TECHNICAL EVALUATION REPORT

BWR SCRAM DISCHARGE VOLUME LONG-TERM MODIFICATIONS (B-58)

NORTHEAST UTILITIES

MILLSTONE NUCLEAR POWER STATION UNIT NO. 1

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FRC TASK 61

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FOREWORD

This Technical Evaluation Report was prepared by Franklin Research Center under a contract with the U.S. Nuclear Regulatory Commission (Office of Nuclear Reactor Regulation, Division of Operating Reactors) for technical assistance in support of NRC operating reactor licensing actions. The technical evaluation was conducted in accordance with criteria established by the NRC.

SUMMARY

This technical evaluation report reviews and evaluates proposed Phase 1 changes in the Millstone Nuclear Power Station Unit No. 1 Technical Specifications for scram discharge volume (SDV) long-term modifications regarding surveillance requirements for SDV vent and drain valves and the limiting condition for operation (LCO)/surveillance requirements for reactor protection system and control rod withdrawal block SDV limit switches. Conclusions were based on the degree of compliance of the Licensee's submittal with criteria from the Nuclear Regulatory Commission (NRC) staff's Model Technical Specifications.

The proposed revision of page 3/4 3-4, paragraph 2, of the Millstone Unit No. 1 Technical Specifications complies with the requirements of paragraph 4.1.3.1.1a of the NRC staff's Model Technical Specifications.

The Licensee's In-Service Inspection Program, which is already incorporated into the Millstone Nuclear Power Station Unit No. 1 Technical Specifications per Section 4.13, provides for cycling the SDV drain and vent valves once every 3 months, and meets the NRC staff's Model Technical Specifications requirements of paragraph 4.1.3.1.1b.

The Licensee's agreement to (1) revise the frequency of the "Instrument Functional Test" for control rod withdrawal block, in Table 4.2.1 on revised page 3/4 2-6, from "Refueling Outage" to "Quarterly," and (2) provide calibration at each refueling outage for SDV water level-high, meets the NRC staff's Model Technical Specifications requirements of paragraph 4.3.6 and Table 4.3.6-1. It does not meet the requirements for control rod withdrawal block SDV scram trip bypassed "Instrument Functional Test," which should be performed monthly, not quarterly. However, the Licensee is installing a second instrument volume containing six additional limit switches. This significantly increases the reliability of the SDV system and provides a technical basis for acceptance of quarterly performance of the "Instrument Functional Test" for control rod withdrawal block SDV scram trip bypassed.

The remaining surveillance requirements are met by:

a. pages 3/4 2-5 and 3/4 2-6 after their second revision*

b. revised pages 3/4 3-4 and 3/4 3-5

c. added pages 3/4 2-5a and 3/4 3-4a

d. pages 3/4 1-2 (Table 3.1.1), 3/4 1-5 (Table 4.1.1), and 3/4 1-7 (Table 4.1.2) without revision.

Table 5-1 on pages 23 and 24 of this report summarizes the evaluation results.

*The Licensee agreed to revise the frequency of the "Instrument Functional Test" in Table 4.2.1 from "Refueling Outage" to "Quarterly".

1. INTRODUCTION

1.1 PURPOSE OF THE TECHNICAL EVALUATION

The purpose of this technical evaluation report (TER) is to review and evaluate the proposed changes in the Technical Specifications of the Millstone Nuclear Power Station Unit No. 1 boiling water reactor (BWR) in regard to "BWR Scram Discharge Volume Long Term Modification," specifically:

- surveillance requirements for scram discharge volume (SDV) vent and drain valves
- limiting condition for operation (LCO)/surveillance requirements for the reactor protection system limit switches
- LCO/surveillance requirements for the control rod withdrawal block SDV limit switches.

The evaluation used criteria proposed by the NRC staff in Model Technical Specifications (see Appendix A of this report). This effort is directed toward the NRC objective of increasing the reliability of installed BWR scram discharge volume systems, the need for which was made apparent by events described below.

1.2 GENERIC ISSUE BACKGROUND

On June 13, 1979, while the reactor at Hatch Unit 1 was in the refuel mode, two SDV high level switches had been modified, tested, and found inoperable. The remaining switches were operable. Inspection of each inoperable level switch revealed a bent float rod binding against the side of the float chamber.

On October 19, 1979, Brunswick Unit 1 reported that water hammer due to slow closure of the SDV drain valve during a reactor scram damaged several pipe supports on the SDV drain line. Drain valve closure time was approximately 5 minutes because of a faulty solenoid controlling the air supply to the valve. After repair, to avoid probable damage from a scram, the unit was started with the SDV vent and drain valves closed except for periodic draining. During this mode of operation, the reactor scrammed due to a high water level in the

SDV system without prior actuation of either the high level alarm or rod block switch. Inspection revealed that the float ball on the rod block switch was bent, making the switches inoperable. The water hammer was reported to be the cause of these level switch failures.

As a result of these events involving common-cause failures of SDV limit switches and SDV drain valve operability, the NRC issued IE Bulletin 80-14, "Degradation of BWR Scram Discharge Volume Capability," on June 12, 1980 [1]. In addition, to strengthen the provisions of this bulletin and to ensure that the scram system would continue to work during reactor operation, the NRC sent a letter dated July 7, 1980 [2] to all operating BWR licensees requesting that they propose Technical Specifications changes to provide surveillance requirements for reactor protection system and control rod block SDV limit switches. The letter also contained the NRC staff's Model Technical Specifications to be used as a guide by licensees in preparing their submittals.

Meanwhile, during a routine shutdown of the Browns Ferry Unit 3 reactor on June 28, 1980, 76 of 185 control rods failed to insert fully. Full insertion required two additional manual scrams and an automatic scram for a total elapsed time of approximately 15 minutes between the first scram initiation and the complete insertion of all the rods. On July 3, 1980, in response to both this event and the previous events at Hatch Unit 1 and Brunswick Unit 1, the NRC issued (in addition to the earlier IE Bulletin 80-14) IE Bulletin 80-17 followed by five supplements [3-8]. These initiated short-term and long-term programs described in "Generic Safety Evaluation Report BWR Scram Discharge System," NRC staff, December 1, 1980 [9] and "Staff Report and Evaluation of Supplement 4 to IE Bulletin 80-17 (Continuous Monitoring Systems)" [10].

Analysis and evaluation of the Browns Ferry Unit 3 and other SDV system events convinced the NRC staff that SDV systems in all BWRs should be modified to assure long-term SDV reliability. Improvements were needed in three major areas: SDV-IV hydraulic coupling level instrumentation, and system isolation. To achieve these objectives, an Office of Nuclear Reactor Regulation (NRR) task force and a subgroup of the BWR Owners Group developed Revised Scram Discharge System Design and Safety Criteria for use in establishing acceptable SDV

systems modifications [9]. Also, an NRC letter dated October 1, 1980 requested all operating BWR licensees to reevaluate installed SDV systems and modify them as necessary to comply with the revised criteria.

In Reference 9, the SDV-IV hydraulic coupling at the Big Rock Point plant, Brunswick Units 1 and 2, the Duane Arnold plant, and Hatch Units 1 and 2 BWRs was judged acceptable. The remaining BWRs will require modification to meet the revised SDV-IV hydraulic coupling criteria, and all operating BWRs may require modification to meet the revised instrumentation and isolation criteria. The changes in Technical Specifications associated with this effort will be carried out in two phases:

Phase 1 - Improvements in surveillance for vent and drain valves and instrument volume level switches.

Phase 2 - Improvements required as a result of long-term modifications made to comply with revised design and performance criteria.

This TER is a review and evaluation of Technical Specifications changes proposed for Phase 1.

1.3 PLANT-SPECIFIC BACKGROUND

The July 7, 1980 NRC letter [2] not only requested all BWR licensees to amend their facilities' Technical Specifications with respect to control rod drive SDV capability, but enclosed the NRC staff's proposed Model Technical Specifications (see Appendix A of this TER) as a guide for the licensees in preparing the requested submittals and as a source of criteria for a technical evaluation of the submittals. This TER is a review and evaluation of Technical Specifications changes for the Millstone Nuclear Power Station Unit No. 1 proposed by the Licensee, Northeast Utilities (NU), in letters dated October 14, 1980 and March 9, 1982 (see Appendices B and C) in regard to "BWR Scram Discharge Volume (SDV) Long-Term Modifications" and, specifically, the surveillance requirements for SDV vent and drain valves and the limiting condition for operation (LCO)/surveillance requirements for the reactor protection system and control rod withdrawal block SDV limit switches. The

TER also assesses the adequacy with which the NU information documented compliance of the proposed Technical Specifications changes with the NRC staff's Model Technical Specifications.

