SELECTED ISSUES PROGRAM

TECHNICAL EVALUATION OF THE
ELECTRICAL, INSTRUMENTATION, AND CONTROL DESIGN ASPECTS
OF THE LOW TEMPERATURE OVERPRESSURE PROTECTION SYSTEM
FOR THE HADDAM NECK NUCLEAR POWER PLANT

by

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March 1979

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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 Haddam Neck nuclear power plant. 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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1. INTRODUCTION

By letter to the Connecticut Yankee Atomic Power Company (CYAPCO) dated August 11, 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 September 3, 1976 and subsequent letters (refer to the Appendix), the Connecticut Yankee Atomic 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 Haddam Neck 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 HADDAM NECK

2.1 INTRODUCTION

Review of the Haddam Neck low temperature overpressure protection system design by CYAPCO 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 newly installed spring-loaded relief valves (SLRV's).

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 as follows:

- (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 CYAPCO OVERPRESSURE PROTECTION SYSTEM DESIGN

The CYAPCO OPS design information detailed in this section was derived from Reference 19 in the Appendix. The CYAPCO design for the Haddam Neck OPS is based on the use of two new SLRV's which are isolated from the pressurizer/reactor coolant system during normal operating conditions by two new motor-operated isolation valves. In conjunction with specific procedural controls, the following requirements are defined:

- (1) Each SLRV will provide sufficient and redundant relief capacity to ensure that the reactor coolant system (RCS) pressure remains within the operating limits. The SLRV low pressure setpoint will be 380 psig.
- (2) Each isolation motor-operated valve (MOV) will be controlled by an independent control switch located on the reactor control board in the main control room.
- (3) A 340°F-temperature interlock and alarm will be provided to ensure SLRV system lineup at RCS temperatures below 340°F. The pressure and temperature limits used for the operation of the mitigating system are based on the projected 14-year operating limit curve, relief valve accumulation, and the objective to retain operational flexibility to the maximum extent practical.
- (4) Additional assurance of preventing inadvertent blow-down at RCS temperatures above 340°F is provided by the inclusion of a 380-psig pressure permissive signal which prevents opening of the isolation MOV's during normal reactor operating conditions and in the high pressure ranges of RCS heatup and cooldown.

The control circuits for the two isolation MOV's in each relief train utilize independent instrument loops and control logic. The two pressure-sensing instrument loops are P-403 and P-404, one for each relief train. The two temperature-sensing loops are T-423 and T-433. The control circuits are separated from one another as much as physical limitations permit. To ensure that the SLRV's are in direct contact with the RCS at the required pressure and temperature, a basic sequence which combines operator action and warning alarms has been incorporated into the isolation MOV circuitry logic. A detailed discussion of these actions and corresponding alarms follows.

In accordance with plant cooldown procedures, both the RCS pressure and temperature are decreased uniformly down to approximately 300°F and 350-400 psig. Prior to cooling the RCS below 340°F, operating procedures will require activation of the OPS by setting the isolation MOV switches to the "open" position. To ensure OPS enabling, a combination of audio/visual alarms will be initiated when the RCS temperature is equal to or less than 340°F and the pressure is equal to or less than 380 psig.

Alarm cancellation requires operator acknowledgment by placing the hand switches in the "open" position. The isolation MOV's cannot be closed at temperatures below 340°F in order to prevent inadvertent valve closures. The only valve closures that will be permitted will be for RCS hydrostatic tests. This interlock assures relief system availability when the RCS is susceptible to low temperature pressure transients. In addition, the 380-psig pressure interlock prevents opening of the isolation MOV's at pressures which would result in unnecessary reactor coolant discharge.

During plant startup, operating procedures will maintain the RCS pressure below 380 psig until the RCS temperature is greater than 340°F. When the RCS temperature exceeds 340°F, operating procedures will require that the overpressurization system be taken out of service (i.e., that the two isolation MOV's be closed). To ensure system isolation once the RCS temperature increases above 340°F, a combination of audio/visual alarms will be activated until the operator cancels the alarms by placing the switches in the "closed" position. Once the relief system is isolated, plant startup will continue accordingly.

2.3 EVALUATION OF HADDAM NECK USING DESIGN BASIS CRITERIA

Haddam Neck was 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 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

The CYAPCO Haddam Neck single failure criterion information detailed in this section was derived from Reference 19. The CYAPCO Haddam Neck OPS 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.

Compliance with the single failure criterion is achieved by providing separate sensors, actuating mechanisms, and two full-capacity SLRV's used in conjunction with procedural and administrative controls described below.

The design basis transients for the proposed RCS overpressure protection system are:

- (1) Energy Addition. Reactor coolant pump (RCP) start with a secondary-to-primary temperature differential of 50°F and a solid water RCS system.
- (2) Mass Addition. Inadvertent high-pressure safety injector pump (HPSIP) actuation and injection into a solid water RCS system.
- (3) Mass Addition. Inadvertent charging pump actuation and injection into a water-solid RCS system with letdown isolated.

Consistent with the single failure criterion, the means of protection for design basis transients are as follows:

- (1) For water-solid RCS conditions, RCS operation will be initiated only when the secondary-to-primary temperature differential is less than 50°F. This temperature limitation will ensure that one SLRV has sufficient discharge capability to mitigate the design transient of the generic analysis and, therefore, to maintain pressure within the low temperature pressure limits of the Haddam Neck projected 14-year operating limit curves. However, it is noted that the administrative limit for a secondary-to-primary temperature differential is 0° to 20°F in order to avoid potential primary coolant discharges (i.e., the secondary side must be equal to or colder than the primary side).
- (2) A single SLRV has sufficient capacity to maintain the primary system pressure within the 14-year operating limits curve due to inadvertent charging pump start.
- (3) By procedural and administrative controls, the HPSIP's and the associated discharge valves will be disabled prior to decreasing the primary system temperature below 340°F. This will eliminate any potential for primary coolant blowdown due to inadvertent HPSI or safety injection signal actuation.

We conclude that the CYAPCO Haddam Neck OPS satisfies the NRC staff single failure criterion.

2.3.3 System Testability

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

CYAPCO states in Reference 19 that adequate testing is considered to be confirmation that an input signal to the isolation MOV control initiates valve opening. A channel functional test of the associated instrumentation and control hardware will be conducted once per cold shutdown to confirm the design logic. Valve testing and frequency will be conducted consistent with the applicable requirements of the ASME Boiler and Pressure Vessel Code, Section XI.

Testing of the SLRV's once each cold shutdown can be accomplished by either removing the SLRV's and testing them on a hydrostatic test rig, or by a controlled RCS pressure increase above 380 psig when the RCS temperature is between 300°F and 340°F. This temperature requirement provides the greatest pressure range between SLRV setpoint and curve limitations (refer to 10 CFR 50, Appendix G).

The safety evaluation report (SER), dated May 1978, by the NRC Reactor Safety Branch/Division of Operating Reactors (RSB/DOR) for the Haddam Neck OPS states that:

- (1) Testability will be provided.
- (2) The Licensee has stated that the four OPS MOV's will be mechanically tested in accordance with the requirements specified in Section XI of the ASME code, and will be electrically tested by confirming proper motor and valve movement in response to an input signal (e.g., opening or closing). A channel functional test associated with the MOV interlocks and controls will be conducted once per refueling shutdown. The SLRV's setpoint will also be verified each refueling outage either by a bench test (removal of the SLRV for test at a testing facility), or by an in-place test (done by pressurizing the RCS up to the SLRV setpoint with alternate sets of MOV's open so that each SLRV can be checked).
- (3) The Licensee's testing requirements are further clarified in the Technical Specifications proposed in their Reference 15 submittal and discussed in Section 5.2 herein, and are acceptable.

We conclude that the CYAPCO Haddam Neck OPS satisfies the NRC staff testability criteria.

2.3.4 Seismic Design and IEEE Std-279-1971 Criteria

CPAPCO states in Reference 19 that seismic Category I and IEEE Std-279-1971 criteria were considered in the design of the OPS. The SLRV's and the 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:

- (1) IEEE Std-279-1971 criteria will be implemented within the limitations of the original plant design and construction philosophy. The control circuits for each valve train will be independent of each other.
- (2) The isolation MOV's and SLRV's were designed and manufactured in accordance with the ASME Boiler and Pressure Vessel Code, Section III (1972 Edition). Isolation MOV's are classified as Class 1 valves and the relief valves are classified as Class 2. The subject valves were designed to be capable of operating during and after a seismic acceleration of 3.0 g in any direction. Therefore, it is concluded that each valve assembly has been designed for Category I seismic design conditions.
- (3) The electrical and control circuitry installation for the isolation MOV's are consistent with the requirements of the original plant design philosophy.

We conclude that the CYAPCO Haddam Neck OPS satisfies the NRC staff seismic design and IEEE Std-279-1971 criteria.

2.4 ALARM SYSTEMS DESIGNS AND OPERATION

Specific details concerning alarm systems design and operation for the OPS 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.

CYAPCO states in Reference 19 that the high-pressure alarm system design is as follows:

- (1) An overpressure-transient alarm is installed to alert the operator whenever RCS pressure approaches the operating limit curves described in Appendix G, 10 CFR 50.

- (2) The overpressure transient alarm operates from new bistables and annunciates on the main control board when the reactor coolant pressure exceeds 400 psig and only when the isolation MOV hand switches are in the "open" position (i.e., when the RCS temperature is below 340°F).
- (3) The pressure signals PT-403 and PT-404 are powered from a 480-Vac source capable of being energized from either of the emergency diesel generators upon loss of normal power.

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.

CYAPCO states in Reference 19 that the isolation valve alarm system design is as follows:

- (1) The two MOV isolation valves are wired into the RCS OPS such that the hand-switch activation of the protection system results in the opening of the isolation valves.
- (2) An open-close indicator for each isolation valve is provided 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.

2.4.3 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 SLRV "low" set-point is activated before the RCS temperature is equal to or less than 340°F.

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

In accordance with plant cooldown procedures discussed in Reference 19, both the RCS pressure and temperature are decreased uniformly down to approximately 300°F and 350-400 psig. Prior to cooling the RCS below 340°F, operating procedures will require the activation of the OPS by setting the isolation MOV switches to the "open" position.

CYAPCO states in Reference 19 that the enable alarm system design is as follows:

- (1) A combination of audio/visual alarms will be initiated to ensure system activation when the RCS temperature is equal to or less than 340°F and the pressure is equal to or less than 380 psig.
- (2) Alarm cancellation requires operator acknowledgment by placing the hand switches in the "open" position.
- (3) The isolation MOV's cannot be closed at temperatures below 340°F in order to prevent inadvertent valve closures. The only valve closures that will be permitted will be for RCS hydrostatic tests. This interlock ensures relief system availability when the RCS is susceptible to low temperature pressure transients. In addition, the 380-psig pressure interlock prevents opening of the isolation MOV's at pressures which could result in unnecessary reactor coolant discharge.

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 SLRV's are reset to the "high" setpoint when the RCS temperature is greater than 340°F.
- (2) The alarm shall be of the audio/visual type and provide unambiguous information to the operator.

CYAPCO states in Reference 19 that the disable alarm system design is as follows:

- (1) During plant heatup, operating procedures will maintain the RCS pressure below 380 psig until the RCS temperature is greater than 340°F. When the RCS temperature exceeds 340°F, operating procedures will require that the OPS be taken out of service (i.e., when the two isolation MOV's are closed).

- (2) To ensure system isolation once the PCS temperature increases above 340°F, a combination of audio/visual alarms will be initiated until the operator cancels the alarms by placing the switches in the "closed" position. Once the relief system is isolated, plant heatup will continue accordingly.

We conclude that this design satisfies the NRC staff position.

2.4.5 SLRV Open Alarm

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

The CYAPCO docketed material makes no reference to an SLRV "open" alarm system.

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 OPS 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.

CYAPCO states in Reference 19 that appropriate instrumentation and recording equipment exists at the Haddam Neck Plant which will provide a continuous and permanent record over the full range of primary system pressure and temperature. The sensing and recording equipment will be in service during startup and shutdown operations as well as during long periods of cold shutdown operations.

We conclude that this implementation 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.

CYAPCO states in Reference 19 that, in general, any component capable of mass or energy input to the RCS would be disabled when its operation is nonessential to plant operation or safety considerations and when its input would result in RCS overpressurization above the minimum operating limit curves. A description of the disabling of essential components not required during cold shutdown follows:

- (1) The start of a reactor coolant pump is prevented with a water-solid system, unless the steam generator secondary-side temperature is equal to or less than 20°F colder than the primary-side temperature.
- (2) The operational procedure governing the SIS startup and shutdown requires that (a) the high-pressure and low-pressure SI pumps are disabled electrically, (b) all valves between the pump and main coolant system are closed, locked or deenergized, and "red tagged", and (c) the electrical supply to the circuitry that operates the pumps and electrically-controlled valves is disconnected. These actions are all performed with plant conditions between 1000 psig/300°F and 500 psig/260°F. This lineup is left intact until reactor startup when it is put back in service at the same plant conditions. The SI pumps are tested every year per surveillance procedures; however, the valves that isolate the system are locked closed during these tests.

CYAPCO includes these procedural limitations as a Technical Specification requirement for low temperature/low pressure operation.

- (3) The pressurizer heaters and charging pumps will be disabled whenever they are not required during solid operations.
- (4) The steam generator secondary-side temperature will be maintained no more than 50°F above the primary coolant temperature when the RCS is in a water-solid condition. This requirement is based on the operability of one SLRV, consistent with the single failure criterion and the design basis criteria.
- (5) Operating procedures administratively prevent starting of an RCP with a water-solid system, unless the steam generator secondary-side temperature is equal to or less than 20°F colder than the primary-side temperature. This temperature differential limitation prevents unnecessary primary coolant discharge from the SLRV's.

- (6) A combination of administrative and procedural controls and SLRV capability will provide the necessary overpressure transient protection for the worst-case design basis transients.

We conclude that this implementation 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 Protection System for Haddam Neck", dated May 1978.

To ensure operation of the OPS, the Licensee has submitted for NRC staff review its proposed Technical Specifications for incorporation into the license for Haddam Neck (refer to the Appendix, Reference 15). These specifications are summarized below:

- (1) The Haddam Neck OPS must be operable whenever the RCS temperature is below 340°F. Operability of the OPS requires that all four isolation MOV's connecting the two SLRV's to the pressurizer are open, and that the SLRV's setpoint is 380 psig. The OPS need not be operable if the RCS is depressurized and vented to containment by a three-inch o.d. opening. If these conditions cannot be met, the RCS shall be depressurized and vented in 24 hours and the inoperable train(s) shall be restored prior to system pressurization.
- (2) If the RCS is in hot standby, startup, or power operation, and an OPS train is discovered to be inoperable, it shall be repaired within seven days (or the NRC staff shall be notified with an estimation of the repair date), and the RCS shall not be cooled down and depressurized to a cold or hot shutdown mode unless personnel safety, plant safety, or other Technical Specification mandate such action.
- (3) When starting an RCP with the RCS temperature below 340°F, the secondary side of the steam generator in that loop shall be less than 20°F hotter than the primary side.
- (4) Whenever the RCS is below 340°F and is not vented by a minimum opening of three inches (o.d.), the HPSIP's shall be deenergized by racking out their power supply breakers, locking closed the breaker cabinet door, and locking closed the HPSIP discharge valves.

- (5) During the core cooling system periodic tests, the RCS shall be vented by a minimum opening of at least three inches (o.d.) with the OPS operable, or by two openings, each a minimum of three inches (o.d.) if both OPS trains are inoperable.

The NRC staff has reviewed the Licensee's proposed Technical Specifications described above and concluded that they are acceptable based on the analyses and methods described.

4. CONCLUSIONS

The electrical, instrumentation, and control (EI&C) design aspects of the low temperature overpressure protection system (OPS) for Haddam Neck 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 CYAPCO Haddam Neck OPS 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
- (7) System testability
- (8) Pressure transient reporting and recording requirements
- (9) Disabling of essential components not required during cold shutdown.

We recommend that the NRC staff find the following EI&C aspects of the CYAPCO Haddam Neck OPS 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. NRC (Schwencer) letter to CYAPCO (Switzer) dated August 11, 1976.
- * 2. CYAPCO (Switzer) letter to NRC (Schwencer) dated September 3, 1976.
- * 3. CYAPCO (Switzer) letter to NRC (Schwencer) dated October 15, 1976.
- * 4. CYAPCO (Switzer) letter to NRC (Schwencer) dated December 3, 1976.
- * 5. NRC (Schwencer) letter to CYAPCO (Switzer) dated January 10, 1977.
- * 6. NRC (Schwencer) letter to CYAPCO (Switzer) dated February 14, 1977.
- * 7. CYAPCO (Switzer) letter to NRC (Schwencer) dated March 1, 1977.
- * 8. CYAPCO (Switzer) letter to NRC (Schwencer) dated March 21, 1977.
- * 9. NRC (Schwencer) letter to CYAPCO (Switzer) dated April 1, 1977.
- * 10. CYAPCO (Switzer) letter to NRC (Schwencer) dated April 26, 1977.
- * 11. CYAPCO (Switzer) letter to NRC (Schwencer) dated June 1, 1977.
- * 12. CYAPCO (Switzer) letter to NRC (Schwencer) dated September 7, 1977.
- * 13. NRC (Schwencer) letter to CYAPCO (Switzer) dated November 1, 1977.
- * 14. CYAPCO (Switzer) letter to NRC (Schwencer) dated November 30, 1977.
- * 15. CYAPCO (Switzer) letter to NRC (Schwencer) dated January 3, 1978.
- * 16. CYAPCO (Switzer) letter to NRC (Schwencer) dated March 6, 1978.
- 17. "Staff Discussion of Fifteen Technical Issues Listed in Attachment G, November 3, 1976 Memorandum from Director NRR to NRR Staff," NUREG-0138, November 1976.
- 18. "Pressure Mitigating System Transient Analysis Results," prepared by Westinghouse for the Westinghouse User's Group on Reactor Coolant System Overpressurization, July 1977 (submitted as Attachment 1 to reference 12 above).
- 19. "Specific Plant Report, Low Temperature RCS Overpressure Protection for Connecticut Yankee," August 1977 (submitted as Attachment 2 to Reference 2 above).

*Copy included in this report.

