK, CALIFORNIA

ATTENTION: 16 TO APPROVAL

Wisconsin Electric POWER COMPANY
231 WEST MICHIGAN, MILWA E, WISCONSIN 53201

April 18, 1977

Mr. Benard C. Rusche, Director
Office of Nuclear Reactor Reg.
U. S. NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555

Attention: Mr. George Lear, Chief
Operating Reactors Branch #3

Dear Mr. Rusche:

DOCKETS 50-266 AND 50-301
REACTOR VESSEL OVERPRESSURIZATION PROTECTION
UNITS 1 AND 2, POINT BEACH NUCLEAR PLANT

Your letter of April 1, 1977, reported that a pressure transient event has recently occurred on an operating PWR as a result of starting a reactor coolant pump (RCP) while in a water solid condition with the steam generator temperature higher than the reactor coolant system (RCS) temperature. The letter also expressed concern that the significance of such transients may not be realized and proposed that procedures be implemented immediately to prevent RCP starts or "jogs" unless necessary, and that the RCS temperature profile be determined so that appropriate measures could be taken to achieve an acceptable profile prior to starting a RCP. The following is provided in response to your request.

A review of our letters of December 20, 1976 and March 2, 1977, on the subject will verify that our current operations are in accordance with your latest request. Specific reference is made to Appendix C of our letter of March 2, 1977 in which the interim improvements already made to prevent overpressurization are described.

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77/150197

April 18, 1977

Operating procedures OP-3C, 4A and 1A were provided as attachments to Appendix A of the letter. These procedures provide for RCP starts only when necessary. Item 7 of Appendix A specifically addresses the problem of RCP starting while in the water solid configuration. You will note that a modification of step 2.17.1 of OP-3C, requiring comparison of the RCS and steam generator temperatures prior to starting RCP's was anticipated. This modification has been implemented. The procedural changes already made along with the interim improvements which have been installed and are operational provide assurance that RCP starts while in a water solid configuration will not cause overpressurization problems at Point Beach. Thus, in our view, we have already complied with your request.

We trust that the foregoing satisfies the concerns expressed in your April 1, 1977, letter.

Very truly yours,

A handwritten signature in black ink, appearing to read "Sol Burstein", with a long horizontal flourish extending to the right.

Sol Burstein

Executive Vice President

Docket Nos. 50-266
and 50-301

Wisconsin Michigan Power Company
Wisconsin Electric Power Company
ATTN: Mr. Sol Burstein
Executive Vice President
231 West Michigan Street
Milwaukee, Wisconsin 53201

Gentlemen:

RE: POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

A number of reported instances of reactor vessel overpressurization in Pressurized Water Reactor (PWR) facilities have occurred in which the Technical Specifications implementing 10 CFR Part 50 Appendix G limitations have been exceeded. The majority of cases have occurred during cold shutdown in which the primary system has been in water solid conditions. These overpressurization events have been initiated by a variety of causes, including the following:

- (1) Isolation of RHR system/letdown system while charging to a water solid primary system,
- (2) Thermal expansion following the starting of a primary coolant pump due to stored thermal energy in steam generators,
- (3) Inadvertent actuation of safety injection accumulators, and
- (4) Initiation of operation of a reactor coolant pump or a high pressure safety injection pump.

In essentially all of the events reported, a single personnel error, equipment malfunction or procedural deficiency has been sufficient to cause the event.

We believe that appropriate steps should be taken promptly by all PWR licensees to minimize the likelihood of additional occurrences of reactor vessel overpressurization. To that end we recently completed a series of meetings with several PWR licensees and NSSS suppliers in which we discussed the reported overpressurization events and assessed the measures that are currently being employed to either avoid or reduce the probability of similar occurrences, or to control the pressure transient to less than Appendix G limits. Examples of those measures identified by the various

licensees are as follows:

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SURNAME	JWetmore:esp	GZech	GLear		
DATE	8/11/76	8/11/76	8/11/76		