2. REVIEW CRITERIA

The criteria established by the NRC staff's Model Technical Specifications involving surveillance requirements of the main SDV components and instrumentation cover three areas of concern:

- o surveillance requirements for SDV vent and drain valves
- o LCO/surveillance requirements for reactor protection system SDV limit switches
- o LCO/surveillance requirements for control rod block SDV limit switches.

2.1 SURVEILLANCE REQUIREMENTS FOR SDV DRAIN AND VENT VALVES

The surveillance criteria of the NRC staff's Model Technical Specifications for SDV drain and vent valves are:

"4.1.3.1.1 - The scram discharge volume drain and vent valves shall be demonstrated OPERABLE by:

a. Verifying each valve to be open* at least once per 31 days and

b. Cycling each valve at least one complete cycle of full travel at least once per 92 days (quarterly).

*These valves may be closed intermittently for testing under administrative controls."

The Model Technical Specifications require testing the drain and vent valves, checking at least once every 31 days that each valve is fully open during normal operation, and cycling each valve at least one complete cycle of full travel under administrative controls at least once per 92 days.

Full opening of each valve during normal operation indicates that there is no degradation in the control air system and its components that control the air pressure to the pneumatic actuators of the drain and vent valves. Cycling each valve checks whether the valve opens fully and whether its movement is smooth, jerky, or oscillatory.

During normal operation, the drain and vent valves stay in the open position for very long periods. A silt of particulates such as metal chips

and flakes, various fibers, lint, sand, and weld slag from the water or air may accumulate at moving parts of the valves and temporarily "freeze" them. A strong breakout force may be needed to overcome this temporary freeze, producing a violent jerk which may induce a severe water hammer if it occurs during a scram or a scram resetting. Periodic cycling of the drain and vent valves is the best method to clear the effects of particulate silting, thus promoting smooth opening and closing and more reliable valve operation. Also, in case of improper valve operation, cycling can indicate whether excessive pressure transients may be generated during and after a reactor scram which might damage the SDV piping system and cause a loss of system integrity or function.

2.2 LCO/SURVEILLANCE REQUIREMENTS FOR REACTOR PROTECTION SYSTEM SDV LIMIT SWITCHES

The paragraphs of the NRC staff's Model Technical Specifications pertinent to LCO/surveillance requirements for reactor protection system SDV limit switches are:

> "3.3.1 - As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE with the REACTOR PROTECTION SYSTEM RESPONSE TIME as shown in Table 3.3.1-2.

Table 3.3.1-1. Reactor Protection System Instrumentation

Functional Unit	Applicable Operational Conditions	Minimum Operable Channels Per Trip System (a)	Action
8. Scram Discharge Volume Water Level-High	1,2,5 (h)	2	4

Table 3.3.1-2. Reactor Protection System Response Times

Functional Unit	Response Time (Seconds)
8. Scram Discharge Volume Water Level-High	NA"

"4.3.1.1 - Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1.1-1.

Table 4.3.1.1-1. Reactor Protection System Instrumentation Surveillance Requirements

Functional Unit	Channel Check	Channel Functional Test	Channel Calibration	Operational Conditions in Which Surveillance Required
8. Scram Discharge Volume Water Level-High	NA	M	R	1,2,5

Notation (a) A channel may be placed in an inoperable status up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

(h) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

Action 4: In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS* and fully insert all insertable control rods within one hour.

*Except movement of IRM, SRM or special movable detectors, or replacement of LPRM strings provided SRM instrumentation is OPERABLE per Specification 3.9.2."

Paragraph 3.3.1 and Table 3.3.1-1 of the Model Technical Specifications require the functional unit of SDV water level-high to have at least 2 operable channels containing 2 limit switches per trip system, for a total of 4 operable channels containing 4 limit switches per 2 trip systems for the

reactor protection system which automatically initiates a scram. The technical objective of these requirements is to provide 1-out-of-2-taken-twice logic for the reactor protection system. The response time of the reactor protection system for the functional unit of SDV water level-high should be measured and kept available (it is not given in Table 3.3.1-2).

Paragraph 4.3.1.1 and Table 4.3.1.1-1 give reactor protection system instrumentation surveillance requirements for the functional unit of SDV water level-high. Each reactor protection system instrumentation channel containing a limit switch should be shown to be operable by the Channel Functional Test monthly and Channel Calibration at each refueling outage.

2.3 LCO/SURVEILLANCE REQUIREMENTS FOR CONTROL ROD WITHDRAWAL BLOCK SDV LIMIT SWITCHES

The NRC staff's Model Technical Specifications specify the following LCO/surveillance requirements for control rod withdrawal block SDV limit switches:

"3.3.6 - The control rod withdrawal block instrumentation channel shown in Table 3.3.6-1 shall be OPERABLE with trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2.

Table 3.3.6-1. Control Rod Withdrawal Block Instrumentation

Trip Function	Minimum Operable Channels Per Trip Function	Applicable Operational Conditions	Action
5. Scram Discharge Volume			
a. Water level-high	2	1, 2, 5**	62
b. Scram trip bypassed	1	(1, 2, 5**)	62

ACTION 62: With the number of OPERABLE channels less than required by the minimum OPERABLE channels per trip function requirement, place the inoperable channel in the tripped condition within one hour.

**With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

Table 3.3.6-2. Control Rod Withdrawal Block Instrumentation Setpoints

Trip Function	Trip Setpoint	Allowable Value
5. Scram Discharge Volume		
a. Water level-high	To be specified	NA
b. Scram trip bypassed	NA	NA"

"4.3.6 - Each of the above control rod withdrawal block trip systems and instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.6-1.

Table 4.3.6-1. Control Rod Withdrawal Block Instrumentation Surveillance Requirements

Trip Function	Channel Check	Channel Functional Test	Channel Calibration	Operational Conditions in Which Surveillance Required
5. Scram Discharge Volume				
a. Water Level-High	NA	Q	R	1, 2, 5**
b. Scram Trip Bypassed	NA	M	NA	(1, 2, 5**)

**With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2."

Paragraph 3.3.6 and Table 3.3.6-1 of the Model Technical Specifications require the control rod withdrawal block instrumentation to have at least 2 operable channels containing 2 limit switches for SDV water level-high and 1 operable channel containing 1 limit switch for SDV scram trip bypassed. The technical objective of these requirements is to have at least one channel containing one limit switch available to monitor the SDV water level when the other channel with a limit switch is being tested or undergoing maintenance. The trip setpoint for control rod withdrawal block instrumentation monitoring

SDV water level-high should be specified as indicated in Table 3.3.6-2. The trip function prevents further withdrawal of any control rod when the control rod block SDV limit switches indicate water level-high.

Paragraph 4.3.6 and Table 4.3.6-1 require that each control rod withdrawal block instrumentation channel containing a limit switch be shown to be operable by the Channel Functional Test once per 3 months for SDV water level-high, by the Channel Functional Test once per month for SDV scram trip bypassed, and by Channel Calibration at each refueling outage for SDV water level-high.

The Surveillance Criteria of the BWR Owners Subgroup given in Appendix A, "Long-Term Evaluation of Scram Discharge System," of "Generic Safety Evaluation Report BWR Scram Discharge System," written by the NRC staff and issued on December 1, 1980, are:

1. Vent and drain valves shall be periodically tested.
2. Verifying and level detection instrumentation shall be periodically tested in place.
3. The operability of the entire system as an integrated whole shall be demonstrated periodically and during each operating cycle, by demonstrating scram instrument response and valve function at pressure and temperature at approximately 50% control rod density.

Analysis of the above criteria indicates that the NRC staff's Model Technical Specifications requirements, the acceptance criteria for the present TER, fully cover the BWR Owners Subgroup Surveillance Criteria 1 and 2 and partially cover Criterion 3.

3. METHOD OF EVALUATION

The NU submittal for Millstone Unit No. 1 was evaluated in two stages, initial and final.

During the initial evaluation, only the NRC staff's Model Technical Specifications requirements were used to determine if:

- the Licensee's submittal was responsive to the July 7, 1980 NRC request for proposed Technical Specifications changes involving the surveillance requirements of the SDV vent and drain valves, LCO/surveillance requirements for reactor protection system SDV limit switches, and LCO/surveillance requirements for control rod block SDV limit switches
- the submitted information was sufficient to permit a detailed technical evaluation.