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Connecticut Yankee Atomic Power Company
ATTN: Mr. Donald C. Switzer
President
Post Office Box 270
Hartford, Connecticut 06101

Gentlemen:

RE: Haddam Neck

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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- (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 power 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 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 RHR 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

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- (2) 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 prompt implementation of the proposed design modifications.

The basic criteria to be applied in determining the adequacy of overpressurization 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.

OFFICE ➤						
SURNAME ➤						
DATE ➤						

AUG 11 1976

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-180225 (R0072); this clearance expires July 31, 1977.

Sincerely,

Original signed by

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

cc: Day, Berry & Howard
Counselors at Law
One Constitution Plaza
Hartford, Connecticut 06103

Russell Library
119 Broad Street
Haddam, Connecticut 06457

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Regulatory

REG-1

CONNECTICUT YANKEE ATOMIC POWER COMPANY

BERLIN, CONNECTICUT

P. O. BOX 270 HARTFORD, CONNECTICUT 06101

September 3, 1976

Docket No. 50-213

Director of Nuclear Reactor Regulation
 Attn: Mr. A. Schwencer, Chief
 Operating Reactors Branch #1
 U. S. Nuclear Regulatory Commission
 Washington, D. C. 20555

Reference: (1) A. Schwencer letter to D. C. Switzer, dated
 August 11, 1976

Gentlemen:

Haddam Neck Plant
Reactor Vessel Overpressurization

In Reference (1), CYAPCO was requested to evaluate system designs to determine the susceptibility to overpressurization events and to notify you within 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 CYAPCO is a participant, to examine the consequences of overpressurization events and to identify similarities between Westinghouse plants for the purpose of determining a generic solution to this issue. The task group will meet with Westinghouse for the purpose 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 their plant's operating procedures and administrative controls during the 60 day period referenced in Reference (1). The purpose of this examination will be to evaluate

procedural changes that will minimize the likelihood of overpressurization events. As an interim measure, the plant operating personnel will be informed of the potential for overpressurization and will be cautioned to minimize conditions which could cause such an event should reduced pressure and temperature operation occur before the detailed review is completed.

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.

Very truly yours,

CONNECTICUT YANKEE ATOMIC POWER COMPANY

D. C. Switzer
President

CONNECTICUT YANKEE ATOMIC POWER COMPANY

BERLIN, CONNECTICUT

P. O. BOX 270 HARTFORD, CONNECTICUT 06101

TELEPHONE
303-866-8000

REGULATORY DOCKET FILE COPY

October 15, 1976

Docket No. 50-213

Director of Nuclear Reactor Regulation
 Attn: Mr. A. Schwencer, Chief
 Operating Reactors Branch #1
 U. S. Nuclear Regulatory Commission
 Washington, D. C. 20555

- References: (1) A. Schwencer letter to D. C. Switzer, dated August 11, 1976.
 (2) D. C. Switzer letter to A. Schwencer, dated September 3, 1976.

Gentlemen:

Haddam Neck Plant
Reactor Coolant System (RCS) Overpressurization

In Reference (1), CYAPCO was requested to evaluate system designs to determine the susceptibility to RCS overpressurization events and to notify the NRC Staff as to the schedule of providing the requested information. In Reference (2), CYAPCO notified the NRC Staff that an owner's group of utilities with Westinghouse plants had been formed for the purpose of identifying the generic implications of this issue. Additionally, each utility member of the group was to examine its plant's operating procedures and administrative controls with the objective of instituting short-term measures to minimize the potential for RCS overpressurization. In Reference (2), CYAPCO stated that a report of the status of the group's efforts and the results of the evaluation of short-term measures would be transmitted to the NRC Staff. In accordance with that commitment, CYAPCO hereby provides the following information.

Meetings between the utility group and the Westinghouse Electric Corporation were held during September, 1976 for the purposes of categorizing all postulated RCS overpressurization initiating events, discussing the details of analytical models, and determining a schedule for completion of the postulated worst-case RCS transient analysis. As a result of these meetings, it has been decided to perform a bounding RCS transient analysis, such that its results are

applicable to all plants within the utility group. The transient analysis will consider overpressurization events induced by both mass and heat input into the RCS considering a single active failure or operator error. In addition to a group subcommittee charged with working with Westinghouse for the purpose of defining the input and assumptions to be used in the analyses, another subcommittee has been formed to review topics associated with the RCS transient analyses.

Based on the review to date of the analytical effort, it is expected that the RCS transient analyses will be completed on or about May 2, 1977. At that time, a description of necessary design changes and a schedule for implementation will be developed and provided to the NRC Staff.

With regard to plant operating procedures, administrative controls, and other short-term measures, CYAPCO has instituted a number of changes to minimize the possibility of RCS overpressurization.

All plant operators were informed of the generic RCS overpressurization events and potential initiating events. This was accomplished by review of References (1) and (2) and other pertinent documents. In addition, the operators have been alerted as to the importance of the 10CFR50 Appendix G limit curves during various operational modes.

Review of plant operating procedures and administrative controls will result in revising procedures relating to plant startup and shutdown, filling and venting, and associated plant tests. The revisions will be completed by November 1, 1976 and are aimed at further assuring that the appropriate cautions and warnings are included in procedures, whenever the potential for RCS overpressurization exists, and at minimizing the time in which the plant operates water solid. Potential situations were derived from review of previous RCS overpressurization events and discussions with Westinghouse and other utilities. Another conclusion of the review of operating procedures was that existing procedures already provide for protection in precluding RCS overpressurization.

Finally, CYAPCO is currently evaluating the feasibility of providing a computer-generated alarm and/or indications on an interim basis to alert the operator whenever the RCS conditions approach the applicable Appendix G limitations. On a permanent basis, the installation of instrumentation to provide a continuous permanent record of RCS pressure and temperature is also being evaluated.

CYAPCO believes that existing procedures, together with the course of action outlined above, will adequately assure that the likelihood of RCS overpressurization events is minimized or precluded.

Very truly yours,

CONNECTICUT YANKEE ATOMIC POWER COMPANY

D. C. Switzer
President

CONNECTICUT YANKEE ATOMIC POWER COMPANY

BERLIN, CONNECTICUT

P. O. BOX 270 HARTFORD, CONNECTICUT 06101

TELEPHONE
203-668-6011

December 3, 1976

Docket No. 50-213

Director of Nuclear Reactor Regulation
 Attn: Mr. A. Schwenker, Chief
 Operating Reactors Branch #1
 U. S. Nuclear Regulatory Commission
 Washington, D. C. 20545

References: (1) A. Schwencer letter to D. C. Switzer, dated August 11, 1976.
 (2) D. C. Switzer letter to A. Schwencer, dated September 3, 1976.
 (3) D. C. Switzer letter to A. Schwencer, dated October 15, 1976.

Gentlemen:

Haddam Neck Plant
Reactor Coolant System (RCS) Overpressurization

In Reference (1), CYAPCO was requested to evaluate system designs to determine the susceptibility to RCS overpressurization events and to notify the NRC Staff as to the schedule of providing the requested information. In Reference (2), CYAPCO notified the NRC Staff that an owner's group of utilities with Westinghouse plants had been formed for the purpose of identifying the generic implications of this issue. Additionally, each utility member of the group was to examine its plant's operating procedures and administrative controls with the objective of instituting short-term measures to minimize the potential for RCS overpressurization. In Reference (2), CYAPCO stated that a report of the status of the group's efforts and the results of the evaluation of short-term measures would be transmitted to the NRC Staff. In accordance with that commitment, CYAPCO informed the NRC Staff in Reference (3) of the schedule for RCS transient analyses and the short-term administrative and procedural measures implemented. CYAPCO also noted that it is our belief that existing procedures, together with the course of action outlined above, will adequately assure that the likelihood of RCS overpressurization events is minimized or precluded.

In response to the NRC Staff request at a meeting on November 4, 1976, this letter is intended to provide additional clarification of the proposed course of action and design criteria for the intended pressure relief system. Based on the results of preliminary evaluations, CYAPCO is planning to utilize power

12487

operate relief valves (PORV), as the pressure relief mechanism, to preclude RCS overpressurization. The detailed design of valve control system is not yet available; nevertheless, it can be presently stated that the design of the control system will be affected by several significant factors: first, the control system will be designed to assure that the Appendix G limit curves are not exceeded; second, the system must be able to provide as much flexibility of operation as possible at low pressure and temperatures, in light of existing Connecticut Yankee design and operating characteristics (e.g. maintenance of minimum RCP seal pressure of 375 psig); and, finally, the resolution of the RCS overpressurization issue must be on a schedule consistent with that already established for the complete replacement of the existing PORV and conversion from a solenoid to air control system, an extensive effort which began shortly after the summer, 1976 refueling outage. This is expected to be completed during the fall, 1977 refueling outage, contingent upon material availability.

These factors have led CYAPCO to give consideration to a PORV modulation-type control system, which will be programmed to follow the Appendix G limit curve throughout the appropriate temperature range.

A thorough analysis of overpressurization transient events employing the LOFTRAN computer code is also currently in progress to confirm the adequacy of relief systems. To utilize the LOFTRAN code, internal revisions are necessary; this will require a development and verification effort. The modified LOFTRAN calculational model, when complete, will provide a technically justifiable and conservative means to determine the adequacy of a relief valve system to mitigate the RCS overpressurization event. Until such time as the calculational model is completed and bounding analyses are performed, exact size requirements for the pressure relief system cannot be determined.

While relief capacity requirements of the relief system are not known at present, meaningful progress towards resolution of the RCS overpressurization issue has been effected by defining the following design criteria requirements of the relief system.

1. Operator Action - Credit for operator action will not be taken until 10 minutes after the initiation of the RCS overpressurization event.
2. Single Failure/Testability - The design basis shall be such that a single failure will not initiate a loss of coolant accident, when the system is in a passive mode. In order to verify that the relief system is operable, the logic will be designed to be testable, with surveillance of the PORV actuation/control system 30 days prior to planned use.
3. Common Equipment Failure (CEF) - CEF will be considered in the relief system design to the extent that the single failure which initiated the overpressurization event will not disable the relief system. Consideration will be given to events such as loss of offsite power, loss of instrument air, and a seismic event.

4. Electrical Criteria - The electrical design of the relief system will be consistent with existing plant criteria. Further, the design will consider electrical separation in light of Item #3 above. Since a single failure is assumed to cause the initiation of the RCS overpressurization, an additional single failure will not be considered as part of the actuation system design.
5. Seismic Criteria - The seismic design of the relief system will be consistent with existing plant criteria and Item #3 above.

As noted above, the current schedule for replacement of the existing PORV's calls for installation during the fall, 1977 refueling outage. Discussions with valve vendors have indicated at least a 10 month lead time; lead times for the relief system controllers are of the order of 16-25 weeks. Therefore, procurement of PORV's for Connecticut Yankee has already been initiated, while procurement of the control/actuation system will await appropriate regulatory concurrences. Additionally, as noted in Reference (3), detailed RCS transient analyses will be available in May, 1977. Thus, in order to avoid equipment procurement delays and to meet the desired installation schedule, timely NRC Staff review of this course of action and design criteria is requested.

It remains the belief of CYAPCO that existing procedures, together with the course of action outlined in Reference (3) and above, will adequately assure that the likelihood of RCS overpressurization events at the Haddam Neck Plant is minimized or precluded.

Very truly yours,

CONNECTICUT YANKEE ATOMIC POWER COMPANY

D. C. Switzer
President

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

January 10, 1977

Docket No. 50-213

Connecticut Yankee Atomic Power Company
ATTN: Mr. Donald C. Switzer
President
Post Office Box 270
Hartford, Connecticut 06101

Gentlemen:

RE: HADDAM NECK PLANT

In August of this year we sent letters to you and the licensees of other operating PWR facilities which expressed our concern over the number of reported instances of reactor vessel overpressurization. We requested that an analysis be provided of the reactor coolant system (RCS) response to pressure transients and that any design modifications be identified that were determined to be necessary to preclude exceeding the limits of Appendix G to 10 CFR Part 50. In addition, we requested that if those design changes could not be made within a few months, interim measures should be implemented immediately to reduce the likelihood of overpressurization events until the permanent design changes can be made.

In your letter of October 15, 1976, you identified the interim measures that were being implemented at your facility. We have completed our review of your submittal and have concluded that additional information will be required for us to determine the adequacy of these interim measures. In addition, our review of the responses from other PWR licensees has identified certain measures which, in our opinion, are significant enough to be required in most PWR facilities. These measures have been identified as staff positions and are included with the additional information requests in the enclosure to this letter. In addition, we have concluded that the procedural and administrative measures you have already instituted will help prevent any future pressure transients and should be continued even after long term hardware improvements are made unless you can demonstrate that this would not be justified.

With regard to your submittal of December 3, 1976, in which a description and schedule information related to your "Reference Mitigating System" for long-term overpressure protection were provided, we find your proposed schedule to be unacceptable and your proposed course of action to be lacking with respect to meeting the design criteria detailed at our November 4, 1976 meeting (meeting summary attached). We called the November meeting to inform all PWR licensees of the specific design criteria that would be required. The response in your December letter is based on your interpretation of the preliminary criteria issued in August and does not adequately address the final criteria as detailed at the November meeting. The overpressure protection system you have proposed, based on your interpretation of the preliminary criteria may be sufficient as an interim remedy, but will not

be acceptable for the long term until the final criteria are addressed and any deviations justified. In addition, the extended schedule you have proposed for analysis and acquisition of hardware will not meet the objective of implementation of improved overpressure protection in all operating PWR facilities by the end of 1977. Therefore, to meet this objective you should select either of the following two implementation options:

1. Commitment to a schedule for the installation of acceptable long-term improvements by December 31, 1977, which meet all the design criteria discussed at the November 4, 1976 meeting held in Bethesda, or
2. Commitment to a schedule to achieve installation of interim hardware improvements by December 31, 1977, and installation of long term improvements that meet all the design criteria discussed at the November meeting during the first scheduled shutdown after December 31, 1977. The dual setpoint system of pressurizer relief valves described in the Westinghouse "Reference Mitigating System" design is an acceptable interim hardware improvement.

The long term hardware improvements should either meet the design criteria discussed at the November 4, 1976 meeting; or, where deviations from the criteria are proposed, a detailed justification should be provided. Interim improvements need not meet all the design criteria discussed at the November meeting, but must represent good engineering practice and must not adversely affect plant safety or introduce potential common mode failures that could both cause the overpressure event and disable the protection system. For example, if air operated dual setpoint pressurizer relief valves are used as an interim improvement, air accumulators on the actuation mechanism and alarms to indicate loss of instrument air should be provided to insure that protection is provided and to alert the operator in the event that instrument air is lost. Whichever implementation option you select, your submittal detailing the proposed design improvements must be submitted in sufficient time to allow our review and approval to meet the objective discussed above.

You are requested to identify to us the implementation option you have selected and to provide your response to the staff positions and requests for additional information identified in the attached enclosure, within 45 days of receipt of this letter.

Sincerely,

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures and cc:
See next page

Connecticut Yankee Atomic
Power Company

- 3 -

January 10, 1977

Enclosures:

1. Positions and Additional
Information Requests
2. Meeting Summary

cc: Day, Berry & Howard
Counselors at Law
One Constitution Plaza
Hartford, Connecticut 06103

Russell Library
119 Broad Street
Middletown, Connecticut 16457

ENCLOSURE
STAFF POSITIONS AND ADDITIONAL INFORMATION REQUESTS
CONNECTICUT YANKEE (HADDAM NECK)
DOCKET NO. 50-213

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 overpressure events which have occurred at PWR facilities. Your October 15, 1976 letter to the NRC says that your operators have been informed of the past events. POSITION: The staff, however will require that formal discussions be held to review the causes of the past events, the plant conditions at the time, the mitigating action that could have been or was taken and the preventive measures that would have avoided the transient. Also, identifying plant similarities and distinctions, and discussing how these relate to plant startup, shutdown and testing operations is necessary. Please furnish 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?
 - 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.
 - 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.
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. Those 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.

- 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.
3. The inadvertent operation of SIS components during cold shutdown conditions has been responsible for a major portion of the over-pressure incidents. POSITION: Based on the licensee submittals, the recent November 3-5, 1976 meetings, and discussions with MSS 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. Indicate the RCS temperature/pressure conditions for which the accumulator isolation valve will be closed, and describe the location of the breaker, and the places from which it may be controlled.

- j. Describe the impact on overall plant operations if you routinely lowered accumulator nitrogen pressure when in a cold shutdown condition.
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?
 - b. What extra measures are taken to prevent an overpressure event during these tests?
5. The staff believes that a high pressure alarm used during low RCS temperature operations is an effective means to attract the operators's attention to a transient in progress. POSITION: Although it clearly is not a mitigating device, if it is not presently installed, the staff will require that it be installed as soon as possible.

Your October 15 letter to the NRC stated the CYAPCO was evaluating the feasibility of installing a computer generated alarm and/or indications. Based on the staff's position discussed above, please provide the following:

- a. Your method to provide the alarm, and the associated time schedule, or your justification for why this cannot be done.
- b. A synopsis of the system modifications that are necessary.
- c. The alarm setpoint, mode of annunciation and sensor.
- d. How you ensure that the alarm is available and operating properly during all water-solid operations and how you minimize its down-time for all other cold shutdown conditions.
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.
7. Reactor coolant system heatups, resulting from improper operation of the reactor coolant pump (RCP) while in a cold, shutdown and water solid condition, have been responsible for approximately 15% of the RCS overpressurization events. POSITION: We will require that all licensees include provisions to prevent RCP starts while in a water solid condition unless absolutely necessary. In those cases where the RCP starts cannot be avoided, the licensee should take appropriate steps to determine and minimize the RCS temperature profile.

Based on the position stated above, please provide the following additional information:

- a. Describe the normal operating conditions during which the RCS is maintained water solid with all RCP's stopped (e.g., fill and vent, pressurizer cooldown).
- b. For each of the above procedures, justify your inability to establish an N₂, air or steam bubble in the pressurizer prior to the start of the first RCP.
- c. What are the limits associated with system temperatures before the first RCP can be started in a solid RCS?
- d. Specify the instruments utilized to determine the RCS temperature profile.
- e. Provide the necessary schematics and procedural descriptions that show that your actions would be to bring the RCS to an isothermal condition.
- f. Summarize any other measures you take to reduce possible RCS pressure spikes during RCP starts, (e.g., open all letdown orifice isolation valves, stop makeup flow, etc.).