- (1) Complete avoidance of water solid conditions by either maintaining a pressurizer steam bubble or by providing a low pressure nitrogen blanket in the pressurizer when a steam bubble cannot be maintained.
- (2) Disabling High Pressure Injection and Safety Injection pumps by disconnecting electrical power supplies when at low primary system temperatures.
- (3) Installation of dual setpoint pressurizer tower relief valve(s) to provide protection against exceeding Appendix G limits while at low primary system temperatures.
- (4) Minimization of time at water solid conditions and upgrading plant procedures to include appropriate warnings and cautions when such operations are necessary, and
- (5) Installation of relief valves in charging pump discharge lines with a setpoint to provide protection against exceeding Appendix G limits.

It was noted in our discussions with the PWR licensees that, for the majority of those plants involved, not all potential overpressurization events would be prevented by the measures they had identified and that some of the remaining measures may have undesirable effects on reactor safety.

Based on the information gathered to date, we have concluded that all PWR licensees should evaluate their system designs to determine the susceptibility to overpressurization events. Specifically, you should provide the following:

- (1) An analysis of the Reactor Coolant System (RCS) response to pressure transients that can occur during startup and shutdown. Any design modifications determined to be necessary to preclude exceeding Appendix G limits are to be incorporated in this analysis. The analysis should include a plot of pressure as a function of time until termination of the event. The analysis should assume the most limiting initial conditions (e.g., one RRT train operating or available for letdown, other components in normal operation when the system is water solid such as pressurizer heaters and charging pumps, and one or more reactor coolant pumps in operation) with the worst single failure or operator error as the initiating event. Justification should be provided for the choice of limiting conditions and worst single failure or operator error assumed in the analysis.

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- (2) A description of those design modifications determined to be necessary, including equipment performance specifications and system operational sequences. The design basis used in the choice of equipment should be included, and
- (3) A schedule for the proper implementation of the proposed design modifications.

The basic criteria to be applied in determining the adequacy of over-pressurization protection are that no single equipment failure or single operator error will result in Appendix G limitations being exceeded.

For those situations in which the necessary design changes identified cannot be implemented within the next few months, you should identify short-term measures to reduce the likelihood that overpressurization events will occur in the interim period until the permanent design changes can be made. Short term measures should be identified separately for immediate implementation within the terms and conditions of your license. Short term measures might include some combination of, but would not be limited to, the following suggestions:

- (1) Procedural changes to minimize the time in which the primary system is in a water solid condition,
- (2) Upgrading existing plant procedures and administrative controls to assure that appropriate warnings and cautions are included to alert the operator whenever the potential for primary system overpressurization exists,
- (3) Provide alarms and/or indications to alert the operator whenever primary system pressure increases toward Appendix G limits,
- (4) Introducing temporary plant modifications for pressure relief, and
- (5) Assignment of additional personnel to monitor plant operations when water solid.

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Wisconsin Michigan Power Company
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Modifications to preclude or minimize the probability of reactor vessel overpressurization events are plant dependent and the examples given may or may not be adaptable to your specific system design. Consideration must also be given to the potential effects of both the short term and long term measures you consider to assure that other aspects of nuclear safety are not compromised.

To verify compliance with Appendix G pressure-temperature limits during startup and shutdown, you should assure that the appropriate instrumentation is installed to provide a continuous permanent record over the full range of both pressure and temperature. This instrumentation should be in service during long periods of cold shutdown as well as during startup and shutdown operations. Reliance upon the plant computer to reconstruct a pressure transient is not considered sufficient because of the likelihood of computer downtime especially during plant shutdown conditions.

We request that you notify us within 20 days after receipt of this letter that you will provide all the information requested within 60 days or explain why you cannot meet this schedule and provide the schedule that you will meet.

This request for generic information was approved by GAO under a blanket clearance number B-150225 (R0072); this clearance expires July 31, 1977.

Sincerely,

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

cc: Mr. Bruce Churchill, Esquire
Shaw, Pittman, Potts and Trowbridge
1800 M. Street, N.W.
Washington, D. C. 20036

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