During the final evaluation, in addition to the NRC staff's Model Technical Specifications requirements, background material in References 1 through 10, pertinent sections of "Northeast Utilities Millstone Nuclear Power Station Unit No. 1 Safety Analysis Report," and Millstone Unit No. 1 Technical Specifications were studied to determine the technical bases for the design of SDV main components and instrumentation. Subsequently, the Licensee's response was compared directly to the requirements of the NRC staff's Model Technical Specifications. The findings of the final evaluation are presented in Section 4 of this report.

The initial evaluation concluded that the Licensee's submittal was responsive to the NRC request of July 7, 1980, but some detailed information was not available. A Request for Additional Information (RAI) was sent to NU by the NRC on September 9, 1981. The present TER is based on both the initial submittal and the Licensee's March 9, 1982 response [11] to the RAI (see Appendix C).

4. TECHNICAL EVALUATION

4.1 SURVEILLANCE REQUIREMENTS FOR SDV DRAIN AND VENT VALVES

NRC STAFF'S MODEL TECHNICAL SPECIFICATIONS

Paragraph 4.1.3.1.1 requires demonstrating that the SDV drain and vent valves are operable by:

a. verifying each valve to be open at least once per 31 days (valves may be closed intermittently for testing under administrative controls)

b. cycling each valve at least one complete cycle of full travel at least once per 92 days.

LICENSEE RESPONSE

The Licensee proposed to revise page 3/4 3-4 and to add page 3/4 3-4a to the Millstone Unit No. 1 Technical Specifications, providing the following information:

"2. The scram discharge volume drain and vent valves shall be verified open at least once per month.

3. The following conditions of operability of the scram discharge volume drain and vent valves shall be verified at least once per cycle in accordance with Section 3.13, Inservice Inspection:

a. Closing time after signal for control rods to scram and
b. Verification of opening when scram signal is reset and when the scram discharge volume trip is bypassed."

The Licensee's answer to the RAI regarding cycling the drain and vent valves at least one complete cycle of full travel at least once per 92 days (quarterly) was as follows (see Appendix C):

"We currently cycle the SDV vent and drain valves once every three months in accordance with the Millstone Unit No. 1 In-Service Inspection Program, which is already incorporated into our Technical Specifications per Section 4.13. This frequency is consistent with ASME Code requirements."

EVALUATION

The proposed revision of page 3/4 3-4, paragraph 2, of the Millstone Unit No. 1 Technical Specifications complies with the requirements of paragraph 4.1.3.1.1a of the NRC staff's Model Technical Specifications. The Licensee's In-Service Inspection Program, which is already incorporated into the Millstone Nuclear Power Station Unit No. 1 Technical Specifications per Section 4.13, provides for cycling the SDV drain and vent valves once every three months and meets the NRC staff's Model Technical Specifications requirements of paragraph 4.1.3.1.1b.

4.2 LCO/SURVEILLANCE REQUIREMENTS FOR REACTOR PROTECTION SYSTEM SDV LIMIT SWITCHES

NRC STAFF'S MODEL TECHNICAL SPECIFICATIONS

Paragraph 3.3.1 and Table 3.3.1-1 require the functional unit of SDV water level-high to have at least 2 operable channels containing 2 limit switches per trip system, for a total of 4 operable channels containing 4 limit switches per 2 trip systems for the reactor protection system which automatically initiates scram.

Paragraph 3.3.1 and Table 3.3.1-2 concern the response time of the reactor protection system for the functional unit of SDV water level-high which should be specified for each BWR (it is not specified in the table). Paragraph 4.3.1.1 and Table 4.3.1.1-1 require that each reactor protection system instrumentation channel containing a limit switch be shown to be operable for the functional unit of SDV water level-high by the Channel Functional Test monthly and by Channel Calibration at each refueling outage. The applicable operational conditions for these requirements are startup, run, and refuel.

LICENSEE RESPONSE

Page 3/4 1-2, Table 3.1.1, Reactor Protection System (Scram) Instrumentation Requirements, of the present Millstone Unit No. 1 Technical Specifications addresses paragraph 3.3.1 and Table 3.3.1-1 requirements of the NRC staff's

Model Technical Specifications, providing the following information for "Trip Function Scram Discharge Volume High Level":

"1. Minimum Number of Operable Instr. Channels per Trip (1) System: 2
2. Trip Level Setting: ≤ 39 gallons
3. Modes in Which Function Must be Operable: Refuel (8), Startup/Hot Standby, Run
4. Action*: A"

NOTES:

"1. There shall be two operable or tripped trip systems for each function.

8. When the reactor is subcritical and the reactor water temperature is less than 212°F, only the following trip functions need be operable:

a. Mode Switch in Shutdown
b. Manual Scram
c. High Flux IRM
d. Scram Discharge Volume High Level
e. APRM Reduced High Flux

*Action: If the first column cannot be met for one of the trip systems, that trip system shall be tripped. If the first column cannot be met for both trip systems, the appropriate actions listed below shall be taken:

a. Initiate insertion of operable rods and complete insertion of all operable rods within four hours."

Paragraphs 3.3.C and 4.3.C, page 3/4 3-4, of the current Millstone Unit No. 1 Technical Specifications address paragraph 3.3.1 and Table 3.3.1-2 of the NRC staff's Model Technical Specifications. They provide the required average scram insertion time and required scram time tests each operating cycle, respectively. From these scram time tests the response time of the reactor protection system for the functional unit of SDV water level-high can be deduced.

Page 3/4 1-5, Table 4.1.1, Scram Instrumentation Functional Tests Minimum Functional Test Frequencies for Safety Instrument and Control Circuits, of the current Millstone Unit No. 1 Technical Specifications provides the following information for "Instrument Channel High Water Level in Scram Discharge":

"1. Group (3): A
2. Functional Test: Trip Channel and Alarm
3. Minimum Frequency (4): (1)"

NOTES:

"1. Initially once per month until exposure hours (M as defined in Figure 4.1.1) is 2.0×10^5, thereafter according to Figure 4.1.1, with an interval not less than one month nor more than three months. Millstone will use data compiled by Commonwealth Edison on the Dresden 2 unit in addition to Millstone Unit 1 data.

3. A description of the three groups is included in the bases of this specification.

4. Functional tests are not required when the systems are not required to be operable or are tripped. If tests are missed, they shall be performed prior to returning the systems to an operable status."

Page 3/4 1-7 contains Table 4.1.2, Scram Instrumentation Calibration Minimum Calibration Frequencies for Reactor Protection Instrument Channel, with the following information for "Instrument Channel High Water Level in Scram Discharge":

"1. Group (1): A
2. Calibration Method: Water Level
3. Minimum Calibration Frequency (2): Every 3 months"

NOTES:

"1. A description of the three groups is included in the bases of this specification.
2. Calibration tests are not required when the systems are not required to be operable or are tripped. It tests are missed, they shall be performed prior to returning the systems to an operable status."

Tables 4.1.1 and 4.1.2 address the NRC staff's Model Technical Specifications requirements of paragraph 4.3.1.1 and Table 4.3.1.1-1.

EVALUATION

The current Millstone Unit No. 1 Technical Specifications fully satisfy the NRC staff's Model Technical Specifications requirements in regard to LCO/Surveillance Requirements for Reactor Protection System SDV Limit Switches. Thus, the original page 3/4 1-2 with Table 3.1.1 complies with the NRC requirements of paragraph 3.3.1 and Table 3.3.1-1. The Millstone Unit 1 reactor

protection system SDV water level-high instrumentation consists of 2 operable channels containing 2 limit switches per trip system, for a total of 4 operable channels containing 4 limit switches per 2 trip systems, making 1-out-of-2-taken-twice logic. The specified trip setting of $\leq$ 39 gallons for scram initiation and applicable operating conditions of refuel, startup/hot standby, and run are acceptable.

Paragraphs 3.3.C and 4.3.C, original page 3/4 3-4, do not explicitly specify reactor protection system SDV water level-high response time. Instead, they provide requirements for scram insertion time and scram time tests from which the required response time can be deduced. Such an approach is acceptable, since Table 3.3.1-2 of the NRC staff's Model Technical Specifications does not provide a numerical value for the response time.

Although the original provisions of the Millstone Technical Specifications in regard to reactor protection system SDV water level-high Channel Functional Test and Channel Calibration differ from the NRC staff's Model Technical Specifications, they are acceptable. The Millstone Technical Specifications provide for the Channel Functional Test "Initially once per month until exposure hours is 2.0×10^5, thereafter according to Figure 4.1.1, with an interval not less than one month nor more than three months" (required: monthly) and for Channel Calibration "Every 3 months" (required: each refueling). Since Channel Calibration will be performed at least four times as often as required, performance of the Channel Functional Test at intervals not less than 1 month nor more than 3 months is acceptable.