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

Attachment (2)

November 17, 1976

DOCKET NOS.: 50-344, 50-213, 50-315, 50-244, 50-247, 50-286, 50-305,
50-266, 50-301, 50-282, 50-306, 50-261, 50-295, 50-304,
50-206, 50-280, 50-281, 50-250, 50-251, and 50-334.

LICENSEE/FACILITY:

ROCHESTER GAS & ELECTRIC CORPORATION (R. E. GINNA)
CONSOLIDATED EDISON COMPANY (INDIAN POINT UNITS 2 & 3)
DUQUESNE LIGHT COMPANY (BEAVER VALLEY UNIT 1)
CONNECTICUT YANKEE ATOMIC POWER COMPANY (HADDAM NECK)
WISCONSIN PUBLIC SERVICE CORPORATION (YEAWAUNEE)
WISCONSIN ELECTRIC POWER COMPANY (POINT BEACH UNITS 1 & 2)
NORTHERN STATES POWER COMPANY (PRAIRIE ISLAND UNITS 1 & 2)
CAROLINA POWER LIGHT COMPANY (H.B. ROBINSON)
SOUTHERN CALIFORNIA EDISON COMPANY (SAN ONOFRE)
VIRGINIA ELECTRIC & POWER COMPANY (SURRY UNITS 1 & 2)
PORTLAND GENERAL ELECTRIC COMPANY (TROJAN)
FLORIDA POWER & LIGHT COMPANY (TURKEY POINT UNITS 1 & 2)
COMMONWEALTH EDISON COMPANY (ZION UNITS 1 & 2)
INDIANA & MICHIGAN POWER COMPANY (D.C. COOK, UNIT 1)

SUMMARY OF MEETING HELD ON NOVEMBER 4, 1976, CONCERNING PROPOSED MEASURES
TO PREVENT REACTOR VESSEL OVERPRESSURIZATION IN OPERATING WESTINGHOUSE
(PWR) FACILITIES.

On November 4, 1976, we met with representatives of PWR licensees with
Westinghouse designed plants to discuss measures being taken to prevent
reactor vessel overpressurization.

A list of attendees is attached.

Significant discussions are summarized below.

We summarized the correspondence and discussions that have occurred with
the Westinghouse licensees since our generic letter on reactor vessel
overpressurization was issued in August 1976. We then identified the
criteria listed below as those that should be applied in the design of
equipment which is to prevent pressure transients that would exceed the
limits of Appendix G to 10 CFR 50.

November 17, 1976

1. Credit of Operator Action - No credit can be taken for operator action until 10 minutes after the operator is aware that a pressure transient is in progress.
2. Single Failure Criteria - The pressure protection system should be designed to protect the vessel given a single failure in addition to a failure that initiated the pressure transient. In this area, redundant or diverse pressure protection systems would be considered as meeting the single failure criteria.
3. Testability - The equipment design should include some provision for testing on a schedule consistent with the frequency that the system is used for pressure protection.
4. Seismic Design and IEEE 279 Criteria - Ideally, the pressure protection system should meet both seismic Category I and IEEE 279 criteria. The basic objective, however, is that the system should not be vulnerable to an event which both causes a pressure transient and causes a failure of equipment needed to terminate the transient.

Representatives of the task group of Westinghouse utilities formed to evaluate the problem of reactor vessel overpressurization provided a description of the steps they have taken to respond to the requirements set forth in our August 1976 letter. A summary was given of the various types of Thermal and mass input transients being considered, and it was indicated that a "bounding" analysis is being performed to consider the worst case situation for all Westinghouse plants. The preliminary results of the mass input analysis show that the pressurizer power relief valves have both the capacity and time response characteristics to limit the resultant pressure surges. The task group, however, indicated that a more detailed analysis would be necessary in the case of the pump-start or thermal type of transient before any similar determination could be made. The detailed plant specific analyses are not scheduled for completion for about six months. Since the power operated relief has evidently been selected by the licensees as the means to limit pressure transients, we urged that efforts be made to begin ordering the necessary equipment now rather than waiting 6 months for the plant specific analyses results. We also urged that the licensees concurrently investigate other factors such that installation times can be minimized. The licensees' task group agreed to look into these matters.

November 17, 1976

We indicated that the "single failure" criteria being assumed by the licensee was not consistent with the conventional single failure criteria required by the staff. The licensee agreed to examine this area further and to provide justification for any deviations from the conventional single failure criteria. This information as well as a discussion of the various administrative measures that the licensee intend to use to prevent pressure transients while shutdown are to be prepared with a target date of submittal to the staff by December 3, 1976.

Gary G. Zech, Project Manager
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
List of Attendees

Meeting Summary for
W plants

- 4 - November 17, 1976

Docket File
NRC PDR
LOCAL PDR
ORB#1 Reading
NRR Reading
B. C. Rusche
E. G. Case
V. Stello
K. R. Goller
D. Eisenhut
T. J. Carter
A. Schwencer
D. Ziemann
G. Lear
R. Reid
R. Clark
L. Shao
R. Baer
W. Butler
B. Grimes
Project Manager
Attorney, OELD
OI&E (3)
S. M. Sheppard
Participants (NRC)
R. Fraley, ACRS (16)
T. B. Abernathy
J. R. Buchanan

NRC STAFF MEETING WITH WESTINGHOUSE PWR LICENSEES

NOVEMBER 4, 1976

ATTENDANCE LIST

NRC

G.G. Zech
R.L. Baer
G. Lanik
L.B. Marsh
J.E. Ouzts
T.J. Carter
I. Villalva
W.A. Paulson
S. Israel
J. Mazetis
D. Tibbitts
D. Neighbors
W.T. Russell
M.B. Fairtile
R.J. Cook
P.B. Erickson
P.E. Harmon
J. Stosnider
R.M. Gamble
R. Clemenson
M. Grotenhuis
D. Hood
W. Pike
R.W. Klecker
J. Wetmore
M. Fletcher
T. Wambach

CAROLINA POWER & LIGHT
CO.

D.B. Waters
R.G. Black

SOUTHERN CALIFORNIA EDISON

W.C. Moody

NUSCO

B. Ilberman
M. Kupinski
P.F. Santoro

CONNECTICUT YANKEE ATOMIC
POWER CO.

J. Levine

DUKE POWER CO.

R. W. Revels
E. M. Geddie, Jr.

VIRGINIA ELECTRIC & POWER CO.

D. W. Speidell, Jr.
A. L. Hogg, Jr.

ALABAMA POWER CO.

J. R. Campbell

COMMONWEALTH EDISON CO.

T.R. Tramm
E.E. O'Brien
W.A. Wogstland

CONSOLIDATED EDISON CO.

C.W. Jackson
P.M. Pivawer
J. Makepeace

POWER AUTHORITY OF THE
STATE OF NEW YORK

P.F. Altern
J.M. Vargess

WISCONSIN ELECTRIC POWER CO.

R. A. Newton

WISCONSIN PUBLIC SERVICE CORP

M.E. STERN

ROCHESTER GAS & ELECTRIC

R.W. Elias
B.A. Snow

AMERICAN ELECTRIC POWER

P.W. Daley
R.L. Shoberg
J.G. Dell Perico

DQUESNE LIGHT COMPANY

S. R. Porter
J. J. Carey

Attendance List

- 2 -

FLORIDA POWER & LIGHT

C.S.Pillar
M.A.Schopmann

WESTINGHOUSE - PWRSD

W. G. Poulson
F. Gilgloff
H. Gutzman
A. M. Sklencar
R. W. Fleming
R. C. Jenkner

NORTHERN STATES POWER CO.

L. L. Taylor

PUBLIC SERVICE E&G CO.

L.K.Miller

SOUTHERN COMPANY SERVICES, INC.

J. R. Lyons

PORTLAND GENERAL ELECTRIC CO.

D. I. Herbon

SNUPPS

F. Schwoerer

February 14, 1977

Connecticut Yankee Atomic Power Company
 ATTN: Mr. Donald C. Switzer
 President
 Post Office Box 270
 Hartford, Connecticut 06101

Gentlemen:

RE: HADDAM NECK PLANT

We have completed a preliminary review of your submittal of December 3, 1976, in which you describe a "Reference Mitigating System" and discuss your plans regarding long term design modifications to provide protection against reactor vessel overpressurization. As indicated in our letter of January 10, 1977, the design criteria that should be applied to your long term design modifications are those identified in our November 4, 1976 meeting. The enclosure to this letter identifies specific design features of the "Reference Mitigating System" which do not meet these criteria and also identifies additional information we will require to continue our review of your submittal.

Please respond to this request for additional information within 45 days of receipt of this letter.

Sincerely,

Original
 Signed by

A. Schwencer, Chief
 Operating Reactors Branch#1
 Division of Operating Reactors

Enclosure:
 Request for Additional
 Information

cc w/enclosure:
 See next page

OFFICE →	ORB#1	ORB#1	ORB#1			
x27433:tsb	WTRussell	GZech	ASchwencer			
SURNAME →						
DATE →	2/14/77	2/14/77	2/14/77			

Connecticut Yankee Atomic Power Company - 2 -

February 14, 1977

cc: Day, Berry & Howard
Counselors at Law
One Constitution Plaza
Hartford, Connecticut 06103

Russell Library
119 Broad Street
Middletown, Connecticut 06457

REQUEST FOR ADDITIONAL INFORMATION

With reference to the acceptance criteria required for your overpressure protection system as contained in our January 1977 letter that also discussed interim measures and installation schedule, this request for additional information identifies those aspects of the Westinghouse proposed Reference Mitigating System (RMS) that are not acceptable.

The specific design features of the RMS are listed in your submittal as paragraphs headed alphabetically a. through i. Those design features that do not meet the acceptance criteria are as follows:

- 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.
- Paragraph b.: The single failure criteria require a separate air accumulator for each relief valve operator.
- Paragraph c.: Provide the basis for any deviations from IEEE275 requirements for electrical components.
- Paragraph d.: An annunciator must be provided to insure that the keylock switch is in the proper position when shutdown.
- Paragraph e.: Provide the basis for any deviations from Seismic Category I design requirements for system components.
- Paragraph f.: The accumulators should also be seismically qualified.
- Paragraph g.: Loss of air supply to the relief valve accumulators should be alarmed during shutdown.

In addition to providing a resolution to each of the above items, the following additional information is requested:

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.

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.
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.
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.
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.
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.
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.
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?

TELEPHONE
203-866-6911

CONNECTICUT YANKEE ATOMIC POWER COMPANY

BERLIN, CONNECTICUT

P. O. BOX 270 HARTFORD, CONNECTICUT 06101

March 1, 1977

Docket No. 50-213

Director of Nuclear Reactor Regulation
Attn: Mr. A. Schwencer
Operating Reactors Branch #1
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

- References:
- (1) A. Schwencer letter to D. C. Switzer dated August 11, 1976.
 - (2) D. C. Switzer letter to A. Schwencer dated September 3, 1976.
 - (3) D. C. Switzer letter to A. Schwencer dated October 15, 1976.
 - (4) D. C. Switzer letter to A. Schwencer dated December 3, 1976.
 - (5) A. Schwencer letter to D. C. Switzer dated January 10, 1977.

Gentlemen:

Haddam Neck Plant Reactor Coolant System (RCS) Overpressurization

In Reference (5), CYAPCO was requested to respond to a number of NRC Staff requests concerning the administrative measures taken to minimize or preclude the potential for RCS overpressurization. These measures were identified by CYAPCO in Reference (2). In addition, the NRC Staff requested that CYAPCO redefine its proposed long-term actions in light of the results of a preliminary review by the NRC Staff of the actions documented in Reference (4). In response to these requests, the following information is provided.

With respect to the implementation of administrative measures, Attachment I to this letter provides the requested information, describing the details of the briefing of operators at the Haddam Neck Plant, the review of plant operating procedures and administrative controls, and the installation of a computer-generated alarm.

As identified in Reference (4) and discussed at a meeting between the CYAPCO and NRC Staff on February 24, 1977, the permanent modification proposed by CYAPCO for the Haddam Neck Plant involves replacement of existing Power Operated Relief Valves (PORV's), a conversion from a solenoid to an air valve control system, and installation of a new logic system which, upon approach to a programmable pressure limit line, would modulate the new PORV's to provide suitable pressure relief. Consideration of this type of modification has been dictated by the plant operating restrictions discussed in the CYAPCO response to Question #2 (see Attachment 1).

The following design bases, which differ slightly from those previously documented, have been established at this time for the conceptual design of this mitigating system.

- (1) Operator Action - Credit for operator action will not be taken until 10 minutes after the initiation of the RCS overpressurization event.
- (2) Common Equipment Failure (CEF) - CEF will be considered in the relief system design to the extent that the event which initiated the overpressurization will not disable the relief system. Consideration will be given to events such as loss of offsite power, loss of instrument air, and a seismic event.
- (3) Electrical Criteria - The electrical design of the relief system will be consistent with existing plant criteria. Further, the design will consider electrical separation in light of Item #2 above.
- (4) Seismic Criteria - The seismic design of the relief system will be consistent with existing plant criteria and Item #2 above.

With respect to Items #3 and #4 above, it should be noted that the design of the proposed modification will be based on the same codes and standards used for the original design of the Haddam Neck Plant. CYAPCO concurs with the design philosophy of the NRC Staff, documented in Reference (5), that the basic objective is not to be vulnerable to a single event which both initiates a pressure transient and disables the mitigating system; this objective is stated in Item #2 above.

The ability of the proposed mitigating system to meet the single failure criteria cannot be fully established at this time. As indicated in discussions during the February 24, 1977 meeting, it is anticipated that the control logic of the proposed system will be redundant; however, the ability of the new PORV's to provide sufficient relief capacity, considering a single failure, cannot be ascertained until the results of the RCS transient analysis are known.

The schedule for the installation of these proposed modifications was identified in Reference (4). Initial phases of procurement of the two new PORV's began shortly after the 1976 refueling outage. An analysis of potential RCS overpressurization transients for the Haddam Neck Plant is currently underway and results are expected to be available in May, 1977, at the earliest. Therefore, it is planned that modifications will be installed during the first refueling outage after December 31, 1977, contingent upon material availability. It is also expected that a detailed description of these modifications and supporting analyses will be submitted to the NRC Staff in the latter half of 1977.

With regard to actions prior to December 31, 1977, CYAPCO is currently evaluating possible alternatives, including the installation of interim hardware improvements such as the use of the existing PORV's with dual setpoint control logic. However, as noted above and during the February 24, 1977 meeting, plant specific operating restrictions limit the feasibility of certain of these alternatives. These restrictions, together with the results of RCS transient analyses and review of other

safety-related considerations pertaining to interim hardware improvements, are important parts of this overall evaluation. It is expected that the results of the CYAPCO evaluation of interim actions will be available during May, 1977.

Very truly yours,

CONNECTICUT YANKEE ATOMIC POWER COMPANY

D. C. Switzer
President

Attachment

ATTACHMENT I

Docket No. 50-213

CONNECTICUT YANKEE ATOMIC POWER COMPANY
HADDAM NECK PLANT
RCS OVERPRESSURIZATION
ADDITIONAL INFORMATION ON ADMINISTRATIVE MEASURES

March, 1977

- 1) The staff considers it essential that all plant operators (i.e., reactor operators, equipment operators, Instrument and Control personnel) be made aware of the details of the overpressure events which have occurred at PWR facilities. Your October 15, 1976 letter to the NRC says that your operators have been informed of the past events. POSITION: The staff, however, will require that formal discussions be held to review the causes of the past events, the plant conditions at the time, the mitigating action that could have been or was taken and the preventive measures that would have avoided the transient. Also, identifying plant similarities and distinctions, and discussing how these relate to plant startup, shutdown and testing operations is necessary. Please furnish the following information:

a) Question

If you have not already completed the required formal discussion, when will you do so?

Answer

As was stated in our 10/15/76 letter we have held discussions during our normal retraining sessions and informed the operators of the concerns in regard to Reactor Coolant System Overpressurization. However, to comply with the staff's requirements, we will have to hold additional training sessions. We plan to complete this training by the end of May 1977.

b) Question

How will the discussions be held?

Answer

The discussions will be held as specially scheduled retraining classes. All licensed operators along with

Reactor Coolant System Overpressurization.

auxiliary operators and I&C Technicians will be required to attend.

c) Question

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.

Answer

The events which are not credible at Conn. Yankee as described in the License Event Reports are:

- 1) Indian Point Unit #2 - (3/8/72), "Reactor Coolant Pump starting swept cold water thru hot steam generator - pressure increase due to thermal expansion."

At Conn. Yankee the loop isolation valves are used in the pump starting sequence. The Tc isolation valve (cold leg) must be closed in order to start the RCP. It takes 3 minutes to fully open the valve and the valve can not be opened unless it is within 20°F of the hottest loop. Therefore, having to start the pump with the Tc valve closed and the bypass open, the loop and steam generator rapidly approach an isothermal condition. The Tc valve interlock of 20°F to the warmest loop prevents injection of cold water through the rest of the system, and the very slow opening time of the Tc valve (3 minutes) prevents a slug of cold water entering the vessel even if it is only maximum of 20°F less than system temperature.

- 2) Oconee Nuclear Station Unit #2 - (11/15/73), "During Zero Power Physics testing, test procedure instructions directed operating personnel to increase reactor coolant pressure to approximately 1860 psig violating the limits."

At Conn. Yankee, with the exception of cold rod drop testing, all zero power physics testing is done at ~ 535°F and 2000 psig in the RCS (not zero power conditions). The cold rod drops are done with a bubble in the pressurizer. This test procedure will be changed to insure that the procedure reflects the necessity of having a bubble in the pressurizer during the test.

Reactor Coolant System Overpressurization

- 3) Indian Point Unit #2 - (2/22/74), "An inadvertent safety injection signal was generated which, by design, caused the accumulator discharge stop valves to open."

Conn. Yankee's safety injection system does not have accumulators in it, therefore, this accident can not happen.

- 4) Prairie Island Unit #1 - (10/31/73), "Reactor Coolant Pump starting swept cold water thru hot steam generator - pressure increase due to thermal expansion."

This can not happen at Conn. Yankee because of the valve pump interlocks as discussed in the first case above, (I.P. Unit #2, 3/8/72).