4.3 LCO/SURVEILLANCE REQUIREMENTS FOR CONTROL ROD WITHDRAWAL BLOCK SDV LIMIT SWITCHES

NRC STAFF'S MODEL TECHNICAL SPECIFICATIONS

Paragraph 3.3.6 and Table 3.3.6-1 require the control rod withdrawal block instrumentation to have at least 2 operable channels containing 2 limit switches for SDV water level-high, and 1 operable channel containing 1 limit switch for SDV trip bypassed. Paragraph 3.3.6 also requires specifying the

trip setpoint for control rod withdrawal block instrumentation monitoring SDV water level-high as indicated in Table 3.3.6-2.

Paragraph 4.3.6 and Table 4.3.6-1 require each control rod withdrawal block instrumentation channel containing a limit switch to be shown to be operable by the Channel Functional Test once per 3 months for SDV water level-high, by the Channel Functional Test once per month for SDV scram trip bypassed, and by Channel Calibration at each refueling outage for SDV water level-high.

LICENSEE RESPONSE

The Licensee proposed to revise pages 3/4 2-5 and 3/4 2-6 and to add page 3/4 2-5a to the Millstone Unit No. 1 Technical Specifications. The revised page 3/4 2-5 and added page 3/4 2-5a contain Table 3.2.3 with pertinent information given below.

"Table 3.2.3. Instrumentation That Initiates Rod Block

Minimum Number of Operable Instrument Channels per Trip System (1)	Instrument	Trip Level Setting
1	Scram Discharge Volume - Water Level High	$\leq$ 18 gallons
1	Scram Discharge Volume - Scram Trip Bypassed	N/A

NOTES:

(1) For the Startup/Hot Standby and Run positions of the Reactor Selector Switch, there shall be two operable or tripped trip systems for each function except the SRM rod blocks; IRM downscale are not operable in the Run position and APRM downscale need not be operable in the Startup/Hot Standby mode. If the first column cannot be met for one of the two trip systems, this condition may exist for up to seven days provided that during that time the operable system is functionally tested immediately and daily thereafter; if this condition lasts longer than seven days, the system shall be tripped. If the first column cannot be met for both trip systems, the systems shall be tripped."

The revised page 3/4 2-6 contains Table 4.2.1 with the following relevant information.

"Table 4.2.1. Minimum Test and Calibration Frequency for Core Cooling Instrumentation Rod Blocks and Isolations

Instrument Channel	Instrument Functional Test (2)	Calibration (2)	Instrument Check (2)
Rod Blocks			
9. Scram Discharge Volume-Water Level High	Refueling Outage	Refueling Outage	--
10. Scram Discharge Volume-Scram Trip Bypassed	Refueling Outage	None	--

NOTES:

(2) Functional test calibrations and instrument checks are not required when these instruments are not required to be operable or are tripped."

The revised page 3/4 2-5 with Table 3.2.3 addresses the NRC staff's Model Technical Specifications requirements of paragraph 3.3.6 with Table 3.3.6-1 and Table 3.3.6-2. The NRC requirements of paragraph 4.3.6 with Table 4.3.6-1 are addressed by the revised page 3/4 2-6 with Table 4.2.1.

The Licensee agreed to revise the frequency of "Instrument Functional Test" in Table 4.2.1 from "Refueling Outage" to "Quarterly."

In addition, the Licensee is installing a second instrument volume with six level switches that have the same functions as in the existing instrument volume.

EVALUATION

The existing Millstone Unit No. 1 scram discharge system has six level switches on the scram discharge volume (see FSAR, Figure X-2.5. CRD Hydraulic System-Piping Diagram) set at three different water levels to guard against

operation of the reactor without sufficient free volume present in the scram discharge headers to receive the scram discharge water in the event of a scram. At the first (lowest) level, one level switch initiates an alarm for operator action. At the second level, with the setpoint of $\leq$ 18 gallons (see revised page 3/4 2-5, Table 3.2.3), one level switch initiates a rod withdrawal block to prevent further withdrawal of any control rod. At the third (highest) level, with the setpoint of $\leq$ 39 gallons (see page 3/4 1-2, Table 3.1.1 of the Millstone Technical Specifications), the four level switches (two for each reactor protection system trip system) initiate a scram to shut down the reactor while sufficient free volume is available to receive the scram discharge water. The installation of a second instrument volume acting in parallel with the existing instrument volume significantly increases the reliability of the SDV system and provides two operable channels containing two limit switches for control rod withdrawal block instrumentation SDV water level-high and one operable channel containing one limit switch for SDV trip bypassed, complying with the NRC staff's Model Technical Specifications requirements of paragraph 3.3.6 and Table 3.3.6-1.

The specified trip level setting of $\leq$ 18 gallons, in Table 3.2.3 of the revised page 3/4 2-5, is acceptable for control rod withdrawal block instrumentation monitoring SDV water level-high. It meets the NRC staff's Model Technical Specifications requirements of paragraph 3.3.6 and Table 3.3.6-2.

The Licensee's agreement to revise the frequency of the "Instrument Functional Test" for control rod withdrawal block, in Table 4.2.1 on revised page 3/4 2-6, from "Refueling Outage" to Quarterly," and to provide calibration each refueling outage for SDV water level-high meets the NRC staff's Model Technical Specifications requirements of paragraph 4.3.6 and Table 4.3.6-1. It does not meet the requirements for control rod withdrawal block SDV scram trip bypassed "Instrument Functional Test," which should be performed monthly, not quarterly. However, the Licensee is installing a second instrument volume containing six additional limit switches. This significantly increases the reliability of the SDV system and provides a technical basis for acceptance of quarterly performance of the "Instrument Functional Test" for control rod withdrawal block SDV scram trip bypassed.

5. CONCLUSIONS

Table 5-1 summarizes the results of the final review and evaluation of the Millstone Nuclear Power Station Unit No. 1 proposed Phase 1 Technical Specifications changes for SDV long-term modification in regard to surveillance requirements for SDV vent and drain valves and LCO/surveillance requirements for reactor protection system and control rod block SDV limit switches. The following conclusions were made:

- The proposed revision of page 3/4 3-4, paragraph 2 of the Millstone Nuclear Power Station Unit No. 1 Technical Specifications complies with the NRC staff's Model Technical Specifications requirements of paragraph 4.1.3.1.1a.

- The Licensee's In-Service Inspection Program, which is already incorporated into the Millstone Nuclear Power Station Unit No. 1 Technical Specifications per Section 4.13, provides for cycling the SDV drain and vent valves once every three months, and meets the NRC staff's Model Technical Specifications requirements of paragraph 4.1.3.1.1b.

- The current Millstone Unit No. 1 Technical Specifications fully satisfy the NRC staff's Model Technical Specifications requirements in regard to LCO/surveillance requirements for reactor protection system SDV limit switches (see pages 3/4 1-2, Table 3.1.1; 3/4 3-4, paragraphs 3.3.C and 4.3.C; 3/4 1-5, Table 4.1.1; and 3/4 1-7, Table 4.1.2).

- The specified trip level setting of $\leq$ 18 gallons, in Table 3.2.3 of revised page 3/4 2-5, is acceptable for control rod withdrawal block instrumentation monitoring SDV water level-high. It meets the NRC staff's Model Technical Specifications requirements of paragraph 3.3.6 and Table 3.3.6-2.

- The Licensee's agreement to revise the frequency of "Instrument Functional Test" for control rod withdrawal block, in Table 4.2.1 on revised page 3/4 2-6, from "Refueling Outage" to "Quarterly," and to provide calibration each refueling outage for SDV water level-high meets the NRC staff's Model Technical Specifications requirements of paragraph 4.3.6 and Table 4.3.6-1. It does not meet the requirements for control rod withdrawal block SDV scram trip bypassed "Instrument Functional Test," which should be performed monthly, not quarterly. However, the Licensee is installing a second instrument volume containing six additional limit switches. This significantly increases the reliability of the SDV system and provides a technical basis for acceptance of quarterly performance of the "Instrument Functional Test" for control rod withdrawal block SDV scram trip bypassed.

Table 5-1. Evaluation of Phase 1 Proposed Technical Specifications Changes for Scram Discharge Volume Long-Term Modifications Millstone Nuclear Power Station Unit No. 1

Surveillance Requirements	Technical Specifications: NRC Staff Model (Paragraph)	Technical Specifications: Proposed by Licensee	Evaluation
SDV DRAIN AND VENT VALVES			
Verify each valve open	Once per 31 days (4.1.3.1.1a)	Once per month (p. 3/4 3-4, revised, p. 3/4 3-4a, added)	Acceptable
Cycle each valve one complete cycle	Once per 92 days (4.1.3.1.1b)	Every 3 months (Section 4.13)	Acceptable
REACTOR PROTECTION SYSTEM SDV LIMIT SWITCHES			
Minimum operable channels per trip system	2 (3.3.1, Table 3.3.1-1)	2 (p. 3/4 1-2, Table 3.1.1)	Acceptable
SDV water level-high response time	NA (3.3.1, Table 3.3.1-2)	NA (p. 3/4 3-4, paragraphs 3.3.C and 4.3.C)	Acceptable
SDV water level-high			
Channel functional test	Monthly (4.3.1.1, Table 4.3.1.1-1)	Initially once per month 1 month $\leq$ interval $<$ 3 months (p. 3/4 1-5, Table 4.1.1)	Acceptable
Channel calibration	Each refueling (4.3.1.1, Table 4.3.1.1-1)	Every 3 months (p. 3/4 1-7, Table 4.1.2)	Acceptable

Table 5-1 (Cont.)

Surveillance Requirements	Technical Specifications: NRC Staff Model (Paragraph)	Technical Specifications: Proposed by Licensee	Evaluation
CONTROL ROD BLOCK SDV LIMIT SWITCHES			
Minimum operable channels per trip function			
SDV water level-high	2 (3.3.6, Table 3.3.6-1)	2 (p. 3/4 2-5, revised, p. 3/4 2-5a, added)	Acceptable*
SDV scram trip bypassed	1 (3.3.6, Table 3.3.6-1)	1 (p. 3/4 2-5, revised, p. 3/4 2-5a, added)	Acceptable*
SDV water level-high			
Trip setpoint	NA (3.3.6, Table 3.3.6-2)	$\leq$ 18 gallons (p. 3/4 2-5, revised, p. 3/4 2-5a, added)	Acceptable
Channel functional test	Quarterly (4.3.6, Table 4.3.6-1)	Quarterly (p. 3/4 2-6, second revision)	Acceptable
Channel calibration	Each refueling (4.3.6, Table 4.3.6-1)	Refueling Outage (p. 3/4 2-6, revised)	Acceptable
SDV scram trip bypassed			
Channel functional test	Monthly (4.3.6, Table 4.3.6-1)	Quarterly (p. 3/4 2-6, second revision*)	Acceptable*

* The Licensee is installing a second instrument volume with six level switches.

6. REFERENCES

1. IE Bulletin 80-14, "Degradation of BWR Scram Discharge Volume Capability"
   NRC, Office of Inspection and Enforcement, June 12, 1980

2. D. G. Eisenhut (NRR), letter "To All Operating Boiling Water Reactors (BWRs)" with enclosure, "Model Technical Specifications"
   July 7, 1980

3. IE Bulletin 80-17, "Failure of 76 of 185 Control Rods to Fully Insert During a Scram at a BWR"
   NRC, Office of Inspection and Enforcement, July 3, 1980

4. IE Bulletin 80-17, Supplement 1, "Failure of 76 of 185 Control Rods to Fully Insert During a Scram at a BWR"
   NRC, Office of Inspection and Enforcement, July 18, 1980

5. IE Bulletin 80-17, Supplement 2, "Failures Revealed by Testing Subsequent to Failure of Control Rods to Insert During a Scram at a BWR"
   NRC, Office of Inspection and Enforcement, July 22, 1980

6. IE Bulletin 80-17, Supplement 3, "Failure of Control Rods to Insert During a Scram at a BWR"
   NRC, Office of Inspection and Enforcement, August 22, 1980

7. IE Bulletin 80-17, Supplement 4, "Failure of Control Rods to Insert During a Scram at a BWR"
   NRC, Office of Inspection and Enforcement, December 18, 1980

8. IE Bulletin 80-17, Supplement 5, "Failure of Control Rods to Insert During a Scram at a BWR"
   NRC, Office of Inspection and Enforcement, February 13, 1981

9. P. S. Check (NRR), memorandum with enclosure, "Generic Safety Evaluation Report BWR Scram Discharge System"
   December 1, 1980

10. P. S. Check (NRR), memorandum with enclosure, "Staff Report and Evaluation of Supplement 4 to IE Bulletin 80-17"
    June 10, 1981

11. W. G. Counsil (NU)
    Letter to D. M. Crutchfield (NRC)
    Subject: Response to RAI on Scram Discharge Volume
    March 9, 1982

APPENDIX A

NRC STAFF'S MODEL TECHNICAL SPECIFICATIONS*

*Note: Applicable changes are marked by vertical lines in the margins.

REACTIVITY CONTROL SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

2. If the inoperable control rod(s) is inserted, within one hour disarm the associated directional control valves either:

   a) Electrically, or

   b) Hydraulically by closing the drive water and exhaust water isolation valves.

3. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

c. With more than 8 control rods inoperable, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The scram discharge volume drain and vent valves shall be demonstrated OPERABLE by:

a. Verifying each valve to be open* at least once per 31 days and

b. Cycling each valve through at least one complete cycle of full travel at least once per 92 days.

4.1.3.1.2 When above the preset power level of the RWM and RSCS, all withdrawn control rods not required to have their directional control valves disarmed electrically or hydraulically shall be demonstrated OPERABLE by moving each control rod at least one notch:

a. At least once per 7 days, and

b. At least once per 24 hours when any control rod is immovable as a result of excessive friction or mechanical interference.

4.1.3.1.3 All control rods shall be demonstrated OPERABLE by performance of Surveillance Requirements 4.1.3.2, 4.1.3.4, 4.1.3.5, 4.1.3.6 and 4.1.3.7.

*These valves may be closed intermittently for testing under administrative controls.

REACTIVITY CONTROL SYSTEMS

CONTROL ROD MAXIMUM SCRAM INSERTION TIMES

LIMITING CONDITION FOR OPERATION

3.1.3.2 The maximum scram insertion time of each control rod from the fully withdrawn position to notch position (6), based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed (7.0) seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

With the maximum scram insertion time of one or more control rods exceeding (7.0) seconds:

a. Declare the control rod(s) with the slow insertion time inoperable, and

b. Perform the Surveillance Requirements of Specification 4.1.3.2.c at least once per 60 days when operation is continued with three or more control rods with maximum scram insertion times in excess of (7.0) seconds, or

c. Be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.2 The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than or equal to 950 psig and, during single control rod scram time tests, the control rod drive pumps isolated from the accumulators:

a. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER following CORE ALTERATIONS or after a reactor shutdown that is greater than 120 days,

b. For specifically affected individual control rods following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods, and

c. For 10% of the control rods, on a rotating basis, at least once per 120 days of operation.

3/4.3 INSTRUMENTATION

3/4.3.1 - REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE with the REACTOR PROTECTION SYSTEM RESPONSE TIME as shown in Table 3.3.1-2.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place at least one inoperable channel in the tripped condition within one hour.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one inoperable channel in at least one trip system* in the tripped condition within one hour and take the ACTION required by Table 3.3.1-1.

c. The provisions of Specification 3.0.3 are not applicable in OPERATIONAL CONDITION 5.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1.1-1.

4.3.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip function shown in Table 3.3.1-2 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one logic train such that both logic trains are tested at least once per 36 months and one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function.

* If both channels are inoperable in one trip system, select at least one inoperable channel in that trip system to place in the tripped condition, except when this would cause the Trip Function to occur.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
8. Scram Discharge Volume Water Level - High	1, 2, 5(h)	2	4
9. Turbine Stop Valve - Closure	1(i)	4(j)	7
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	1(i)	2(j)	7
11. Reactor Mode Switch in Shutdown Position	1, 2, 3, 4, 5	1	8
12. Manual Scram	1, 2, 3, 4, 5	1	9

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION

ACTION 1 - In OPERATIONAL CONDITION 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS* and fully insert all insertable control rods within one hour.

ACTION 2 - Lock the reactor mode switch in the Shutdown position within one hour.

ACTION 3 - Be in at least STARTUP within 2 hours.

ACTION 4 - In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS* and fully insert all insertable control rods within one hour.

ACTION 5 - Be in at least HOT SHUTDOWN within 6 hours.

ACTION 6 - Be in STARTUP with the main steam line isolation valves closed within 2 hours or in at least HOT SHUTDOWN within 6 hours.

ACTION 7 - Initiate a reduction in THERMAL POWER within 15 minutes and reduce turbine first stage pressure to ≤ (250) psig, equivalent to THERMAL POWER less than (30)% of RATED THERMAL POWER, within 2 hours.

ACTION 8 - In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 3 or 4, verify all insertable control rods to be fully inserted within one hour.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS* and fully insert all insertable control rods within one hour.