- 5) Prairie Island Unit #1 - (1/16/74), "While conducting safeguards logic train, a monthly surveillance test, a SI signal was initiated when a step which puts train A in test was inadvertently missed. The SI signal opened No. 11 accumulator outlet isolation valve. Residual Heat Removal system isolation occurred as designed at 600 psig."

This can not happen at Conn. Yankee for two reasons:

a) CY does not have RCS accumulators, and b) CY does not have automatic isolation of RHR.

- 6) Prairie Island Unit #2 - (11/27/74), "A test signal injected into the letdown controller instrument loop caused a letdown control valve to go closed. This isolated the letdown path. RHR system automatically isolated."

Although closure of the letdown control valve is possible, CY does not have automatic isolation of the RHR, therefore, the RHR relief valve would be in-service.

- 7) Surry Unit #1 - (1/28/73), "During process of filling and venting the RCS, "A" accumulator motor operated discharge isolation valve was opened to sweep any air trapped in accumulator discharge line into RCS. The opening of the valve caused the accumulator to cause the increase in RCS pressure."

Conn. Yankee does not have accumulators on the reactor coolant system or the safety injection system.

Reactor Coolant System Overpressurization

- 8) Turkey Point Unit #3 - (12/3/74), "In preparation for starting a reactor coolant pump, the operator placed the letdown control valve in automatic in order to increase reactor coolant pressure. At 465 psig the RHR system loop suction isolation valve automatically closed isolating letdown."

Since Conn. Yankee does not have automatic isolation of the RHR system, the RHR 500 lb. relief valve would have prevented this incident.

- 9) Zion Unit #1 - (6/13/73), "Charging pump 1A, with suction from Refueling Water Storage Tank, was started to increase reactor system pressure. Normal pressure control of continuous charging and letdown was not being used since Volume Control Tank was unavailable. Operator was distracted by a telephone call and left the area of the pump control switch. Unattended pump continued to pressurize system. RHR suction relief valve failed to lift and RHR system later isolated automatically at 600 psig."

At Conn. Yankee the first part of the incident is possible, but since the RHR system is not automatically isolated from the system, relief protection would be available.

- 10) Zion Unit #2 - (9/18/75), "Station personnel were performing a RHR valve interlock test in which the RHR System is automatically isolated from the reactor coolant system. When the applied test signal reached the setpoint, the RHR isolation valves closed removing the letdown path."

There is no automatic isolation of the RHR system at Conn. Yankee. Also at CY if normal letdown is not being used, the alternate letdown path would be through the loop drain valves not the RHR system.

- 11) Peach Bottom Unit #2 - (3/6/74), "Following a main steam line isolation test, portions of the reactor vessel shell temperatures decayed to 125 F while reactor pressure remained at approximately 400 psig."

Conn. Yankee being a PWR can not get into this condition. This condition only exists on BWR's

- 12) St. Lucie Unit #1 - (6/17/76), "With Shutdown Cooling System secured, a reactor coolant pump was started and approximate RCS temperature rose to 140°F - reason unknown as of 6/22/76."

Conn. Yankee valve-pump interlock would prevent this from happening. An individual loop could possibly be warmed up but the valve would not be opened until it was within 20°F of the warmest running loop.

Reactor Coolant System Overpressurization

- 13) Indian Point Unit #2 - (1/23/74), "Starting of a single reactor coolant pump caused pressure surge. A nitrogen blanket in the pressurizer to act as a surge volume had been established; however, the amount of nitrogen added to the pressurizer was insufficient."

Starting an Reactor Coolant Pump at Conn. Yankee would not cause a pressure surge increase unless the water in the loop was hotter than the rest of the system. The 20°F valve interlock would prevent this loop from being cut into the system.

- 14) Indian Point Unit #2 - (2/17/72), "Operator isolated letdown without verifying availability of letdown thru RHR system."

Zion Unit #1 - (6/3/75), "Operator failed to stop the centrifugal charging pump when he secured the RHR system to replace the RHR suction relief valve. When the RHR system was secured, letdown was also secured."

Trojan - (7/22/75), "The RHR suction valve from the RCS was closed by an unknown person (i.e., this isolated letdown) while the positive displacement charging pump was operating."

- 15) Point Beach Unit #2 - (2/28/76), "Operational reasons required the RHR system to be isolated from the reactor coolant system. Reduced letdown resulted in pressure increase."

On the above four units it is obvious that their letdown is lined up to the RHR system while in cold shutdown. Therefore, isolation of RHR isolates letdown. At Conn. Yankee letdown is achieved in one of two ways:

- a) Normal letdown is out of Loop 1 hot leg through the letdown orifices into the chemical and volume control system. Refer to Attachment 1 & 2.
- b) Alternate letdown would be through each separate hot leg drain valve which connect to a header and then through the drain header throttle valve.

If letdown were taking place on the CY system, closure of the RHR isolation valves would not change letdown.

- 15) Beaver Valley Unit #1 - (2/24/76), "Operator error in transferring electrical buses caused instrument spike isolating letdown from RHR System."

Conn. Yankee does not have automatic isolation of the RHR system, therefore, an instrument spike would not isolate our RHR system.

Reactor Coolant System Overpressurization

- 16) Beaver Valley Unit #1 - (3/5/76), "Instrument Technician tripped wrong B/S during MSP, then Ops placed inverter in service with output breaker open, deenergizing #1 vital bus, causing Safety Injection System which isolated letdown".

This would not happen at Conn. Yankee because of the use of tie-breakers in the system. Any time a vital bus is lost it is picked up by the second vital bus. Also, if a feed breaker on one vital bus is opened, the connecting bus breaker is automatically closed because of the tie-bar. Refer to Attachment #3.

b) Question

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.

Answer

The events which are credible at Conn. Yankee as described in the License Event Reports are:

- 1) Indian Point Unit #2 - (4/6/72), "Operator inadvertently isolated letdown."

Palisades - (9/1/74), "A procedure "CAUTION" statement was not rigorously adhered to while performing a primary coolant system leak test."

Point Beach Unit #2 - (12/10/74), "Following repair, a safety injection pump was lined up for a test run. However, safety injection pump discharge was not isolated from injecting into the reactor coolant system. Pressure transient caused by starting of SI pump.

The above three incidents would all be procedural violations at Conn. Yankee. There is no way that human error can be completely removed from operating a plant in any mode of operation. However, we feel that Conn. Yankee's operating experience and record to date are a reflection of good operating procedures and well trained and highly qualified operators. Conn. Yankee has not experienced an overpressurization incident in its 10 years of operation. The mitigating system which we propose to install will eliminate the hazard of such an error by providing sufficient relief protection during solid operation.

Reactor Coolant System Overpressurization

As was stated in our 10/15/76 letter to the NRC we have reviewed and revised our procedures to minimize the time solid and alert the operators to the possibility and danger of overpressurization. Attachment #4 is a listing of the procedural changes made.

- 2) D. C. Cook Unit #1 - (4/14/76), "During RPS testing, inadvertent letdown isolation was initiated."

During our normal Reactor Protective System testing at Conn. Yankee, we are normally hot and at power. However, these tests are also done during the loss of AC and Core Cooling tests. In this case, we would be cold and vented. The core cooling system would be isolated from the system to prevent overpressurization.

- 3) Indian Point Unit #2 - (5/18/73), "Closure of certain air operator valves in reactor coolant letdown system caused by freezing of moisture in air supply line."

Although this incident is credible at Conn. Yankee it is extremely unlikely. The control air system which supplies air to these valves is periodically checked for moisture. The system is operated at a -60°C dew point. Also all valves associated with letdown are within enclosures which make the incident even more remote.

- 4) St. Lucie Unit #1 - (8/12/75), "Letdown isolation valve failed closed when I&C personnel removed cover from sealing relay associated with letdown isolation valve. When relay cover was removed, broken wires on relay became disconnected causing letdown valve to close."

This incident is one which is virtually impossible to protect against. Apparently the I&C technician had no way of knowing there were loose wires in the relay until the cover was removed. We feel that the NRC's requirement that I&C personnel attend training on these issues will help eliminate such an incident.

- 2) The majority of the reported pressure transient 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. Those operations in which the plant is in a water solid condition must be fully justified. Accordingly, please provide the following information:

Reactor Coolant System Overpressurization

Question

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

Answer

It is necessary for CY to operate in the water solid condition for a limited amount of time. The following evolutions require water solid operation:

Filling and venting the entire system (NOP 2.4-1)
Hydrostatic Test (SUR 5.1-1)
Plant Startup, Cold Shutdown to Hot Standby (NOP 2.1-1)
Plant Shutdown, Hot Standby to Cold Shutdown (NOP 2.3-4)

Before describing the reasons for going solid in each of the above evolutions, it is important to point out the common limiting design criteria which dictate the amount of time we must remain in the solid condition:

- 1) The minimum Reactor Coolant System pressure needed to operate a Reactor Coolant Pump is 375 psig. This limitation is the addition of three factors: 275 psig ΔP across #1 seal; 25 psig backpressure on #1 seal leakoff; and 75 psig to account for instrument error and small system transients while starting the pump.
- 2) The ΔT between the pressurizer surge line and the reactor coolant system is 200°F.

Filling and Venting The Entire System - In order to properly vent the Reactor Coolant System, it is absolutely necessary to go solid. Each of the loops, vessel, pressurizer, have vents on them which are opened during the filling process. As the system is being filled, the air is being forced through the vents. Once the system is full, the RCP's are run, one at a time, to remove any trapped air in the loops (pump casings, steam generators, etc.) and force it into the reactor vessel head area to be vented off. In order to run the RCP we must have a minimum system pressure of 375 psig. While venting the system, temperature must be maintained less than 200°F to avoid steaming off through the vents and causing air or steam pockets in the system. At 375 psig the RCS temperature would have to be $\sim 440^\circ\text{F}$ to produce saturated steam. With the limitation of 200°F between pressurizer surge line and reactor coolant system the maximum temperature in the pressurizer would be 400°F. Therefore,

Reactor Coolant System Overpressurization

It is evident that a steam bubble can not be formed with these limitations being adhered to. It is very important to remember that the venting is done to protect the system. Removing air from the system prevents possible stress corrosion. Addition of a nitrogen bubble in the pressurizer is not recommended by our NSSS supplier because of the possible formation of nitric acid in the system. The system chemistry specs must not be violated.

Hydrostatic Test - This test is required for verification of the Reactor Coolant System leak tightness after the system has been opened for refueling or maintenance. The entire system must be solid in order to detect all possible system leaks. If air, steam, or any gas, is in the area of a leaking component, it is very likely that the leak will not be found until the system is up in temperature and pressure. This test is performed after completion of filling and venting the entire system. The reactor coolant pumps are run in order to bring the system temperature up between 220°F to 240°F, and are periodically run during the test to maintain an isothermal condition throughout the system until completion of the test.

Cold Shutdown To Hot Standby - After completion of filling and venting, and the hydrostatic test, the system temperature is maintained <200°F, pressure is atmospheric with the pressurizer vented at 85% level. The system temperature is increased by running the RCP's and the pressurizer heaters. In order to run the RCP the system pressure must be at least 375 psig which would require a minimum of 440°F in the pressurizer to form a steam bubble. With a 200°F ΔT the RCS temperature would have to be a minimum of 240°F. It is important to remember that these values are all minimums and normal operations would be performed at more conservative values. Once seal water flow is established to the pump seals, a charging pump is started. The seal water flowing into the system fills the pressurizer and the vent is closed. The system is solid and letdown is lined up and in service. The RCP is now started and all heaters are energized in the pressurizer. When the pressurizer water temperature increases to ~450°F with the Reactor Coolant System Temperature 250°F to 300°F and pressure approximately 450 psig a bubble is formed in the pressurizer. This is done by draining the RCS through the drain header throttle valve to the VCT or WDS. The pressurizer is drained to 30% level. It is important to note that if the dissolved oxygen concentration was not less than 0.10 ppm the RCS heatup would be stopped at 230°F until the spec. was met. This would prolong the time in the solid mode until the affect of the added hydrazene removed the oxygen from the system.

Reactor Coolant System Overpressurization

Hot Standby to Cold Shutdown - This evolution starts with the reactor coolant system at 533°F and 2000 psig. The intent of the procedure is to bring the plant to atmospheric pressure and ~100°F. The 200°F ΔT limitation applies to the pressurizer sprays as well as the surge line. In order to cool down the pressurizer top half (steam phase) it is necessary to go solid. If a steam bubble were in the top half of the pressurizer, the ΔT limit between the spray nozzle and Reactor Coolant system would be quickly exceeded. The bubble is broken at approximately 450°F in the pressurizer steam phase area and 600 psig in the Reactor Coolant System. This must be done in order to allow for an even system cooldown and allow operation of the Reactor Coolant Pump. When the pressurizer temperature has been reduced to approximately 300°F and system pressure to 400 psig the Reactor Coolant Pumps are stopped and the RHR system placed in service. To ensure an even cooldown the RCP's are periodically run for 10 minutes to completely mix the system and bring it to an isothermal condition.

- 3) The inadvertent operation on 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 de-energizing of SIS pumps and closure of SI header/discharge valve 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) Question

A schematic diagram of the SIS showing the flowpaths into the RCS.

Answer

Refer to Attachment #5, Core Cooling System.

b) Question

The head flow characteristics of each of the SIS pumps.

Answer

- 1) High Pressure Safety Injection Pumps (HPSI):
Flow = 1750 gpm, head = 2250 ft.
- 2) Low Pressure Safety Injection Pumps (L.P.S.I.):
Flow = 5500 gpm, Head = 590 ft.
- 3) Charging Pumps:
Flow = 360 gpm, Head = 2750 psig.

Reactor Coolant System Overpressurization

- 4) Residual Heat Removal Pumps (RHR):
Flow = 2200 gpm, Head = 300 ft.

c) Question

Identify on the schematic diagrams the pumps and the valves to be closed and de-energized.

Answer

Refer to Attachment #5, Core Cooling System.

d) Question

Your time schedule for implementing these administrative and operating procedural changes to meet this position.

Answer

De-energizing the SIS pumps and closure of the discharge valves is and has been part of Conn. Yankee's Core Cooling line-up procedure for shutdown condition.

e) Question

Indicate all circumstances for which the SIS pumps and valves may not be isolated and de-energized and for those situations, describe the manner in which SIS injection would be prevented.

Answer

During a cold shutdown (refueling) the Partial Loss of AC with Core Cooling Actuation is performed. During this test the MOV's are energized and the safety injection pumps are energized. However, the discharge valves are locked closed on the pumps and only one train is in this configuration at a time. The pumps and MOV's are energized to verify that they operate on a core cooling signal. During this test the system is vented. Also during this test the charging pump is running with the manual discharge valve closed and pressure being maintained through the Reactor Coolant Pump Seals. The seal water return valve must be open during this test to verify it will go closed on a core cooling signal.

f) Question

The location of the breakers that will be opened, and the placed from which they can be controlled.

Reactor Coolant System Overpressurization

Answer

The motor operated valve breakers are on MCC-5 which is located in Switchgear. With the breakers energized the valves can be operated from the Main Control Board. The valves can be manually operated at the valve. When the breaker is de-energized, it is tagged at the MCC and a tag is placed on the valve motor operator handwheel.

The safety injection pump breakers are racked out and tagged when the pumps are de-energized. The control switches are located on the main control board and are placed in the trip pullout position and tagged.

g) Question

Describe the position indication and status signals which will be lost as a result of de-energizing these components.

Answer

The de-energized pumps would have the hand switch in the trip pullout position and the racked out breaker would cause a loss of light signal on the MCB.

The de-energized valves would have a loss of light signal on the MCB also. Neither the green (closed) or red (open) lights would be lite.

h) Question

Describe in detail, the administrative procedures which will be used to assure proper equipment alignment and the supervisory personnel responsible for maintaining control.

Answer

The operating procedures have valve alignment checkoff lists attached to the procedures. These checkoff lists have the normal position and the position the valve is to be put in. If the position of any of these valves is found to be incorrect, the Shift Supervisor must be informed. The valve list is reviewed by the Control Room Operator performing the test and approved by the Shift Supervisor.

It is the Control Operators responsibility to:

- 1) Direct the Auxiliary Operators in essential support operations.
- 2) In the absence of the Supervising Control Operator, perform all duties and exercise all authority of that position until relieved.

It is the Supervising Control Operators responsibility to:

Reactor Coolant System Overpressurization

- 1) Assume responsibility for the safe and orderly conduct of plant operations during assigned shift.
- 2) Assume responsibility for plant operation in accordance with applicable licenses, Technical Specifications, governmental regulations and permits, procedures, company orders, rules, instructions and policy requirements.
- 3) Assume responsibility for conducting necessary tests and inspections as scheduled or otherwise required.
- 4) Assume responsibility for keeping the Shift Supervisor informed of any condition which affects or may affect plant status or operability and all unusual or abnormal plant conditions.

It is the Shift Supervisors responsibility to:

- 1) Assume responsibility and has the authority for insuring the safe and efficient operation of the plant and its supporting systems during assigned periods in accordance with applicable licenses, governmental regulations and permits, Technical Specifications, procedures, company orders, rules, instructions and policy requirements.
- 2) Assume responsibility and authority for approving/disapproving all work and testing activities which may affect the operation of the plant and/or its supporting systems prior to commencement.
- 3) Assume responsibility for the safe and orderly conduct of plant operations during assigned shift.
- 4) Assume responsibility for conducting the necessary tests and inspections as scheduled or otherwise required.
- 5) Maintain communication with the Control Room and grants permission for significant operating activities in advance.

1) Question

Indicate the RCS temperature/pressure conditions for which the accumulator isolation valve will be closed, and describe the location of the breaker, and the places from which it may be controlled.

Answer

Since Conn. Yankee does not have safety injection accumulators, this question does not apply.

Reactor Coolant System Overpressurization

j) Question

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

Answer

Since Conn. Yankee does not have safety injection accumulators, this question does not apply.

- 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) Question

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

b) Question

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

Answer

During the Partial Loss of AC coincident with Core Cooling Actuation test, the safety injection pumps are started. As explained in the answer to question 3c, this test is performed one train at a time, with the manual operated discharge valves locked closed. The system is vented while performing this test. Also the pumps are pre-run, with all valves closed, before the valve line-up is changed for the test. This would insure back-up protection if the discharge valves were not closed and the discharge valves would then be closed for the test.