ACTION 9 In OPERATIONAL CONDITION 1 or 2, be in at least HOT SHUTDOWN within 6 hours.

In OPERATIONAL CONDITION 3 or 4, lock the reactor mode switch in the Shutdown position within one hour.

In OPERATIONAL CONDITION 5, suspend all operations involving CORE ALTERATIONS* and fully insert all insertable control rods within one hour.

*Except movement of IRM, SRM or special movable detectors, or replacement of LPRM strings provided SRM instrumentation is OPERABLE per Specification 3.9.2.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

b) The "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn* and shutdown margin demonstrations performed per Specification 3.10.3.

(c) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than (11) LPRM inputs to an APRM channel.

(d) These functions are not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed per Specification 3.10.1.

(e) This function shall be automatically bypassed when the reactor mode switch is not in the Run position.

(f) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(g) Also actuates the standby gas treatment system.

(h) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(i) These functions are automatically bypassed when turbine first stage pressure is ≤ (250) psig, equivalent to THERMAL POWER less than (30)% of RATED THERMAL POWER.

(j) Also actuates the EDC-RPT system.

*Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - Upscale	NA
b. Inoperative	NA
2. Average Power Range Monitor[A]:	
a. Neutron Flux - Upscale, (15)%	NA
b. Flow Biased Simulated Thermal Power - Upscale	≤ (0.09)[AA]
c. Fixed Neutron Flux - Upscale, (118)%	≤ (0.09)
d. Inoperative	NA
e. LPRM	NA
3. Reactor Vessel Steam Dome Pressure - High	≤ (0.55)
4. Reactor Vessel Water Level - Low, Level 3	≤ (1.05)
5. Main Steam Line Isolation Valve - Closure	≤ (0.06)
6. Main Steam Line Radiation - High	NA
7. Primary Containment Pressure - High	NA
8. Scram Discharge Volume Water Level - High	NA
9. Turbine Stop Valve - Closure	≤ (0.06)
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≤ (0.08)#
11. Reactor Mode Switch in Shutdown Position	NA
12. Manual Scram	NA

[A]Neutron detectors are exempt from response time testing. Response time shall be measured from the detector output or from the input of the first electronic component in the channel. (This provision is not applicable to Construction Permits docketed after January 1, 1970. See Regulatory Guide 1.10, November 1977.)

[AA]Not including simulated thermal power time constant.

#Measured from start of turbine control valve fast closure.

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
8. Scram Discharge Volume Water Level - High	NA	M	R	1, 2, 5
9. Turbine Stop Valve - Closure	NA	M	R	1
10. Turbine Control Valve Fast Closure Trip Oil Pressure - Low	NA	M	Q	1
11. Reactor Mode Switch in Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
12. Manual Scram	NA	M	NA	1, 2, 3, 4, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
(b) Within 24 hours prior to startup, if not performed within the previous 7 days.
(c) The IRM and SRM channels shall be determined to overlap for at least ( ) decades during each startup and the IRM and APRM channels shall be determined to overlap for at least ( ) decades during each controlled shutdown, if not performed within the previous 7 days.
(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER ≥ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference greater than 2%. Any APRM channel gain adjustment made in compliance with Specification 3.2.2 shall not be included in determining the absolute difference.
(e) This calibration shall consist of the adjustment of the APRM readout to conform to a calibrated flow signal.
(f) The LPRMs shall be calibrated at least once per 1000 effective full power hours (EFPH) using the TIP system.

INSTRUMENTATION

3/4.3.6 CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6. The control rod withdrawal block instrumentation channels shown in Table 3.3.6-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2.

APPLICABILITY: As shown in Table 3.3.6-1.

ACTION:

a. With a control rod withdrawal block instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.6-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, take the ACTION required by Table 3.3.6-1.

c. The provisions of Specification 3.0.3 are not applicable in OPERATIONAL CONDITION 5.

SURVEILLANCE REQUIREMENTS

4.3.6 Each of the above required control rod withdrawal block trip systems and instrumentation channels shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.6-1.

TABLE 3.3.6-1

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1. ROD BLOCK MONITOR(a)			
a. Upscale	2	1^	60
b. Inoperative	2	1^	60
c. Downscale	2	1^	60
2. APRM			
a. Flow Biased Simulated Thermal Power - Upscale	4	1	61
b. Inoperative	4	1, 2, 5	61
c. Downscale	4	1	61
d. Neutron Flux - Upscale, Startup	4	2, 5	61
3. SOURCE RANGE MONITORS			
a. Detector not full in(b)	3	2	61
	2	5	61
b. Upscale(c)	3	2	61
	2	5	61
c. Inoperative(c)	3	2	61
	2	5	61
d. Downscale(d)	3	2	61
	2	5	61
4. INTERMEDIATE RANGE MONITORS			
a. Detector not full in (e)	6	2, 5	61
b. Upscale	6	2, 5	61
c. Inoperative(e)	6	2, 5	61
d. Downscale	6	2, 5	61
5. SCRAM DISCHARGE VOLUME			
a. Water Level-High	2	1, 2, 5^^	62
b. Scram Trip Bypassed	1	(1, 2, 5^^)	62
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW			
a. Upscale	2	1	62
b. Inoperative	2	1	62
c. (Comparator) (Downscale)	2	1	62

TABLE 3.3.6-1 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION

ACTION

ACTION 60 - Take the ACTION required by Specification 3.1.4.3.

ACTION 61 - With the number of OPERABLE Channels:

a. One less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 7 days or place the inoperable channel in the tripped condition within the next hour.

b. Two or more less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour.

ACTION 62 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within one hour.

NOTES

* With THERMAL POWER ≥ (20)% of RATED THERMAL POWER.

** With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

a. The RBM shall be automatically bypassed when a peripheral control rod is selected.

b. This function shall be automatically bypassed if detector count rate is > 100 cps or the IRM channels are on range (2) or higher.

c. This function shall be automatically bypassed when the associated IRM channels are on range 8 or higher.

d. This function shall be automatically bypassed when the IRM channels are on range 3 or higher.

e. This function shall be automatically bypassed when the IRM channels are on range 1.

TABLE 3.3.6-2

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale	$\leq$ 0.66 W + (40)%	$\leq$ 0.66 W + (43)%
b. Inoperative	NA	NA
c. Downscale	$\geq$ (5)% of RATED THERMAL POWER	$\geq$ (3)% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Simulated Thermal Power - Upscale	$\leq$ 0.66 W + (42)%[A]	$\leq$ 0.66 W + (45)%[A]
b. Inoperative	NA	NA
c. Downscale	$\geq$ (5)% of RATED THERMAL POWER	$\geq$ (3)% of RATED THERMAL POWER
d. Neutron Flux - Upscale Startup	$\leq$ (12)% of RATED THERMAL POWER	$\leq$ (14)% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	$\leq (2 \times 10^5)$ cps	$\leq (5 \times 10^5)$ cps
c. Inoperative	NA	NA
d. Downscale	$\geq$ (3) cps	$\geq$ (2) cps
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	$\leq$ (100/125) of full scale	$\leq$ (110/125) of full scale
c. Inoperative	NA	NA
d. Downscale	$\geq$ (5/125) of full scale	$\geq$ (3/125) of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level High	To be specified	NA
b. Scram Trip Bypassed	NA	NA
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	$\leq$ (___/___) of full scale	$\leq$ (___/___) of full scale
b. Inoperative	NA	NA
c. (Comparator) (Downscale)	$\leq$ (10)% flow deviation	$\leq$ (__)% flow deviation

[A]The Average Power Range Monitor rod block function is varied as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with Specification 3.2.2.

TABLE 4.3.6-1

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. ROD BLOCK MONITOR				
a. Upscale	NA	S/U[b],M	Q	1^
b. Inoperative	NA	S/U[b],M	NA	1^
c. Downscale	NA	S/U[b],M	Q	1^
2. APRM				
a. Flow Biased Simulated Thermal Power - Upscale	NA	S/U[b],M	Q	1
b. Inoperative	NA	S/U[b],M	NA	1, 2, 5
c. Downscale	NA	S/U[b],M	Q	1
d. Neutron Flux - Upscale, Startup	NA	S/U[b],M	Q	2, 5
3. SOURCE RANGE MONITORS				
a. Detector not full in	NA	S/U[b],W[c]	NA	2, 5
b. Upscale	NA	S/U[b],W[c]	Q	2, 5
c. Inoperative	NA	S/U[b],W[c]	NA	2, 5
d. Downscale	NA	S/U[b],W[c]	Q	2, 5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	NA	S/U[b],W[c]	NA	2, 5
b. Upscale	NA	S/U[b],W[c]	Q	2, 5
c. Inoperative	NA	S/U[b],W[c]	NA	2, 5
d. Downscale	NA	S/U[b],W[c]	Q	2, 5
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High	NA	Q	R	1, 2, 5^^
b. Scram Trip Bypassed	NA	M	NA	(1, 2, 5^^)
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW				
a. Upscale	NA	S/U[b],M	Q	1
b. Inoperative	NA	S/U[b],M	NA	1
c. (Comparator) (Downscale)	NA	S/U[b],M	Q	1

TABLE 4.3.6-1 (Continued)

CONTROL ROD WITHDRAWAL BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

NOTES:

a. Neutron detectors may be excluded from CHANNEL CALIBRATION.

b. Within 24 hours prior to startup, if not performed within the previous 7 days.

c. When making an unscheduled change from OPERATIONAL CONDITION 1 to OPERATIONAL CONDITION 2, perform the required surveillance within 12 hours after entering OPERATIONAL CONDITION 2.

* With THERMAL POWER $\geq$ (20)% of RATED THERMAL POWER.

** With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

GE-STS 3/4 3-35

APPENDIX B

NORTHEAST UTILITIES LETTER OF OCTOBER 16, 1980

AND

SUBMITTAL WITH PROPOSED TECHNICAL SPECIFICATIONS CHANGES

FOR

MILLSTONE NUCLEAR POWER STATION UNIT NO. 1

October 16, 1980

Docket No. 50-245
A01127

Director of Nuclear Reactor Regulation
Attn: Mr. Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Reference: (1) D. G. Eisenhut letter to All Operating Boiling Water Reactors dated July 7, 1980.

Gentlemen:

Millstone Nuclear Power Station, Unit No. 1
Proposed Technical Specification Changes
for Scram Discharge Volume System

Pursuant to 10CFR50.90, Northeast Nuclear Energy Company (NNECO) hereby proposes to amend its operating license for Millstone Unit No. 1 (DPR-21) by incorporating the attached changes into the Technical Specifications. These proposed changes are in response to the NRC's request as stated in Reference (1).

The enclosed "Description of Changes and Safety Evaluation Summary" should provide sufficient information for the NRC Staff to review and approve the recommended changes to the Technical Specifications. The off-site Nuclear Review Board has reviewed and approved these proposed changes and has determined that these changes do not constitute any unreviewed safety questions pursuant to 10CFR50.59.

The appropriate fee for a Class III amendment of $4,000 (four thousand dollars) in accordance with 10CFR170, is enclosed with this transmittal.

NNECO respectfully requests that these changes be reviewed and approved as soon as practicable. If you have any questions, please feel free to contact me.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

W. G. Counsil
Senior Vice President

Attachment

DOCKET NO. 50-245

ATTACHMENT

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 1

PROPOSED TECHNICAL SPECIFICATION CHANGES

FOR SCRAM DISCHARGE VOLUME SYSTEM

OCTOBER, 1980

DESCRIPTION OF CHANGES AND SAFETY EVALUATION SUMMARY

The proposed changes include adding scram discharge volume (SDV) rod block functions and surveillance requirements for SDV vent and drain valves to the Technical Specifications. The two (2) conditions for initiation of control rod block that have been added to the Technical Specifications are for high water level in the SDV system and when the SDV scram trip is bypassed. The inclusion of the above SDV rod block functions in the Technical Specifications provides additional limiting conditions of operation. The proposed surveillance requirements will increase the reliability of the SDV system.

Pursuant to 10CFR50.59, these changes have been reviewed and it has been determined that they do not involve any unreviewed safety questions in that they do not increase the probability of occurrence or the consequences of an accident or malfunction of equipment, create a possibility for a different type of accident or malfunction, or reduce the margin of safety as defined in the Technical Specifications.

TABLE 3.2.3

INSTRUMENTATION THAT INITIATES ROD BLOCK

Minimum Number of Operable Instrument Channels per Trip System(1)	Instrument	Trip Level Setting
2	APRM Upscale (Flow Biased)	See Specification 2.1.2B
2	APRM Downscale	≥ 3/125 Full Scale
1 (6)	Rod Block Monitor Upscale (Flow Biased)	≤ .65 w + 42 (2)
1 (6)	Rod Block Monitor Downscale	≥ 3/125 Full Scale
3	IRM Downscale (3)	≥ 3/125 Full Scale
3	IRM Upscale	≤ 108/125 Full Scale
2	SRM Detector not in Startup Position	(4)
2 (5)	SRM Upscale	≤ 10^5 counts/sec.
1	Scram Discharge Volume - Water Level High	≤ 18 gallons
1	Scram Discharge Volume - Scram Trip Bypassed	N/A

(1) For the Startup/Hot Standby and Run positions of the Reactor Mode Selector Switch, there shall be two operable or tripped trip systems for each function except the SRM rod blocks; IRM downscale are not operable in the RUN position and APRM downscale need not be operable in the Startup/Hot Standby mode. If the first column cannot be met for one of the two trip systems, this condition may exist for up to seven days provided that during that time the operable system is functionally tested immediately and daily thereafter; if this condition lasts longer than seven days, the system shall be tripped. If the first column cannot be met for both trip systems, the systems shall be tripped.

(2) W is the total core flow in percent of design (69 x 10^6 #/hr.). Trip level setting is in percent of full power.

(3) IRM downscale may be bypassed when it is on its lowest range.

(4) This function may be bypassed when the count rate is ≥ 100 cps or when all IRM range switches are above Position 2.

(5) One of these trips may be bypassed. The SRM function may be bypassed in the higher IRM ranges when the IRM upscale rod block is operable.

Table 3.2.3 Continued
Instrumentation That Initiates Rod Block

(6) The trip may be bypassed when the reactor power is $\leq$ 30% of rated. An RBM channel will be considered inoperable if there are less than half the total number of normal inputs from any LPRM level.

TABLE 4.2.1

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CORE COOLING INSTRUMENTATION ROD BLOCKS AND ISOLATIONS

Instrument Channel	Instrument Functional Test(2)	Calibration(2)	Instrument Check(2)
ECCS Instrumentation			
1. Reactor Low-Low Water Level	(1)	Once/3 Months	--
2. Drywell High Pressure	(1)	Once/3 Months	--
3. Reactor Low Pressure (Pump Start)	(1)	Once/3 Months	--
4. Reactor Low Pressure (Valve Permissive)	(1)	Once/3 Months	--
5. APR LP Core Cooling Pump Interlock	(1)	Once/3 Months	--
6. Containment Spray Interlock	(1)	Once/3 Months	--
7. Loss of Normal Power Relays	Refueling Outage	None	--
8. Power Available Relays	(1) (5)	None	--
9. Reactor High Pressure		Once/3 Months	--
Rod Blocks			
1. APRM Downscale	(1) (3)	Once/3 Months	(1)
2. APRM Flow Variable	(1) (3)	Once/3 Months	(1)
3. IRM Upscale	(6)	(6)	(6)
4. IRM Downscale	(6)	(6)	(6)
5. RBM Upscale	(1) (3)	Once/3 Months	(1)
6. RBM Downscale	(1) (3)	Once/3 Months	(1)
7. SRM Upscale	(6)	(6)	(6)
8. SRM Detector not in Startup Position	(6)	(6)	(6)
9. Scram Discharge Volume - Water Level High	Refueling Outage	Refueling Outage	--
10. Scram Discharge Volume - Scram Trip Bypassed	Refueling Outage	None	--
Main Steam Line Isolation			
1. Steam Tunnel High Temperature	Refueling Outage	Refueling Outage	--
2. Steam Line High Flow	(1)	Once/3 Months	Once/Day
3. Steam Line Low Pressure	(1) (3)	Refueling Outage	None
4. Steam Line High Radiation	(1) (3)	Once/3 Months(4)	Once/Day

Amendment No. 44, 47

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

5. During operation with limiting control rod patterns, as determined by the reactor engineer, either:

   a. Both RBM channels shall be operable; or

   b. Control rod withdrawal shall be blocked; or

   c. The operating power level shall be limited so that the MCPR will remain above 1.06 assuming a single error that results in complete withdrawal of any single operable control rod.

C. Scram Insertion Times

1. The average scram insertion time, based on the deenergization of the scram pilot valve solenoids as time zero, of all operable control rods in the reactor power operation condition shall be no greater than:

% Inserted From Fully Withdrawn	Average Scram Insertion Times (Sec.)
5	0.375
20	0.900
50	2.000
90	3.500

4. Prior to control rod withdrawal for startup or during refueling, verify that at least two source range channels have an observed count rate of at least three counts per second.