- 5) The staff believes that a high pressure alarm used during low RCS temperature operations is an effective means to attract the operators' attention to a transient in progress. POSITION: Although it clearly is not a mitigating device, if it is not presently installed, the staff will require that it be installed as soon as possible.

Your October 15 letter to the NRC stated the CYAPCO was evaluating the feasibility of installing a computer generated alarm and/or indications. Based on the staff's position discussed above, please provide the following:

Reactor Coolant System Overpressurization

a) Question

Your method to provide the alarm, and the associated time schedule, or your justification for why this cannot be done.

Answer

A computer generated alarm was put into service on 10/27/76. A program was written which monitors reactor coolant pressure and each cold leg temperature, and alarms the operator when an overpressure condition exists. This alarm was installed only as an interim measure. If our mitigating system is accepted by the NRC an independent alarm (not computer generated) will be installed with the new control system.

b) Question

A synopsis of the system modifications that are necessary.

Answer

The final alarm system will be actuated off a bi-stable which will get its signal from a signal conditioner. This signal conditioner (two, one for heatup, one for cooldown) will be programmed to follow the applicable Appendix C curves. These signal conditioners are taking the lowest RCS loop temperature and converting it to a corresponding pressure as derived from the Appendix C curve. The bi-stable will also receive a signal from the actual system pressure and will be activated on the comparison of the two.

c) Question

The alarm setpoint, mode of annunciation and sensor.

Answer

50 The alarm setpoint for the interim system can be seen on
RCS Attachment #6. If the setpoint is exceeded the "Data Logger
Message" Alarm will annunciate and a message will be
printed on the data logger, as shown on Attachment #7,
which is an example. The sensors being used are wide range
reactor coolant pressure and the cold leg temperature RTD's.

d) Question

How you ensure that the alarm is available and operating properly during all water-solid operations and how you minimize its down-time for all other cold shutdown conditions.

Reactor Coolant System Overpressurization

Answer

With the interim alarm system, the alarm is available as long as the computer is operating. The computer will normally be operating during all evolutions which require water-solid operations. Unless the computer was being serviced it would be operating during all other cold shutdown conditions. The proposed mitigating system alarm would not be computer dependent, and would be available during all cold and water-solid conditions.

- 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) Question

RHR (or SCS) design pressure.

Answer

The design pressure is 500 psig on the Conn. Yankee RHR system.

b) Question

A description of the system isolation valves and their arrangement (e.g., number and configuration of valves installed, and pneumatic or motor operated).

Answer

Refer to Attachment #5. There are four motor operated isolation valves on the RHR system. Two of these are key-operated valves (RH-MOV-781, RH-MOV-803) and two are pressure-interlock valves (RH-MOV-780, RH-MOV-804).

c) Question

Interlocks, interlock setpoints, and alarms associated with each isolation valve.

Answer

The two pressure-interlock valves can not be opened unless the system pressure is below 400 psig. There is no automatic isolation of RHR on the Conn. Yankee System, and no alarms associated with the interlock.

Reactor Coolant System Overpressurization

d) Question

Nominal stroke time of isolation valves.

Answer

The normal stroke time of these valves is approximately 2 minutes in either direction.

e) Question

The setpoint and capacity of RHR (or SCS) relief and safety valves.

Answer

There is one relief valve on the discharge side of the RHR pumps. This relief valve is set at 500 psig and is designed to relieve at a rate of 960 gpm at 500 psig.

f) Question

All pressure alarms, setpoints and associated annunciation for the system.

Answer

There are no pressure alarms on the system. There is a pressure indication on the main control board. There is a low flow alarm set at 2200 gpm. There is a temperature indicator on the main control board, but there are no associated alarms for system temperature.

- 7) Reactor coolant system heatups, resulting from improper operation of the reactor coolant pump (RCP) while in a cold, shutdown and water solid condition, have been responsible for approximately 15% of the RCS overpressurization events.
POSITION: We will require that all licensees include provisions to prevent RCP starts while in a water solid condition unless absolutely necessary. In those cases where the RCP starts cannot be avoided, the licensee should take appropriate steps to determine and minimize the RCS temperature profile.

Based on the position stated above, please provide the following additional information:

a) Question

Describe the normal operating conditions during which the RCS is maintained water solid with all RCP's stopped (e.g., fill and vent, pressurizer cooldown).

Reactor Coolant System Overpressurization

Answer

As described in answers 2a and b, there are four conditions in which the plant requires solid operation:

- 1) Filling and Venting the Entire System - During this evolution the plant starts out cold ($\sim 100^{\circ}\text{F}$), atmospheric pressure, and vented. The objective is to remove as much air or gas from the vented system as possible. This is accomplished by filling the system through the fill header, using a charging pump. As the water is injected into the system, the gases are vented off to the waste disposal system. Once the system is filled the vents are closed, however, there are still pockets of gases in the system. In order to remove these gases the reactor coolant pumps must be run. This forces the gas out of the loops into the vessel head where it is then vented off. If the pumps were not run the gases would eventually dissolve into the reactor coolant (during heatup) and chemistry requirements could not be met. This would cause prolonged operation in the solid mode of operation.
- 2) Hydrostatic Test - During the hydrostatic leak test the plant is brought from atmospheric pressure to 2000 psig. The pressure is provided by use of the charging pump. This evolution takes place after filling and venting the system, because the system must be solid. In order to bring the reactor coolant system pressure up to 2000 psig and stay within the Appendix G curves, the system temperature must be brought up to 220°F - 240°F and maintained at that temperature. The system is $\sim 100^{\circ}\text{F}$ at the start. The service water to the RHR heat exchangers is cut off and the system temperature is slowly brought up by decay heat in the core. The RHR pumps continue to run to provide mixing in the system for an even heatup. Unfortunately decay heat (after a normal refueling outage) is not sufficient to bring the system up to the required temperature. This is also an extremely slow method of heatup. Therefore, the RHR system is secured and a Reactor Coolant Pump is run, one loop at a time. This process provides a reasonable heatup rate and an even system heatup. Once the system is at the required temperature a RCP is periodically run to maintain the required temperature during the test. Once the test is complete the system is de-pressurized the temperature taken to $< 200^{\circ}\text{F}$, and the pressurizer vented with the level down to $\sim 85\%$.
- 3) Cold Shutdown to Hot Standby - This evolution will normally take place right after completion of the Hydrostatic Test. Therefore, the system pressure is atmospheric with the pressurizer vent and system temperature $< 200^{\circ}\text{F}$. It is necessary to heat the system up at this time while solid. The pressurizer heaters are energized, however, these are not sufficient to heat up the RCS also. The 200°F ΔT must be adhered to. Approximately 450°F at

Reactor Coolant System Overpressurization

400 psig is required in the pressurizer to pull a steam bubble, therefore, the minimum temperature in the RCS would be ~250°F. To get 250°F in the RCS a Reactor Coolant Pump must be run. To run an RCP the minimum RCS pressure is 375 psig. Once the pressurizer temperature is 450°F a bubble will be formed. It is essential to maintain a high pressure (~400 psig) in the system to insure a bubble does not form in the vessel or a loop.

- 4) Hot Standby to Cold Shutdown - This evolution starts with the reactor coolant system at 533°F and 2000 psig. The intent of the procedure is to bring the plant to atmospheric pressure and ~100°F. The 200°F ΔT limitation applies to the pressurizer sprays as well as the surge line. In order to cool down the pressurizer top half (steam phase) it is necessary to go solid. If a steam bubble were kept in the top half of the pressurizer the ΔT limit between the spray nozzle and Reactor Coolant System would be quickly exceeded. The bubble is broken at approximately 450°F in the pressurizer steam phase area and 600 psig on the Reactor Coolant System. This must be done in order to allow for an even system cooldown and allow operation of the Reactor Coolant Pump. When the pressurizer temperature has been reduced to approximately 300°F and system pressure to 400 psig, the Reactor Coolant Pumps are stopped and the RHR system placed in service. To insure an even cooldown the RCP's are periodically run for 10 minutes to completely mix the system and bring it to an isothermal condition.

b) Question

For each of the above procedures, justify your inability to establish an N_2 , air or steam bubble in the pressurizer prior to the start of the first RCP.

Answer

In all of the above procedures, neither air nor N_2 are introduced in the system. As explained in previous answers air is not desirable because of the likelihood of stress corrosion to system components. N_2 is not desirable in the system because of the possibility of Nitric Acid Formation. It has also been explained that because of our ΔT limitations between pressurizer and RCS, a bubble can not be formed (heatup) or collapsed (cooldown) any sooner than it is presently being done.

Reactor Coolant System Overpressurization

c) Question

What are the limits associated with system temperatures before the first RCP can be started in a solid RCS?

Answer

There are no interlock type limitations on starting of the first RCP. However, once one pump is running the other three loop temperatures must be within 20°F of the warmest running loop before the loop Tc valve can be opened. In order to start any pump in any configuration, the Tc valve must be fully closed and either the Th or bypass valve must be open. It takes ~3 minutes to open the Tc valve, therefore, the loop is isothermal before it can be cut into the system. Administratively, the procedures are being changed so that the loop can not be cut into service unless it is within $\pm 20^\circ\text{F}$ of the system temperature. Presently the valve temperature interlock only prevents a cold loop from being cut in.

d) Question

Specify the instruments utilized to determine the RCS temperature profile.

Answer

Individual loop Tc and loop ΔT indicators are on the main control board, vessel head temperature, pressurizer water phase and pressurizer steam phase indications, and RHR inlet and outlet temperatures are all located on the Main Control Board. During the heatup and cooldown the vessel head temperature is put into trend on the computer. This is the temperature which is monitored to keep within our administrative heatup and cooldown rates of $<50^\circ\text{F/hr}$.

e) Question

Provide the necessary schematics and procedural descriptions that show that your actions would be to bring the RCS to an isothermal condition.

Answer

As previously stated the affected procedures will be changes to require that a loop with a running RCP will not be cut into the system unless it is within $\pm 20^\circ\text{F}$ the system temperature. These are limits and the loop should be as close to system temperature as possible.

Reactor Coolant System Overpressurization

f) Question

Summarize any other measures you take to reduce possible RCS pressure spikes during RCP starts, (e.g., open all letdown orifice isolation valves, stop makeup flow, etc.)

Answer

It is our experience that if any pressure transients results from the starting of an RCP, it is a sag in pressure not a spike. This is one of the considerations in the determination of the minimum system pressure needed before starting a pump. However, if a pressure increase were to be seen, the drain header throttle valve would be opened to bring the pressure back down.

Fig. 5,2-7-1
2/79

Vital Bus Arrangement

RCPS - Reactor Control Protection System
NIS - Nuclear Instrumentation System

Reactor Coolant System Overpressurization

ATTACHMENT #4

The following are changes which have been made to existing procedures. These changes are intended to point out the areas where overpressure can occur and possibly shorten the time solid:

NOP 2.1-1 Plant Startup, Cold Shutdown to Hot Standby.

5.4 Original Statement - The reactor coolant pressure must be maintained in accordance with the pressure-temperature curve included in section 3.4 of the technical specifications for the specified service period (Figure 3.4-2,-4,-6).

Addition to original statement - These curves were calculated in accordance with Appendix G in Section III of the ASME boiler and pressure vessel code. During water solid conditions these limits could be exceeded easily. Strict attention and care must be maintained during this period of operation to avoid overpressurization of the RCS.

NOP 2.3-4 Plant Shutdown, Hot Standby to Cold Shutdown

5.9 Original statement - The reactor coolant pressure must be maintained in accordance with the pressure-temperature curve, Figures 3.4-3, 3.4-5, or 3.4-7 of Section 3.4 of Technical Specifications.

Addition to original statement - These curves were calculated in accordance with Appendix G in Section III of the ASME Boiler & Pressure vessel code. During water solid conditions, these limits could be exceeded easily. Strict attention and care must be maintained during this period of operation to avoid overpressurization of the RCS.

- 6.23 Original Statement - When the pressurizer steam phase temperature has decreased to approximately 450°F and reactor coolant pressure is approximately 600 psig, collapse pressurizer steam bubble by increasing level in the pressurizer to >100%. A solid system (filled pressurizer) is indicated by an increasing system pressure.

Note: Observe system cooldown curves, figures 3.4-3, 3.4-5 or 3.4-7 of section 3.4 of Technical Specifications.

Change the note statement to-

Caution: Without the steam bubble the RCS is always susceptible to overpressurization-strict adherence must be kept to the system cooldown curves figures 3.4-3, 3.4-5, or 3.4-7 of Section 3.4 of Technical Specifications.

- 6.31.8 Original Statement - Repeat this operation every 2-4 hours but start a different reactor coolant pump each time the reactor coolant is circulated.

Addition to the original statement -

Caution: Starting the reactor coolant pump in a solid system can cause a thermal input into the RCS which in turn would cause a pressure increase. System pressure and temperature must be monitored very closely during this operation in order not to exceed the heatup and cooldown curves in Section 3.4 of Technical Specifications.

Reactor Coolant System Overpressurization

ATTACHMENT #4

NOP 2.4-1 RCS, Filling and Venting of Entire System

- 6.2.12 Original statement- Continue filling the pressurizer until water issues from the vent connection, then close valve VH-V-551,
Addition to original statement-

Caution: With the system now solid, extreme care must be taken not to exceed the pressure temperature curves in section 3.4 of Tech. Specs.

- 6.2.26 Original statement- Use the drain header throttle valve (DF-MOV-301) to establish 400-500 psig in the RCS.

NOTE: Observe pressure-temperature limitations as shown in Tech. Specs. Manual, Section 3.4, Figures 3.4.1 through 3.4.7.

Addition to original note- These curves must be used to insure the RCS temperature is sufficient for the pressure the system is being taken to.

- 6.2.29 Original statement- Vent the vessel head and the T_h and bypass lines of the loop which was circulated per steps 6.2.18 & 6.2.21.

NOTE: Maintain RCS pressure at 400-500 psig.

Changed limits in original note-

NOTE: Maintain RCS pressure at 400-425 psig.

NOP 2.9-1 RHR system, placing system in service.

- 5.6 Original statement- Insure strict compliance with all plant cool-down and shutdown limitations (Tech. Specs., Section 3.4, figures 3.4-3, 5 and 7).

Addition to original statements- These curves must be adhered to, in order to insure that the reactor coolant system is not over-pressurized.

NOP 2.12-1 CCS, Lineup for shutdown and at power operation.

- 6.1 Original Statement- Lineup the core cooling systems for cold shutdown operation as follows:

Addition to original statement- (This is to be done with the system pressure between 500 and 1000 psig and between 260°F and 300°F).

- 6.1.6 Original statement- De-energize the core cooling WL switches.

Addition to original statement- This is to be done at ≥ 500 psig and ≤ 1000 psig.

- 6.2 Original statement-Lineup the core cooling and associated systems for hot or at power operation as follows:

Addition to original statement- (This is to be done with the system pressure between 500 and 1000 psig, and between 260°F and 300°F).

Reactor Coolant System Overpressurization

ATTACHMENT #4

6.2.7 Original statement- Re-energize core cooling WL switches.

Addition to original statement- This is to be done at ≥ 500 psig and ≤ 1000 psig.

SUR 5.1-1 Hydrostatic Test

4.2.1 Original Statement- The reactor is shutdown and coolant temperature is between 205°F and 225°F.

Changed statement limits to- between 220°F and 240°F.

5.1 Original statement- The maximum allowable hydrostatic pressure with the core installed is 2735 psig; insure compliance with DTT.

Changed statement to- The maximum allowable hydrostatic pressure with the core installed is 2735 psig; insure compliance with the hydrostatic and leak testing curves in section 3.4 of the Technical Specifications.

6.1.6 Original statement- Announce the start of the test over the PA system and increase pressure to 1000 ± 50 psig as read on the sample room heise gage by increasing the charging pump flow through the fill header flow control valve (FH-FC-295) and/or by regulating the drain header remote throttle valve (DH-MOV-310).

Addition to original statement- Caution: The limits for temperature and pressure for the hydrostatic and leak test curves in section 3.4 of Tech. Specs. must be adhered to, in order to avoid overpressurization of the reactor coolant system.

6.1.7 Original statement- Verify the system temperature is within 205°F - 225°F as read from the computer printout.

Changed the limits of original statement to 220°F to 240°F.

6.2.7 Original statement- Announce the start of the test over the PA system and increase pressure to 1000 ± 50 psig as read on the sample room heise gage by increasing the charging pump flow through the fill header flow control valve (FH-FC-295) and/or by regulating the drain header remote throttle valve (DH-MOV-310).

Addition to original statement- Caution: The limits for temperature and pressure for the hydrostatic and leak test curves in section 3.4 of Tech. Specs. must be adhered to, in order to avoid overpressurization of the reactor coolant system.

6.2.8 Original statement- Verify the system temperature is within 205°F - 225°F as read from the computer printout.

Changed the limits of original statement to 220°F to 240°F.

Reactor Coolant System Overpressurization

ATTACHMENT #4

SUR 5.1-18 Partial Loss of AC Coincident With Core Cooling Actuation Test

- 2.4 Added this statement under license or administrative requirements - Technical Specifications, Section 3.4, combined heatup cooldown and pressure limitations.
- 4.12 Added statement to procedure - RHR isolation valves are open placing 500# relief valve in service on RCS.
- 5.7 Added statement to procedure - Pressure and temperature limitations of Technical Specifications, Section 3.4 must be adhered to at all times.

SUR 5.1-19 Partial Loss of AC Coincident With Core Cooling Actuation Test

- 2.4 Added this statement under license or administrative requirements - Technical Specifications, Section 3.4 combined heatup cooldown and pressure limitations.
- 4.12 Added statement to procedure - RHR isolation valves are open placing 500# relief valve in service on RCS.
- 5.7 Added statement to procedure - Pressure and temperature limitations of Technical Specifications, Section 3.4 must be adhered to at all times.

NOP 2.0-A (Appendix) Precautions, Limitations and Setpoints

- A-2.6 Original Statement - Based on the reactor vessel DTT limitations, the maximum allowable reactor vessel pressure versus reactor coolant temperature as shown in Figures 1 & 2 must not be exceeded. As a general recommendation throughout the life of the plant, the reactor vessel must not be pressurized above 500 psig unless the temperature is above 90°F.

New Statement - Based on the reactor vessel DTT limitations, the maximum allowable reactor vessel pressure versus reactor coolant temperature as defined in Technical Specifications, Section 3.4, combined heatup, cooldown and pressure limitations must not be exceeded.

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NOV 21 1975

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CONDUCTIVITY-TEMPERATURE DEPENDENT SYSTEM DESIGN LIMITATIONS

Projected to 14 Effective Full Power Years

(T error = 10%, P error = 0.5 PSIG)

Figure 3.4-6

Note 1: Below 300°F pressure is not to exceed 10 PSIG in pressurizer and primary side of steam generators.

NOV 21 1975

CLB, BUILD RCPA

ROC CARDIN 0011 LEV.0

CLR, RCOPA LD XO

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4/ END OF ALL JOBS

Attachment #1

CONNECTICUT YANKEE ATOMIC POWER COMPANY

BERLIN, CONNECTICUT

P. O. BOX 270 HARTFORD, CONNECTICUT 06101

TELEPHONE
203-666-6911

Regulatory

File Cy.

March 21, 1977

Docket No. 50-213

Director of Nuclear Reactor Regulation
Attn: Mr. A. Schwencer, Chief
Operating Reactors Branch #1
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

References: (1) A. Schwencer letter to D. C. Switzer dated February 14, 1977.
(2) D. C. Switzer letter to A. Schwencer dated March 1, 1977.
(3) D. C. Switzer letter to A. Schwencer dated December 3, 1976.

Gentlemen:

Haddam Neck Plant
Reactor Coolant System (RCS) Overpressurization

In Reference (1), CYAPCO was requested to respond to eight questions concerning a Reference Mitigating System (RMS), proposed to provide protection against RCS overpressurization. CYAPCO respectfully notes that in neither References (2) nor (3) was such a system proposed for the Haddam Neck Plant; nevertheless, the comments made by the NRC Staff in Reference (1) are generally applicable to the long-term modifications proposed by CYAPCO and are addressed below.

With regard to Questions 1, 6, and 8, the responses have been provided as part of the information submitted in Reference (2) and, also, in the Facility Description and Safety Analysis (FDSA) for the Haddam Neck Plant. The responses to the remaining questions relate to the assumptions and methods used in the analysis of RCS pressure transients. As noted in Reference (2), the results of this analysis for the Haddam Neck Plant will be submitted to the NRC Staff in the latter half of 1977.

We trust this is responsive to the NRC Staff requests.

Very truly yours,

CONNECTICUT YANKEE ATOMIC POWER COMPANY

D. C. Switzer
D. C. Switzer
President

7708990303

APR 1 1977

Docket No. 50-213

Connecticut Yankee Atomic Power Company
 ATTN: Mr. Donald C. Switzer
 President
 Post Office Box 270
 Hartford, Connecticut 06101

Gentlemen:

RE: HADDAM NECK PLANT

A pressure transient recently occurred at an operating PWR facility when a reactor coolant pump (RCP) was started while in a water-solid condition. At the time, the temperature of the steam generator in that loop was greater than the reactor coolant system (RCS) temperature and a pressure transient from about 400 psig to about 600 psig resulted.

In our January 10, 1977 letter we stated that all PWR licensees should include provisions to prevent RCP starts while in a water-solid condition. The purpose of such provisions is to prevent pressure transients due to temperature differentials within the RCS. We also stated that should it become necessary to start a RCP while water-solid, appropriate steps are to be taken to determine and minimize the temperature profile. Such action was not taken prior to the recently reported event.

Based on this recent pressure transient and our review of PWR submittals describing RCP operating procedures, we are concerned that the significance of this event may not be fully realized.

We, therefore, now request that you implement procedures to prevent RCP starts (or "jogs") while in a water-solid condition when the steam generator temperature is greater than RCS temperature. In the event that a RCP start while water-solid cannot be avoided we request that your plant operating procedures include non-optional pre-requisite steps to ensure that the RCS temperature profile is determined and if that profile is excessive, that appropriate measures are specified to achieve an acceptable profile prior to starting a RCP.

OFFICE →	ORB#1 <i>WTR</i>	ORB#1	ORB#1 <i>AS</i>			
SURNAME →	WTRussell	GZech	ASchweizer			
DATE →	3/31/77	3/1/77	4/1/77			

APR 1 1977

We further request that these procedures be implemented immediately, and that they remain in effect until we have received and approved an analysis justifying any deviations from these provisions. You are requested to advise us within 15 days of receipt of this letter that you have implemented the provisions discussed above.

In addition, it is strongly recommended that you immediately conduct a critical review of your procedures to ensure that they are fully responsive to all of the positions stated in our letter of January 10, 1977 and compatible with the pertinent results of your RCS pressure response analysis.

Sincerely,

Original Signed By

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

cc: See next page

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DATE →	3/31/77	3/ 177	3/1 177

CONNECTICUT YANKEE ATOMIC POWER COMPANY

BERLIN, CONNECTICUT

P. O. BOX 270 HARTFORD, CONNECTICUT 06101

TELEPHONE
803-686-6911

April 26, 1977

CYH 77-141

50-213

U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Attn: Mr. A Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Dear Mr. Schwencer:

In reply to your letter of 4/1/77 Connecticut Yankee Atomic Power Company has taken the following steps:

An Operating Department Instruction has been placed in effect that administratively prevents starting of a reactor coolant pump, with a solid system, unless the secondary side of the steam generator is equal to or no greater than 20°F colder than the primary side temperature. Operating procedures are being revised accordingly.

Analyses are being performed to provide justification for any deviation from the administrative limits which may become necessary. Results of these analyses will be submitted to your office when complete for review and approval.

Very truly yours,

CONNECTICUT YANKEE ATOMIC POWER COMPANY

D. C. Switzer
President

DCS:MSM/jhb

771240106

TELEPHONE
103-862-9911

CONNECTICUT YANKEE ATOMIC POWER COMPANY

BERLIN, CONNECTICUT

P. O. BOX 270 HARTFORD, CONNECTICUT 06101

June 1, 1977

Docket No. 50-213

Director of Nuclear Reactor Regulation
Attn: Mr. A. Schwencer, Chief
Operating Reactors Branch #1
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Reference: (1) D. C. Switzer letter to A. Schwencer dated March 1, 1977.

Gentlemen:

Haddam Neck Plant
Reactor Coolant System (RCS) Overpressurization

In Reference (1), CYAPCO indicated that it was evaluating possible alternative actions regarding interim hardware improvements to protect against RCS overpressurization events. This evaluation was to consider topics such as plant operating restrictions unique to the Haddam Neck Plant and the preliminary results of RCS pressure transient analyses.

As a result of this evaluation and of a concentrated procurement effort, CYAPCO can now state that a hardware improvement can be procured and installed by the end of the fall, 1977 refueling outage. This improvement will consist of the replacement of the two existing relief valves with two new air operated valves and the installation of a programmed setpoint control logic system. Based on the current schedule, detailed design descriptions of the relief valves, the air control system, and the control logic system will be available on or about July 15, 1977. Also, by that time, the CYAPCO evaluation of the proposed modification for the Haddam Neck Plant, in light of the initial results of the recently completed Westinghouse generic pressure transient analyses, will be completed.

CYAPCO would also like to note that discussions with the plant operators concerning recent RCS overpressurization events at other PWR facilities have not yet been completed, as originally planned (see Question #1a in Reference (1)). This is attributable to a delay in getting specific information concerning the similarities and distinctions of the Haddam Neck Plant with respect to other PWR facilities and how those relate to plant operations. It is expected that the requisite information can be obtained, and the training completed, on or about July 15, 1977.

Very truly yours,

CONNECTICUT YANKEE ATOMIC POWER COMPANY

D. C. Switzer
D. C. Switzer
President

771-10012

November 1, 1977

cc: Day, Berry & Howard
Counselors at Law
One Constitution Plaza
Hartford, Connecticut 06103

Russell Library
119 Broad Street
Middletown, Connecticut 16457

Superintendent
Haddam Neck Plant
RFD #1
Post Office Box 127E
East Hampton, Connecticut 06424

REQUEST FOR INFORMATION

1. Section 6.0 of your report titled "Low Temperature RCS Overpressure Protection for Connecticut Yankee" dated August 1977, discussed the design criteria for the long-term overpressure protection system at Haddam Neck. Under the category of "Mass Addition" you list the inadvertent high pressure safety injection (HPSI) pump actuation and injection into a water-solid RCS as one of the "Design Base Transients". However, later in your submittal you eliminate this scenario as a design basis consideration by the implementation of procedural and administrative controls of the HPSI pumps and their associated discharge valves. This is not in accordance with the NRC criteria which requires PWR licensees to include the mass injection from a single HPSI pump as a credible event, and to provide analyses showing the proposed overpressure protection system affords sufficient response time and relief capacity to avoid exceeding the Appendix G limit during this event. Therefore, you should provide an analysis of the RCS pressure response assuming the mass injection from one HPSI Pump into a water-solid RCS to determine the pressure transient which would occur based upon actuation of one spring operated relief valve at backpressures of 10 psig and 100 psig. You should also determine the pressure transient which would occur if both spring operated relief valves operate.
2. Your report briefly discusses the HPSI pump surveillance procedures performed yearly SUR 5.1-18 and 4.1-19. Please describe further the anticipated plant conditions and procedures to be used while performing this surveillance program. Also, describe the surveillance procedures performed to satisfy your Technical Specification 4.3.A; if different from the surveillance procedures described above. You should further describe the proposed procedural and administrative controls used to disable the HPSI pumps and identify the specific valves which are closed and the manner in which the HPSI pumps will be disabled.
3. Discuss the impact upon facility operation of removing power from the motor operated isolation valves upstream of the spring operated relief valves. Removing power from these valves would prevent spurious valve motion from isolating a required path when low temperature overpressure protection is required.

November 1, 1977

TELEPHONE
203-846-6911

CONNECTICUT YANKEE ATOMIC POWER COMPANY

BERLIN, CONNECTICUT

P. O. BOX 270 HARTFORD, CONNECTICUT 06101

November 30, 1977

Docket No. 50-213

Director of Nuclear Reactor Regulation
Attn: Mr. A. Schwencer, Chief
Operating Reactors Branch #1
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

- References: (1) D. C. Switzer letter to A. Schwencer dated September 7, 1977 transmitting "Specific Plant Report, Low Temperature RCS Overpressure Protection".
(2) A. Schwencer letter to D. C. Switzer dated November 1, 1977.

Gentlemen:

Baddam Neck Plant
Additional Information
Low Temperature Reactor Coolant System (RCS)
Overpressurization Protection System

In Reference (2), the NRC Staff requested information in support of the Low Temperature RCS Overpressurization Protection System (OPS) described by CYAPCO in Reference (1). Specifically, requests for information were made concerning the RCS pressure transient for the High Pressure Safety Injection (HPSI) pump start, the annual HPSI surveillance procedures, and the possible disabling of the OPS isolation valves in appropriate operational modes. In response to these requests, CYAPCO hereby submits the attached additional information.

Reference (2) also stated that the NRC Staff had determined, as a result of a preliminary review, that the OPS does not meet all of the NRC Staff-established design criteria; and, thus, cannot be considered as a "permanent" solution. Accordingly, it was requested that CYAPCO propose system modifications and provide a value-impact assessment on the schedule and cost of those modifications. CYAPCO has initiated this assessment; based on a preliminary scoping of this effort, it has been determined that this assessment will be available for submittal on or about March 6, 1978.

CYAPCO also wishes to note that the proposed revisions to the Technical Specifications (discussed in Reference (1)) regarding the disabling of the HPSI pumps in appropriate operational modes, limits on temperature difference

between the RCS and the steam generator, and OPS operability and surveillance requirements will be submitted on or about December 30, 1977 in support of the OPS operation.

Very truly yours,

CONNECTICUT YANKEE ATOMIC POWER COMPANY

D. C. Switzer
President

Attachments

HADDAM NECK PLANT

ADDITIONAL INFORMATION

Question 1

Section 6.0 of your report titled "Low Temperature RCS Overpressure Protection for Connecticut Yankee", dated August, 1977, discussed the design criteria for the long-term overpressure protection system at Haddam Neck. Under the category of "Mass Addition" you list the inadvertent high pressure safety injection (HPSI) pump actuation and injection into a water-solid RCS as one of the "Design Base Transients". However, later in your submittal, you eliminate this scenario as a design basis consideration by the implementation of procedural and administrative controls of the HPSI pumps and their associated discharge valves. This is not in accordance with the NRC criteria which requires PWR licensees to include the mass injection from a single HPSI pump as a credible event, and to provide analyses showing the proposed overpressure protection system affords sufficient response time and relief capacity to avoid exceeding the Appendix G limit during this event. Therefore, you should provide an analysis of the RCS pressure response assuming the mass injection from one HPSI Pump into a water-solid RCS to determine the pressure transient which would occur based upon actuation of one spring operated relief valve at backpressures of 10 psig and 100 psig. You should also determine the pressure transient which would occur if both spring operated relief valves operate.

Response

The results of the analysis of the RCS pressure response assuming the mass injection from one HPSI pump are displayed on Figures 1 and 2.

Figure 1 shows the resultant RCS pressure with one and two OPS spring relief valve(s) operational, assuming a 10 psig backpressure; and Figure 2 shows the results assuming a 100 psig backpressure.

The resultant RCS pressures were obtained by cross-plotting the capacity for one and two OPS spring relief valve(s) and the HPSI pump performance (from Figures 6 and 4, respectively, of Reference (1)) assuming the appropriate backpressure.

Question 2

Your report briefly discusses the HPSI pump surveillance procedures performed yearly SUR 5.1-18 and 4.1-19. Please describe further the anticipated plant conditions and procedures to be used while performing this surveillance program. Also, describe the surveillance procedures performed to satisfy your Technical Specification 4.3.A; if different from the surveillance procedures described above. You should further describe the proposed procedural and administrative controls used to disable the HPSI pumps and identify the specific valves which are closed and the manner in which the HPSI pumps will be disabled.

[NOTE: The correct CY procedure number is 5.1-19]

Response

a) Plant Conditions

The plant would be at a Cold Shutdown Condition at 120°F to 200°F as determined by the loop T_c recorders. The pressurizer level would be below 95% and the pressurizer would be vented.

The RCS would be vented by one of two means during the test: The head removed or one of the high pressure code relief valves would be removed.

The following reactor plant systems would be lined up as shown on Attachment A; Chemical and Volume Control System, Residual Heat Removal System; Safety Injection System, Primary Drain System and the Boric Acid System. The following equipment would be secured during the initial test: the Reactor Coolant Pumps, the residual Heat Removal Pumps, the Circulating Water Pumps (on 4160V Bus 1-2), and the Residual Heat Removal Purification System. All necessary plant equipment would be transferred to the unaffected electrical busses (4160V Busses 3 and 9; 480V Busses 6 and 7) and all unnecessary equipment shutdown.

b) Surveillance Procedure

The procedure used to satisfy the requirements of Specification 4.3.A is as follows:

- (1) Apply a test signal to two of the pressurizer level channels and raise the level indication above the core cooling actuating setpoint of 5.5%.
- (2) Unblock core cooling by holding the core cooling block switch in the RESET position until the alarms clears.

- (3) Open 4160V Bus 1-2 supply Breakers 3891 to initiate partial loss of AC power followed immediately by the initiation of a core cooling signal (simulated 5.5% level in pressurizer by the two test circuits).
- (4) Check and verify that the core cooling valves go to the prescribed position after core cooling initiation.
- (5) Verify that the core cooling systems have started at the prescribed times:

1B LPSI	$\leq$ 13 seconds
1B HPSI	$\leq$ 20 seconds
1B Charging	$\leq$ 30 seconds

- (6) Verify that the pump discharge pressure meets the prescribed requirements:

1B LPSI	$\geq$ 295 psig
1B HPSI	$\geq$ 1400 psig
1B Charging	$\geq$ 2150 psig

The above procedure is used to test the "1B" train LPSI, HPSI, and Charging pumps. A similar procedure is used for the "1A" train of pumps.

c) Disabling HPSI Pumps

The procedures and administrative controls used to disable the HPSI pumps are described as follows:

- (1) When the RCS has cooled down to 260°F to 300°F and the pressure is between 500 psig and 1000 psig, the HPSI pumps are de-energized by either of the following methods:
 - a) Restricting access to the circuit breaker by physically modifying the cabinet door such that racking in and out the breaker can only

be accomplished with the door open. During low temperature RCS operation, the circuit breakers to the HPSI pumps would be racked out, the cabinet door would be closed, and a lock installed on the door.

- b) Racking out the breakers, physically removing the breaker from the cabinet, and locking the cabinet door closed.

Note that the means to lock the cabinet will not be installed and operational until December 30, 1977.

- (2) When the system conditions listed in #1 above are reached, close, lock and red tag the HPSI pump discharge valves (SI-V-855A and B).

Question 3

Discuss the impact upon facility operation of removing power from the motor operated isolation valves upstream of the spring operated relief valves. Removing power from these valves would prevent spurious valve motion from isolating a required path when low temperature overpressure protection is required.

Response

While CYAPCO recognizes the positive considerations regarding overpressurization protection involved in removing power from the motor-operating isolation valves, it is felt that there are potentially serious operational implications from that action.

Removing power from these valves would lead to a higher risk of reactor coolant pump seal damage due to either an inadvertant relief valve actuation or a lack of capability to isolate a train with a relief valve stuck in a partially or fully open position.

Further, the removal of power from the motor-operated isolation valves also removes power from the position indicating circuitry.

In light of this and additional discussions with the NRC Staff, CYAPCO is still currently evaluating the feasibility of removing power from the motor-operated isolation valves. The results of this evaluation will be discussed in the proposed revisions to Technical Specifications to be submitted on or before December 30, 1977.