5. When a limiting control rod pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s) and daily thereafter.

C. Scram Insertion Times

1. During each operating cycle, each operable control rod shall be subjected to scram time tests from the fully withdrawn position. If testing is not accomplished during reactor power operation, the measured scram insertion times shall be extrapolated to the reactor power operation condition utilizing previously determined correlations.

2. The scram discharge volume drain and vent valves shall be verified open at least once per month.

3. The following conditions of operability of the scram discharge volume drain and vent valves shall be verified at least once per operating cycle in accordance with Section 3.13, Inservice Inspection:

LIMITING CONDITION FOR OPERATION		SURVEILLANCE REQUIREMENTS
		a. Closing time after signal for control rods to scram and b. Verification of opening when scram signal is reset and when the scram discharge volume trip is bypassed.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
2. The average of the scram insertion times for the three fastest control rods of all groups of four control rods in a two by two array shall be no greater than: % Inserted From Fully Withdrawn / Average Scram Insertion Times (sec.) 5 / 0.398 20 / 0.954 50 / 2.120 90 / 3.800	
3. a. The maximum scram insertion time for 90% insertion of any operable control rod shall not exceed 7.00 seconds. b. The scram discharge volume drain and vent valves will close in less than 30 seconds after receipt of a signal for control rods to scram.	
D. Control Rod Accumulators At all reactor operating pressures, a rod accumulator may be inoperable provided that no other control rod in the nine-rod square array around this rod has a: 1. Inoperable accumulator. 2. Directional control valve electrically disarmed while in a non-fully inserted position. 3. Scram insertion greater than maximum permission insertion time.	D. Control Rod Accumulators Once a shift, check the status in the control room of the pressure and level alarms for each accumulator.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
If a control rod with an inoperable accumulator is inserted "full-in" and its directional control valves are electrically disarmed, it shall not be considered to have an inoperable accumulator.	

APPENDIX C

NORTHEAST UTILITIES LETTER OF MARCH 9, 1982

WITH

RESPONSE TO RAI

FOR

MILLSTONE NUCLEAR POWER STATION UNIT NO. 1

[Note: Best copy available.]

ED MACHA:
FOR YOUR USE;
KEN ECCLESTON

27231

TO: J.J. Shea
From: Paul Bla

P.O. BOX 270
HARTFORD, CONNECTICUT 06
(203) 666-6911

March 9, 1982

Docket No. 50-245
A01975

Director of Nuclear Reactor Regulation
Attn: Mr. Dennis M. Crutchfield
Chief Operating Reactors Branch #5
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

References: (1) D. G. Eisenhut letter to All Operating Boiling Water Reactors, dated July 7, 1980.

(2) W. G. Counsil letter to D. M. Crutchfield, dated October 16, 1980.

(3) D. M. Crutchfield letter to W. G. Counsil, dated September 9, 1981.

(4) W. G. Counsil letter to B. M. Grier, dated July 27, 1981.

(5) W. G. Counsil letter to D. G. Eisenhut, dated December 9, 1981.

(6) W. G. Counsil letter to B M. Grier, dated August 27, 1980.

Gentlemen:

MILLSTONE NUCLEAR POWER STATION, UNIT No. 1
SCRAM DISCHARGE VOLUME

In Reference 1, Northeast Nuclear Energy Company (NNECO) was requested to propose Technical Specification changes for Millstone Unit No. 1 to provide surveillance requirements for scram discharge volume (SDV) vent and drain valves, and limiting conditions for operation and surveillance requirements for reactor protection system (RPS) and control rod block SDV limit switches. We responded to these requests in Reference 2. The NRC Staff and their consultant have reviewed Reference 2 and have identified the need for additional information. The specific items requiring additional information were transmitted to us by Reference 3. This letter is in response to Reference 3.

ITEM NO. 1:

Provide a revised page of the Technical Specifications for Millstone Nuclear Power Station Unit No. 1 where the requested change for surveillance requirements of the drain and vent valves is incorporated, or

Provide technical bases why the requested change is not applicable to Millstone Nuclear Power Station Unit No. 1.

RESPONSE:

The only surveillance requirement for the SDV drain and vent valves contained in Reference 1 is the following:

"4.1.3.1.1 The scram discharge volume drain and vent valves shall be verified open at least once per 31 days.

Our response contained in Reference 2 was fully responsive to the NRC Staff's recommended change. It appears that our Reference 2 submittal was not reviewed against Reference 1 but instead against the NRC Standard Technical Specifications, Revision 3, which we were not requested to nor are we required to comply with.

We currently cycle the SDV vent and drain valves once every three months in accordance with the Millstone Unit No. 1 In-service Inspection Program, which is already incorporated into our Technical Specifications per Section 4.13. This frequency is consistent with ASME Code requirements.

Additionally, as a result of modifications to meet the intent of the BWR Owners' Group Evaluation Criteria regarding the scram discharge system, we intend, during the next refueling outage, to install redundant vent and drain valves and route the vent and drain lines directly to the torus. These measures together with the current cycling frequency will more than adequately assure that the vent and drain valves will operate as required.

Therefore, no increase in the frequency of cycling of the valves nor any further changes to the Millstone Unit No. 1 Technical Specifications are deemed necessary.

ITEM NO. 2:

Provide a reference to that section of the Technical Specifications for Millstone Nuclear Power Station Unit No. 1 which indicates that the reactor protection system SDV water level-high consists of 2 OPERABLE channels containing two limit switches per trip system, for a total of 4 OPERABLE channels containing 4 limit switches per two trip systems, making 1-out-of-2 taken twice logic.

RESPONSE:

The Millstone Unit No. 1 SDV high water level instrumentation for the reactor scram function (i.e., 30-gallon trip) is composed of four OPERABLE channels per two trip systems, constituting a 1-out-of-2 taken twice logic. This information can be found in the Technical Specifications in Section 3.1, Table 3.1.1, and Section 3.1 in the Bases.

ITEM NO. 3:

Provide a reference to the pages of the Technical Specifications for Millstone Nuclear Power Station Unit No. 1 where the requested changes for LCO/surveillance requirements of reactor protection system SDV limit switches are incorporated, or

If the proposed frequency of the required surveillance for Millstone Nuclear Power Station differs from the frequency requested by the Model Technical Specifications, provide technical bases for it.

RESPONSE:

Calibration tests are performed for the NPS SDV high water level instrumentation once every three months for the "RUN", "STARTUP/HOT STANDBY", and "REFUEL" modes of operation. This frequency exceeds that requested by the NRC Staff in References 1 and 2.

This same instrumentation is required to be functionally tested once every one-to-three months for the above-mentioned modes of operation. The flexibility in the Technical Specifications can only be utilized after an adequate data base is developed and even then is dependent on the number of unsafe failures that are discovered. We are currently functionally testing this instrumentation once a month.

The above information is identified in the Technical Specifications in Sections 3.1 and 4.1, Tables 3.1.1, 4.1.1 and 4.1.2, and Figure 4.1.1 in the Bases.

ITEM NO. 4:

Since the proposed frequency of the required surveillance for Millstone Nuclear Power Station Unit No. 1 differs from the frequency of surveillance requested by the Model Technical Specifications, provide technical bases for it.

RESPONSE:

The purpose of functionally testing the control rod block high water level and scram trip bypassed instrumentation is to increase the reliability of such instrumentation. We contend that testing this instrumentation at each refueling outage adequately assures the reliability of this instrumentation, especially in light of the excellent history we have had with this type of instrumentation at Millstone Unit No. 1. As indicated in References 4 and 5, the six

(5) level switches in the scram discharge system have never experienced a failure in more than ten (10) years of operating experience at Millstone Unit No. 1.

If such instrumentation was to fail, the currently existing 3-gallon alarm would provide anticipatory indication to the control room operator of a SDV water level problem. Furthermore, a severe leak into the SDV would result in an automatic scram at 39 gallons.

Additionally, the most likely reason for damage to this instrumentation would be due to a scram. As indicated in Reference 6, we are currently testing the six (6) float level switches after each scram.

Therefore, functionally testing this instrumentation once each refueling outage is deemed adequate to assure the reliability of the above instrumentation.

The calibration tests for the control rod block high water level and scram trip bypassed instrumentation are performed at the frequency recommended by Reference 1.

With respect to Items No. 2 and 3, the Millstone Unit No. 1 Technical Specifications are issued by and readily accessible to the NRC Staff. Therefore, the request for NUSCO to verify the existence of the required limiting conditions for operation and surveillance requirements is questioned.

We trust that the above information adequately resolved the NRC Staff's concerns identified in Reference 3.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

W. G. Counsil

W. G. Counsil
Senior Vice President