VALVE LINE-UP FOR PARTIAL LOSS OF AC AND CORE COOLING SURVEILLANCE TEST

SYSTEM	VALVE #	VALVE TITLE	POSITION BEFORE TEST	POSITION AFTER ACTUATION
Chemical and Volume Control	CH-V-206	Charging Header Isolation	Closed	No Change
"	CH-V-264	1A Charging Pump Discharge	Closed	No Change
"	CH-V-267	1A Charging Pump Recirculation	Open	No Change
"	CH-MOV-292B	Charging Line Stop Valve Loop #2	Closed	Open
"	CH-MOV-292C	Charging Line Stop Valve Loop #2	Closed	Open
"	CH-MOV-311	RCP #1 Seal Return	Open	Closed
"	CH-MOV-312	RCP #2 Seal Return	Open	Closed
"	CH-MOV-313	RCP #3 Seal Return	Open	Closed
"	CH-MOV-314	RCP #4 Seal Return	Open	Closed
"	CH-MOV-257	Volume Control Tank Outlet	Open	Closed
"	CH-FCV-110	Charging Flow Control	Throttled	Throttled
"	CH-TV-334	Seal Water To Volume Control Tank Trip	Closed	No Change
"	LD-AOV-202	Letdown Orifice Stop	Closed	No Change
"	LD-AOV-203	Letdown Orifice Stop	Closed	No Change
"	LD-AOV-204	Letdown Orifice Stop	Closed	No Change
"	LD-MOV-200	Letdown Orifice Stop	Closed	No Change
Boric Acid	BA-MOV-373	Refueling Water Storage Tank to Charging Pump Suction	Closed	Open
Primary Drains	DH-V-383	Drain Header to VCR	Open	Open
"	DH-V-407	Drain Header to Primary Drain Tank	Closed	Closed
Residual Heat Removal	RH-MOV-22	Containment Sump	Closed	Closed
"	RH-MOV-21	Refueling Water Storage Tank to RHR Pump	Closed	Closed
"	RH-FCV-602	RHR Flow Controller	Throttled	Open
"	RH-MCV-795	RHR Heat Exchanger Discharge	Throttled	Open
"	RH-MOV-780	RHR Isolation	Open	Open
"	RH-MOV-781	RHR Isolation	Open	Open
"	RH-MOV-803	RHR Isolation	Open	Open
"	RH-MOV-804	RHR Isolation	Open	Open

ATTACHMENT A

Page 2 of 2

VALVE LINE-UP FOR PARTIAL LOSS OF AC AND CORE COOLING SURVEILLANCE TEST

SYSTEM	VALVE #	VALVE TITLE	POSITION BEFORE TEST	POSITION AFTER ACTUATION
Safety Injection	SI-V-105A	1A LPSI Pump Discharge	Closed	Closed
"	SI-V-855A	1A HPSI Pump Discharge	Closed	Closed
"	SI-V-875A	1A HPSI Pump Recirculation	Open	Open
"	SI-V-106A	1A LPSI Pump Recirculation	Open	Open
"	SI-MOV-861A	Loop #1 Safety Injection Stop	Closed	Open
"	SI-MOV-861B	Loop #2 Safety Injection Stop	Closed	Open
"	SI-MOV-861C	Loop #3 Safety Injection Stop	Closed	Open
"	SI-MOV-861D	Loop #4 Safety Injection Stop	Closed	Open
"	CD-MOV-871A	Core Deluge MOV	Closed	Open
"	CD-MOV-871B	Core Deluge MOV	Closed	Open
"	CD-V-873	Core Deluge	Closed	Closed

FIGURE 1

TELEPHONE
803-680-6911

CONNECTICUT YANKEE ATOMIC POWER COMPANY

BERLIN, CONNECTICUT
P. O. BOX 270 HARTFORD, CONNECTICUT 06101

January 3, 1978

Docket No. 50-213

*Received
1/10/78*

Director of Nuclear Reactor Regulation
Attn: Mr. A. Schwencer, Chief
Operating Reactors Branch #1
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

- References: (1) D. C. Switzer letter to A. Schwencer dated September 7, 1977 transmitting "Specific Plant Report, Low Temperature RCS Overpressure Protection".
(2) D. C. Switzer letter to A. Schwencer dated November 30, 1977.

Gentlemen:

**Haddam Neck Plant
Low Temperature Reactor Coolant System (RCS)
Overpressurization Protection System (OPS)
Proposed Revisions to Technical Specifications**

In References (1) and (2), Connecticut Yankee Atomic Power Company (CYAPCO) stated that revisions to the Technical Specifications for the Haddam Neck Plant would be proposed in connection with the operation of the RCS OPS. In accordance with those commitments, CYAPCO hereby proposes to amend its operating license, No. DPR-61, by incorporating the attached proposed revisions. These proposed revisions pertain to the disabling of the High Pressure Safety Injection (HPSI) pumps in appropriate operational modes, limits on the temperature difference (ΔT) between the RCS and the steam generator, and OPS operability and surveillance requirements.

Specifically, additions are proposed to Specification 3.3 to require that the ΔT between the primary and secondary side not exceed 20°F when starting a Reactor Coolant Pump (RCP) and the RCS temperature is below 340°F . Incorporation of this specification would limit the heat addition due to an inadvertent RCP start to values within the design capabilities of the RCS OPS, as analyzed in Reference (1).

A second addition to Specification 3.3 would require that the RCS OPS be in operation when the RCS temperature is below 340°F , unless the RCS is vented through an appropriate size opening. This specification thus requires overpressurization protection, as analyzed in Reference (1), through both of the RCS OPS relief trains or a minimum equivalent vent area. In addition, consideration has been given to conditions under which the RCS OPS is taken out of service during various operational modes. During cold shutdown, venting of the RCS within 24 hours is required

and the inoperable train restored to the operable condition prior to pressurizing. During the hot standby, startup, or power operation modes, seven days are allowed to restore the inoperable train and continue in that mode, since during those modes, an RCS overpressurization transient outside the design capabilities of the RCS OPS is very unlikely to occur; in addition, before proceeding into the cold or hot shutdown modes, the inoperable OPS train must be restored unless a shutdown is dictated by personnel and/or plant safety or another Technical Specification.

Specification 3.6 has been amended to require disabling the HPSI pumps whenever the RCS temperature is below 340°F and the RCS not vented. As indicated in Reference (2), the disabling will be accomplished by racking out the breakers to the pumps, locking the breaker cabinets, and locking the pump discharge valves. These actions assure that the initiation of an RCS overpressurization event due to inadvertent starting of either or both of the HPSI pumps will be substantially minimized.

With respect to ECCS pump testing, a requirement has been added to Specification 4.3 to provide for appropriate venting of the RCS during the core cooling test conducted during each refueling outage. This will assure that the limits of Specification 3.4 are not exceeded during the test.

Finally, additions to Technical Specification Tables 4.2-1 and 4.2-2 provide for calibration and testing of the RCS OPS logic channels each refueling and functional testing of the OPS isolation valves, interlocks, and alarms once per cold shutdown. These proposed revisions assure that the OPS is operable per Specification 3.3 in accordance with the NRC Staff design criteria for testability of the OPS.

A technical review of these proposed revisions has been conducted. It has been concluded that the proposal is technically acceptable and consistent with the analyses documented in Reference (1). A safety evaluation of the proposal has also been performed and has concluded that these additional restrictions do not constitute an unreviewed safety question.

The Connecticut Yankee Nuclear Review Board has reviewed and approved these proposed revisions and concurred in the above determinations.

With respect to the feasibility of removing power from the OPS motor operated isolation valves, the CYAPCO evaluation discussed in Reference (2) has indicated that power be removed from the valves when the OPS is required to be operable. CYAPCO proposes to implement this course of action by opening the breakers to the valve motors. Although, as noted in Reference (2), this would also affect the position indicating circuitry, CYAPCO believes that there would be adequate assurance regarding the position of these isolation valves. This is due, in part, to the fact that the position of the valves would be verified just prior to removing power from the valves; this, in conjunction with plant administrative controls, would assure that the isolation valves are in the correct position.

Very truly yours,

CONNECTICUT YANKEE ATOMIC POWER COMPANY

D. C. Switzer
President

Attach

STATE OF CONNECTICUT)

COUNTY OF HARTFORD)

ss. Berlin

Jan. 3, 1978

Then personally appeared before me D. C. Switzer, who being duly sworn, did state that he is President of Connecticut Yankee Atomic Power Company, the Licensee herein, that he is authorized to execute and file the foregoing information in the name and on behalf of the Licensee herein and that the statements contained in said information are true and correct to the best of his knowledge and belief.

Sheila M. Oates
Notary Public

My Commission Expires March 31, 1981

DOCKET NO. 50-213

HADDAM NECK PLANT
PROPOSED REVISIONS TO TECHNICAL SPECIFICATIONS

3.3 REACTOR COOLANT SYSTEM OPERATIONAL COMPONENTS

Applicability: Applies to the operating status of the reactor coolant system.

Objective: To specify those limiting conditions for operation of the reactor coolant system which must be met to insure safe reactor operation.

Specification:

- A. At least one pressurizer code safety valve shall be in service whenever the reactor is subcritical and the reactor coolant system is above 375°F or 350 psig except during hydrostatic tests.
- B. One or more reactor coolant pumps or the residual heat removal system shall be in operation when changes are made in the boron concentration of the reactor coolant.
- C. The reactor shall not be critical unless the following conditions have been satisfied:
 - (1) Three self-actuated, spring loaded safety valves, having a combined relieving capability of 720,000 #/hr. shall be in service and shall be in accordance with Section VIII of the ASME Boiler and Pressure Code.
 - (2) Above 1 percent of Nominal Operating Power Level, at least one reactor coolant pump operating.
 - (3) Above 10 percent of Nominal Operating Power Level, at least three reactor coolant pumps operating.
 - (4) Above 65 percent of Nominal Operating Power Level. Four reactor coolant pumps operating.
 - (5) One steam generator is capable of performing its heat transfer function.
- D. Each steam generator in a non-isolated reactor coolant loop shall be restored to operable status prior to increasing Tave above 200°F. When starting a reactor coolant pump and the reactor coolant system temperature is below 340°F the temperature of the secondary side shall not be more than 20°F higher than the primary side.
- E. The RCS Overpressure Protection System (OPS) shall be in operation when the RCS temperature is below 340°F unless the RCS is vented through a minimum opening of three (3) inches (nominal diameter) or its equivalent. If one or more of the relief trains is taken out of service and the RCS is not vented, the following actions shall be taken:

be depressurized and vented within 24 hours and the inoperable train shall be restored to the operable condition prior to pressurizing,

- (2) if in the hot standby, startup or power operation modes, the inoperable OPS train shall be restored to operable status within seven (7) days or a report filed stating the reasons why the repairs couldn't be completed within 7 days and an estimate of when the repairs will be completed.
- (3) if in the hot standby, startup or power operation modes, the inoperable OPS train shall be returned to the operable status before cooling down and depressurizing to the Cold or Hot shutdown modes unless personnel and/or plant safety or another Technical Specification requires cooldown to the cold shutdown mode.

Basis:

Each of the pressurizer code safety valves is designed to relieve 240,000 lbs per hr. of saturated steam at the valve set point. They are described more fully in FDSEA Section 5.2.2. Below 375°F and 350 psig in the reactor coolant system, the residual heat removal system can remove decay heat and thereby control system temperature and pressure. If no decay heat were removed by any of the means available, the amount of steam which could be generated at safety valve relief pressure would be less than half the valves' capacity. One valve therefore provides adequate defense against over-pressurization.

When the boron concentration of the reactor coolant system is to be changed, the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the primary system volume in approximately one-half hour.

All pressurizer code safety valves are to be in service prior to criticality to permit the design relieving flow to occur if required.

Part C of the specification requires that a sufficient number of reactor coolant pumps be operating to provide core cooling in the event loss of flow occurs. The flow provided in each case will keep DNB well above 1.30 as discussed in FDSEA Section 10.3.2. Therefore, cladding damage release of fission products to the reactor coolant cannot occur.

By limiting the temperature differential between the primary and secondary sides to twenty (20) degrees in Part D, the resulting pressure transient will be prevented by the RCS OPS (See Reference 1) from exceeding the limits in Specification 3.4.

Basis (Cont'd)

As described in Reference (1), the RCS OPS, in conjunction with administrative controls, prevents exceeding the temperature and pressure limits in Specification 3.4 while RCS temperature is under 340°F. Considerations have been incorporated to provide for operation under various operational modes when one OPS train (relief valve, motor operated isolation valve, and associated instrumentation) is out of service.

Reference (1) D. C. Switzer letter to A. Schwencer, dated September 7, 1977

<u>VALVE NO.</u>	<u>LOCATION</u>	<u>ACTION</u>
RH-MOV-22	Containment Sump Suction	Valve locked in open position, circuit breaker locked out during post-LOCA long term cooling phase.
SI-MOV-24	RWST line	Valve locked in open position, circuit breaker locked out whenever reactor critical.
SI-FCV-875	HPSI mini-flow line	Valve blocked and locked in open position whenever reactor critical.
RH-MOV-874	RHR recirculation	Valve locked in closed position and circuit breaker locked open whenever reactor is critical and reactor coolant temperature is above 350 F

C. The following actions shall be taken to disable the High Pressure Safety Injection Pumps whenever the RCS temperature is below 340 F and the RCS is not vented by a minimum opening of three (3) inches (nominal diameter) or its equivalent.

1. De-energize the HPSI pumps by racking out the breakers and locking the cabinets.

2. Close and lock the HPSI pump discharge valves (ST-V-855A & B)

BASIS

This specification assures that adequate emergency core cooling capacity is available whenever the reactor is critical. Based on the loss of coolant analysis, melting of the cladding is prevented with only one high pressure safety injection pump, one low pressure safety injection (core deluge) pump and one charging pump in operation. Each of the two trains of emergency core cooling equipment includes these three pumps. With the pumps associated with both trains of emergency core cooling equipment operable, substantial margin exists whenever normal power supplies or both diesel generators are available. With only one diesel generator operating and the pumps associated with that diesel operable as required in Item (2) of Specification 3.12, the necessary three pumps would be started. With the exception of the RHR pump, which is used only for long term recirculation, all of the specified components start automatically when the safety injection signal is given.

The placing of the valves in section "B" in the above positions will provide protection against "fail to function" and "undesirable function" as stated in NRC Branch Technical Position EICS #18.

As described in Reference 3, the RCS OPS, in conjunction with administrative controls, prevents exceeding the temperature and pressure limits in specification 3.4 while RCS temperature is under 340°F or the RCS is not vented. Part C establishes a limiting condition for operation regarding the disabling of the HPSI pumps to further assure that a pressure transient is not initiated.

Reference: (1) FDSEA Section 5.2.7

(2) FDSEA Section 5.2.8

(3) D. C. Switzer letter to A. Schwencer dated September 7, 1977

4.3 CORE COOLING SYSTEMS - PERIODIC TESTING

Applicability: Applies to periodic testing requirements for the core cooling system.

Objective: To verify the operability of the core cooling systems

Specification: A) During each refueling shutdown, a test shall be conducted to demonstrate proper automatic operation of the core cooling systems. The test shall be performed in accordance with written procedure as summarized below:

- 1) The two independent core cooling systems will be tested separately. A test signal will be applied to initiate a loss of normal AC power to each of the emergency power systems while a coincident signal is applied which initiates operation of the emergency core cooling equipment associated with the system being tested.
- 2) Verification shall be made that the diesel generator and its associated pumps have started in the proper sequence and that the high pressure safety injection, low pressure safety injection (core deluge) and charging pump attain required discharge heads of 1400 psig, 295 psig and 2150 psig, respectively. Following this check of the automatic sequence, either a charging or safety injection pump will be secured and the associated RHR pump will be started manually. Verification will be made that the RHR pump attains the required discharge head of 145 psig.
- 3) The test will be considered satisfactory if control board indication and visual observations indicate all components have operated and sequenced properly.
- 4) A prerequisite for this test shall be that either:
 - a) the RCS shall be vented by a 3" or larger opening and the RCS OPS shall be inservice, or
 - b) a minimum of two three inch openings shall be used for venting if both trains of the RCS OPS are out of service.

- B) During normal operating periods, a manual test of all actuated components shall be conducted to demonstrate operability. The test shall be performed monthly in accordance with written procedure as summarized below:
- 1) Each of the high pressure safety injection pumps, each of the low pressure safety injection (core deluge) pumps, and each of the residual heat removal (RHR) pumps shall be individually test run on recirculation.
 - 2) The charging pumps and metering pump shall be individually test run.
 - 3) All safety injection and core deluge valves will be cycled under "no-flow" conditions.
- C) If one of the high pressure safety injection pumps or one of the low pressure safety injection (Core deluge) or one of the residual heat removal (RHR) pumps is out of service, the remaining pump shall be tested within 2 hours and at subsequent intervals of not greater than 72 hours.
- D) During each refueling shutdown, the remotely controlled, motor-operated containment spray water valve shall be operated under a "no-flow" condition. The test will be considered satisfactory if visual observation shows that the valve has operated satisfactorily.

Basis:

The core cooling systems are the principal plant safeguard. They provide the means to insert negative reactivity and limit core damage in the event of a loss-of-coolant incident.

Pre-operational performance tests of the components are performed in the manufacturer's shop. An initial system flow test demonstrates proper dynamic functioning of the system. Thereafter, periodic tests demonstrate that all components are functioning properly.

In order to assure that a pressure transient occurring during the testing of the HPSI pumps will not exceed the pressure and temperature limits of specification 3.4, there must be appropriate relief paths available; this is provided for in specification 4.3.A.4.

The separation of emergency power systems and associated core cooling equipment into two independent groupings permits complete functional testing of the individual systems and equipment.

Table 4.2-1 (continued)

<u>Channels</u>	<u>Action</u>	<u>Minimum Frequency</u>
8. Variable Low Pressure trip set point calculator	Calibrate Check	Each refueling Each shift
9. Rod Position Digital Voltmeter	Calibrate Check with counters	Each refueling Every six inches of rod motion when data logger is out of service.
10. Rod Position Counters	Test Check with Digital Voltmeter	Each refueling Every six inches of rod motion when data logger is out of service.
11. Steam generator level	Calibrate Check	Each refueling Each shift
12. Steam generator flow mismatch	Calibrate Check	Each refueling Each shift
13. Charging Flow	Calibrate	Each refueling
14. Residual Heat Pump Flow	Calibrate	Each refueling
15. Boric Acid Tank Level	Calibrate Check	Each refueling Each week
16. Refueling Water Storage Tank Level	Calibrate Test	Each refueling 90 days
17. Volume Control Tank Level	Calibrate Test	Each refueling 90 days
18. Reactor Containment Pressure	Calibrate Check	Each refueling Each day
19. Radiation Monitoring System	Calibrate Test	Each refueling Each day
20. Boric Acid Control	Calibrate	Each refueling
21. Containment Sump Level	Calibrate	Each refueling
22. Valve Temperature Interlocks	Test	Each refueling
23. Pump-Valve Interlock	Check	Each refueling
24. Reactor Coolant System OPS	Calibrate Test	Each refueling Each refueling

Table 4.2-2

Minimum Equipment Check and Sampling Frequency

	<u>Check</u>	<u>Frequency</u>
1. Reactor Coolant Samples	Radio-chemical analysis	5 days/week
2. Reactor Coolant Boron	Boron concentration	5 days/week
3. Refueling Water Storage Tank Water Sample	Boron concentration	Weekly
4. Control Rods	Check rod drop times of all rods to be less than 2.5 seconds	Each Refueling
5. Control Rod	Partial movement of all rods	Every 2 weeks
6. Pressurizer Safety Valves	Check set point	Each refueling
7. Main Steam Safety Valves	Check set point	Each refueling
8. Main Steam Isolation Valves	Check Functioning	Each refueling
9. Reactor Containment Trip Valves	Check Functioning	Each refueling
10. Refueling System Interlocks	Check Functioning	Each refueling
11. Boric Acid Pumps	Test Run Pumps	Weekly
12. RCS Overpressure Protection System Isolation Valve Interlocks and Alarms	Check Functioning	Once/Cold Shutdown

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BERLIN, CONNECTICUT

P. O. BOX 270 HARTFORD, CONNECTICUT 06101

March 6, 1978

Docket No. 50-213

Director of Nuclear Reactor Regulation
Attn: Mr. D. L. Ziemann
Operating Reactors Branch #2
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

- References:
- (1) D. C. Switzer letter to A. Schwencer dated November 30, 1977.
 - (2) D. C. Switzer letter to A. Schwencer dated January 3, 1978.
 - (3) D. C. Switzer letter to A. Schwencer dated September 7, 1977, transmitting "Specific Plant Report, Low Temperature RCS Overpressurization Protection".
 - (4) A. Schwencer letter to D. C. Switzer dated November 1, 1977.

Gentlemen:

Haddam Neck Plant
Low Temperature Reactor Coolant System (RCS)
Overpressurization Protection System

In Reference (4), the NRC Staff requested that Connecticut Yankee Atomic Power Company (CYAPCO) provide information in support of the Low Temperature RCS Overpressurization Protection System (OPS) for the Haddam Neck Plant described in Reference (3). In References (1) and (2), CYAPCO provided Additional Information and proposed Technical Specifications in response to the NRC Staff requests.

In addition, the NRC Staff requested that CYAPCO propose system modifications and provide a value-impact assessment on the schedule and cost of those modifications. In response to that request, CYAPCO hereby submits the attached additional information describing the reevaluation of the OPS and the value-impact assessment of system modifications, in accordance with NRC Staff criteria.

As detailed in the Attachment, CYAPCO believes that the original proposed RCS-OPS is justified, on a value-impact basis, as the long-term solution to postulated low temperature RCS overpressurization transients.

Very truly yours,

CONNECTICUT YANKEE ATOMIC POWER COMPANY

D. C. Switzer
President

Attachment

DOCKET NO. 50-213

HADDAM NECK PLANT

ADDITIONAL INFORMATION

LOW TEMPERATURE REACTOR COOLANT SYSTEM (RCS)

OVERPRESSURIZATION PROTECTION SYSTEM

INCLUDING

VALUE-IMPACT ASSESSMENT

MARCH 1978

1.0 INTRODUCTION

This submittal is forwarded in response to the additional information requested in Mr. A. Schwencer's letter to Mr. D. C. Switzer, dated November 1, 1977. In addition, the information herein supplements that data previously submitted to the NRC Staff in correspondence addressed to Mr. A. Schwencer from the Connecticut Yankee Atomic Power Company (CYAPCO), dated September 7, 1977, November 30, 1977, and January 3, 1978.

The information presented herein describes: (1) changes already made to administrative controls and/or proposed technical specifications as part of a solution to NRC Staff concerns; (2) a reevaluation of the relief capacity of the newly installed RCS Overpressure Protection System (OPS); (3) a revised HPSI pump injection transient and RCS system response with respect to peak pressure; and (4) a value impact evaluation of additional system modifications in accordance with NRC Staff criteria.

As discussed in previous correspondence, the Haddam Neck Plant has several significant plant specific characteristics which affect the method by which RCS overpressurization transients are mitigated. For all transients, except as noted below, CYAPCO has provided sufficient relief capacity to limit the RCS pressure below the applicable 10CFR50, Appendix G curves. In the case of the HPSI pump start scenario, this accomplishment is impracticable due to the high capacity of the pump. Specifi-

cally, the NRC criteria requires PWR licensees to include the mass injection from a single HPSI pump as a credible event, and to provide analyses showing that the OPS affords sufficient response time and relief capacity to avoid exceeding the Appendix G limit during this event. In the case of Haddam Neck, it is the opinion of CYAPCO that administrative controls (including technical specifications and associated operator actions) reduce the chance of inadvertant HPSI pump start to such an exceedingly small level, that this transient is not deemed a credible event. Notwithstanding this, there is sufficient relief capacity to limit RCS peak pressures well below hydro limits if a HPSI pump did by some means start.

Therefore, as detailed in the following sections, CYAPCO considers the original CYAPCO proposal justified on a value-impact basis as the long term solution to postulated low temperature RCS overpressurization transients.

2.0 ADMINISTRATIVE CONTROLS/TECHNICAL SPECIFICATIONS

As described in References 1 and 2, CYAPCO has implemented administrative controls and proposed technical specifications to establish requirements for OPS operability, limiting the ΔT between the primary and secondary sides of the steam generators, and disabling the HPSI pumps under certain operating modes.

The specific requirements include:

- 2.1 When the RCS has cooled down to 260°F to 300°F and the pressure is between 500 psig and 1000 psig, the HPSI pumps are deenergized by racking out the breakers, physically removing the breaker from the cabinet, and locking the cabinet door closed. (See Reference 1.)
- 2.2 When the system conditions listed in 2.1 above are reached, the HPSI pump discharge valves (SI-V-855A and B) are closed, locked and red tagged. (See Reference 2h.)
- 2.3 Revised Technical Specification 3.6 has been proposed to require disabling of the HPSI pumps whenever the RCS temperature is below 340°F and the RCS is not vented, as noted in 2.1 and 2.2 above. (See Reference 2c.)
- 2.4 A requirement has been proposed for Specification 4.3 to provide for appropriate venting of the RCS during scheduled ECCS pump testing when the RCS temperature is less than 340°F and solid. A single three inch (3") nominal vent path (i.e., removal of a code safety valve) provides relief capacity in excess of that required to eliminate the HPSI pump injection transient. (See Reference 2c.)

3.0 RCS INJECTION AND RELIEF RATES

As part of the value-impact assessment, CYAPCO has refined its original evaluation of the plant specific characteristics affecting mass injection rates and relief flows. Specifically, various system pressure drops were determined which result from frictional losses due to piping configuration and their effect on flow rates. The results are presented below:

- 3.1 The pressure drop between the pressurizer and OPS spring relief valves does not result in RCS pressures which exceed Appendix G limits for flow rates of approximately 550 to 575 gpm for a single OPS relief train or approximately 680 gpm for simultaneous relief through both trains. Consequently, the spring relief valves, with relief capacities and characteristics as previously reported, are not the limiting factor in the determination of system capabilities.
- 3.2 The evaluation performed for the charging and relief systems show that the OPS has the relief capability to mitigate the charging-letdown mismatch pressure transient.
- 3.3 In Reference 2b CYAPCO provided its original analysis of the RCS pressure response, assuming the mass injection from the HPSI pump into a water-solid RCS, to determine

the pressure transient which would occur based upon actuation of one and two spring operated relief valve(s) at back pressures of 10 psig and 100 psig.

The revised figures herein showing the resultant RCS pressures were obtained by adjusting the relief valve capacity curves and HPSI pump discharge curve for reduced flows resulting from friction losses.

Figure 1A shows the resultant RCS pressure with one and two OPS spring relief valve(s) operational, assuming a 10 psig back pressure; Figure 2A shows the results assuming a 100 psig back pressure.

It should be noted that peak RCS system pressures are significantly less than originally represented based on the actual mass injection rates at the RCS boundary as compared to the HPSI pump discharge rates from its performance curve without regard for system configuration and resultant system resistance losses.

4.0 ADDITIONAL SYSTEM MODIFICATIONS

CYAPCO has reviewed system modifications deemed practicable which would provide additional RCS overpressurization relief capacity for the complete mitigation of the HPSI pump injection transient. It has been determined that the most practical method of achieving this goal would be to install a relief

piping train, similar to those presently installed, which would tie into a lateral between the pressurizer and a code safety valve and discharge into the ten inch (10") discharge pipe as near to the pressurizer relief tank as feasible. This installation would provide a direct and independent relief path from the RCS without affecting the safety related functions of other equipment.

Alternative modifications to provide increased relief capacity which were considered included (1) increasing the capacity of the existing OPS, and (2) installation of a relief train in the HPSI piping between the pump and RCS loop. These alternatives were not considered in further detail since (1) they would require more extensive piping rework than the more practical method because of the present flow limiting piping configuration, or (2) they could affect the safety related function of the ECCS. Modifications to utilize the RHR system were previously evaluated and rejected.

In addition to the consideration of alternatives to increase relief capacity, an evaluation of possible HPSI system modifications was conducted to consider the viability of either limiting HPSI pump flow or tripping the HPSI pump in the event of an inadvertant injection. The hardware changes, control circuitry and added administrative requirements were determined to degrade the ECCS safety function on the basis

of: (1) reducing the system's capability to limit fuel rod cladding temperatures in the event of a postulated accident, and (2) reduced reliability, and were, therefore, considered not to be viable alternatives.

5.0 COST AND SCHEDULE

CYAPCO has currently spent in excess of \$1,000,000 for the present RCS Overpressure Protection System. The additional expenses which would be incurred as a result of the installation of the additional relief capacity described in Section 4.0 are estimated to be approximately \$400,000 based on a comparison of the scopes of work involved, exclusive of both ALARA and replacement power cost.

As part of the value impact consideration and our ALARA Occupational Exposure Program, the dollar value for each occupational radiation exposure person-rem must be included in the above estimate.

CYAPCO has placed the total dollar value of an occupational exposure person-rem at \$5,000, which includes both the manpower replacement cost and the societal cost. Consequently, the estimate increases to a total of \$675,000, including the cost of approximately 1100 manhours of work in a radiation zone of 50 milli-rem per hour.

The earliest practical installation schedule, contingent upon the availability of components and their deliveries, would be the next refueling outage, currently scheduled for early 1979. It should be noted that the estimated cost presented above is based on a non-critical path schedule and that the performance of the subject modification on a critical path or forced outage basis would result in additional costs for replacement energy of approximately \$300,000 per day.

6.0 VALUE IMPACT ASSESSMENT

As previously presented, CYAPCO proposes that the presently installed OPS and the Technical Specifications submitted in our correspondence of January 3, 1978 provide a combination of hardware and software measures which adequately serve the function of RCS overpressurization protection. The OPS limits peak RCS pressures below hydro limits during the HPSI pump injection transient, while the probability of the event is minimized (if not eliminated) by the requirements established in the Technical Specifications.

Evaluating that portion of plant life during which time the RCS is vulnerable to a HPSI pump injection and the probability of the occurrence which would result in this event diminishes the significance of the transient dramatically.

The Haddam Neck Plant is licensed to operate into the year 2004, which establishes the plant's remaining life at twenty-six (26) years or approximately sixty percent (60%) of the design life. Based on the average period of fourteen (14) months between refuelings, twenty-two (22) scheduled shutdowns are expected during the balance of plant life. CYAPCO has also assumed that there will be one (1) additional forced outage between each of the scheduled outages. This figure is considered conservative since the majority of the off-the-line time is expended with the nuclear plant in hot standby. As a result, there are forty-three (43) shutdown (heatup and cooldown) cycles during which the Haddam Neck Plant will be in an operational mode susceptible to the HPSI transient. It should also be noted that during the major portion of the shutdown period, the RCS is in a vented condition (i.e., not solid and closed).

The analysis of administrative and procedural errors which could lead to the activation of a HPSI system during undesirable conditions was performed on the basis of human reliability since the primary or contributing cause of the HPSI pump injection is the result of incorrect or lack of correct operator action. The probability of each event/error and the sequence of events which would result in the activation of a HPSI pump is presented in the Overpressure Condition Fault Tree

in Figure 3. The fault tree shows the probability of the HPSI Pump Overpressurization Transient per cooldown/heatup cycle. Multiplying by the number of cycles assumed over the life of the plant leads to a value of $.46 \times 10^{-6}$ for the probability of the event taking place.

The number of operations and the type that are performed in each cycle are approximate, but conservative. Probabilities were calculated from data in Reference 3, Table III 6-1 and are shown in Table 1. Additionally, the probabilities presented herein are considered conservative since a second individual/shift supervisor is present in the control room to note and correct the errors and thus reduces the probability of occurrence.

Momentarily disregarding the credibility of the event, the peak RCS pressure resulting from the HPSI injection are plotted against the Connecticut Yankee Limit Curves for Hydrostatic and Leak Testing in Figure 4. CYAPCO considers the comparison of HPSI transient pressure to hydro pressure both relevant and significant in determining any cumulative impact of the given transient on the lifetime cyclic transients to which the reactor vessel is subjected. Based on the assumption that a hydrostatic test is performed during each of the remaining twenty-two (22) scheduled outages in the plant's life, the slight probability of the HPSI transient and the considerably smaller magnitude of pressure to which

the RCS is subjected are deemed negligible, in relation to the slope of permissible pressures in the Appendix G Hydrostatic Curves.

In conclusion, CYAPCO does not consider the HPSI pump injection transient a credible event based on the Fault Tree Analysis presented above and the 10^{-6} probability value for the postulated HPSI injection incident. The Overpressure Protection System presently installed at the Haddam Neck Plant in conjunction with administrative measures adequately protects the Reactor Coolant System against all low temperature overpressurization transients, including the HPSI pump injection.

CYAPCO does not consider the installation of additional relief capacity costing in excess of \$600,000 justified on a value impact basis, since the additional protection would provide inconsequential benefit based on risk considerations.

Consequently, CYAPCO considers the original proposal as the long term solution to postulated low temperature RCS overpressurization transients justified on a technical and value impact basis. Hence, no further action is believed necessary by CYAPCO to mitigate the water solid reactor coolant overpressurization events.

REFERENCES

1. Specific Plant Report, Low Temperature RCS Overpressure Protection for Connecticut Yankee. Docket No. 50-213.
2. CYAPCO Letters to NRC, Docket No. 50-213, dated:
(a) September 7, 1977; (b) November 30, 1977; and
(c) January 3, 1978.
3. WASH-1400, Reactor Safety Study - An Assessment of Accident Risks in U. S. Commercial Nuclear Power Plants, October 1975.
4. Facility Description and Safety Analysis, Connecticut Yankee Atomic Power Company, Haddam Neck Plant.

HADDAM NERK PLANT
 ADDITIONAL INFORMATION
 RCS OVERPRESSURE PROTECTION
 SOCKET NO. 50-213

FIGURE 1A

HADDAM NECK PLANT
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FIGURE 2A

FIGURE 3
OVERPRESSURE CONDITION FAULT TREE

GIVEN: 43 CYCLES REMAINING IN LIFE OF PLANT

∴ PROBABILITY OF EVENT OVER LIFE OF PLANT

$$= (1.08 \times 10^{-8} \text{ PER CYCLE})(43 \text{ CYCLES})$$

$$= .46 \times 10^{-6}$$

TABLE 1

PROBABILITIES DEVELOPED AS FOLLOWS

1. PROBABILITY OF ACCIDENTAL ACTUATION OF HPSI
PUMP SWITCH ON CONTROL BOARD C (EACH)

NUMBER OF OPERATIONS ON ADJACENT BOARD D TOTAL 72 EACH COOLDOWN-STARTUP CYCLE.

24 SIMILAR SWITCH--DISSIMILAR LOCATION

PROBABILITY OF ACTUATING HPSI SWITCH -- 24×10^{-3}

12 DISSIMILAR SWITCH--ADJACENT LOCATION

PROBABILITY OF ACTUATING HPSI SWITCH -- 12×10^{-3}

36 DISSIMILAR SWITCH--DISSIMILAR LOCATION

PROBABILITY OF ACTUATING HPSI SWITCH -- 36×10^{-6}

NUMBER OF OPERATIONS ON BOARD C TOTAL 28 EACH COOLDOWN-STARTUP CYCLE.

12 SIMILAR SWITCH--DISSIMILAR LOCATION

PROBABILITY OF ACTUATING HPSI SWITCH -- 12×10^{-3}

2 DISSIMILAR SWITCH--ADJACENT LOCATION

PROBABILITY OF ACTUATING HPSI SWITCH -- 2×10^{-3}

14 DISSIMILAR SWITCH--DISSIMILAR LOCATION

PROBABILITY OF ACTUATING HPSI SWITCH -- 14×10^{-6}

AVERAGE PROBABILITY OF ACTUATING HPSI SWITCH GIVEN THE WRONG SWITCH FOR A GIVEN OPERATION IS ACTUATED.

$$\frac{50.05 \times 10^{-3}}{100 \text{ TOTAL OPERATIONS}}$$

$$P = .5 \times 10^{-3} \text{ PER CYCLE}$$

2. PROBABILITY OF WRONG SWITCH BEING OPERATED FOR GIVEN CYCLE

100 TOTAL OPERATIONS $\times 3 \times 10^{-3}$ PER OPERATION

$$P = 3 \times 10^{-1} \text{ PER CYCLE}$$

3. PROBABILITY OF RACKING OUT WRONG BREAKER

$$3 \times 10^{-3}$$

4. PROBABILITY OF FORGETTING TO RACK OUT BREAKER OR LOCK OUT VALVE

$$3 \times 10^{-3}$$

5. PROBABILITY OF LOCKING OUT VALVE IN INCORRECT STATE

$$3 \times 10^{-3}$$

FIGURE 4

HADDON NECK PLANT

ADDITIONAL INFORMATION

ACS OVERPRESSURE PROTECTION

DOCKET NO. 50-213

OPS PRESSURE LIMITING CAPABILITY

HFST FWH INJECTION TRANSIENT

CONNECTIONS FROM UNIT CURVES FOR STEADY STATE AND TRANSIENT
FOR INDICATED EFFECTIVE FULL POWER (ASSET) OF OPERATION