TECHNICAL EVALUATION REPORT

BWR SCRAM DISCHARGE VOLUME LONG-TERM MODIFICATIONS

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION

NRC DOCKET NO. 50-219 FRC PROJECT C5506

NRC TAC NO. 42215 FRC ASSIGNMENT 2

NRC CONTRACT NO. NRC-03-81-130 FRC TASK 58

Prepared by

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Prepared for

Nuclear Regulatory Commission
Washington, D.C. 20555

Lead NRC Engineer: K. Eccleston

January 27, 1982

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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 Oyster Creek Nuclear Station 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 are 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 revised page 4.2-2, with the Licensee's agreement to incorporate a revision in the proposed specifications changes that requires cycling each valve at least one complete cycle of full travel at least quarterly, complies with the NRC staff's Model Technical Specifications requirements of paragraphs 4.1.3.1.1a and 4.1.3.1.1b.

The proposed revisions of pages 3.1-7, 3.1-11, 3.1-12a, 4.1-6a (after deleting "Instrument Channel 27b"), and 4.2.2, and unrevised pages 3.2-5 and 4.1-5 meet the remaining surveillance requirements. Table 5-1 on pages 21 and 22 summarizes the evaluation results.

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 Oyster Creek Nuclear Generating Station boiling water reactor (BWR) in regard to "BWR Scram Discharge Volume Long Term Modification," specifically:

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

The evaluation uses 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. 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, Brunswick Units 1 and 2, Duane Arnold, 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 I.

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 an FRC technical evaluation of the submittals. In this TER, FRC has reviewed and evaluated Technical Specifications changes for the Oyster Creek Nuclear Generating Station proposed in a [redacted] by the Licensee, the Jersey Central Power & Light Company (JCP&L), 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. FRC assessed the adequacy with which the JCP&L 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 drain and vent 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.

*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 in 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 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 two operable channels containing two limit switches per trip system, for a total of four operable channels containing four limit switches per two 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 SCRAM DISCHARGE VOLUME 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 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 two operable channels containing two limit switches for SDV water level-high and one operable channel containing one 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 JCP&L submittal for the Oyster Creek Nuclear Generating Station 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:

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

- o 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 "Facility Description and Safety Analysis Report, Oyster Creek Power Plant Unit 1," Vols. I and II, and Oyster Creek 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's request of July 7, 1980 and that the submittal contained sufficient information to permit preparation of a TER without a request for additional information.

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 (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 4.2-2 of the Oyster Creek Technical Specifications as follows (see Appendix B):

"H. The scram discharge volume drain and vent valves shall be verified open at least once per 31 days, except in shutdown mode.*

I. All withdrawn control rods shall be determined OPERABLE by demonstrating the scram discharge volume drain and vent valves OPERABLE. This will be done at least once per refueling cycle by placing the mode switch in shutdown and by verifying that:

a. The drain and vent valves close within 60 seconds after receipt of a signal for control rods to scram, and

b. The scram signal can be reset and the drain and vent valves open when the scram discharge volume trip is bypassed.

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

Basis: The core reactivity limitations (Specification 3.2.A) requires that core reactivity be limited such that the core could be made subcritical at any time during the operating cycle, with the strongest operable control rod fully withdrawn and all other operable rods fully inserted. Compliance with this requirement can be demonstrated conveniently only at the time of refueling."

In addition, the Licensee agreed to revise proposed specifications changes on page 4.2-2 to require cycling each valve at least one complete cycle of full travel at least quarterly.

FRC EVALUATION

The revised page 4.2-2, with the Licensee's agreement to incorporate a revision in the proposed specifications changes that requires cycling each valve at least one complete cycle of full travel at least quarterly, complies with the NRC staff's Model Technical Specifications requirements of paragraphs 4.1.3.1a and 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 two operable channels containing two limit switches per trip system, for a total of four operable channels containing four limit switches per two 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

In response to the NRC staff's Model Technical Specifications requirements of paragraph 3.3.1 and Table 3.1.1-1, the Licensee proposed revising pages 3.1-7 and 3.1-12a of the Oyster Creek Technical Specifications. The revised page 3.1-7 contains Table 3.1.1, "Protective Instrumentation Requirements," with the following information for function - scram on SDV high water level:

"1. Trip setting: $\leq$ 37 gal.

2. Reactor modes in which function must be operable:

Refuel (a), Startup (z), Run (z)

3. Min. No. of Operable or Operating (Tripped) Trip systems: 2

4. Min. No. of Operable Instrument channels per Operable Trip Systems: 2

NOTES:

a. Permissible to bypass, with control rod block, for reactor protection system reset in refuel mode.

z. The bypass function to permit scram reset in the shutdown or refuel mode with control rod block must be operable in this mode." (Note z is taken from the revised page 3.1-12a.)

Page 3.2-5 of the Oyster Creek Technical Specifications gives the reactor protection system response time as follows:

"In the analytical treatment of the transients, 290 milliseconds are allowed between a neutron sensor reaching the scram point and the start of motion of the control rods. This is adequate and conservative when compared to the typical delay of about 210 milliseconds estimated from scram test results."

This addresses the requirements of paragraph 3.3.1 and Table 3.3.1-2.

In response to the requirements of paragraph 4.3.1.1 and Table 4.3.1.1-1 the Licensee submitted the original page 4.1-5 of the Oyster Creek Technical Specifications without revision. This contained Table 4.1.1, "Minimum Check, Calibration and Test Frequency for Protective Instrumentation," with the following information regarding instrument channel SDV high water level:

"1. Check: N/A

2. Calibrate: 1/3 mo.

3. Test: Note 1

4. Remarks (Applies to Test Calibration): By varying level in switch columns.

NOTE 1: Initially once/mo., thereafter according to Fig. 4.1.1, with an interval no less than one month nor more than three months."

FRC EVALUATION

The Licensee's response to the NRC staff's Model Technical Specifications requirements of paragraph 3.3.1 is acceptable. The Oyster Creek reactor protection system SDV water level-high instrumentation consists of two operable channels containing two limit switches per trip system, for a total of four operable channels containing four limit switches per two trip systems, making 1-out-of-2-taken-twice logic. The revised page 3.1-7 with Table 3.1.1 also specifies $\leq$ 37 gal as a trip setting for scram initiation and applicable operating conditions of refuel, startup, and run, which are acceptable.

The reactor protection system response time of 290 milliseconds specified on page 3.2-5 of the Oyster Creek Tech. ical Specifications addresses the requirements of paragraph 3.3.1 and Table 3.3.1-2 and is acceptable.

The original provisions of the Oyster Creek Technical Specifications given in Table 4.1.1, page 4.1-5 (see Appendix B), in regard to reactor protection system SDV water level-high calibration and test frequency for protective instrumentation are acceptable although they differ from paragraph 4.3.1.1 and Table 4.3.1.1-1 of the NRC staff's Model Technical Specifications, which require Channel Calibration each refueling outage (provided by Oyster Creek once per 3 months) and a Channel Functional Test monthly (provided by Oyster Creek initially once per month and thereafter at intervals no shorter than 1 month or longer than 3 months).

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 two operable channels containing two limit switches for SDV water level-high, and one operable channel containing one 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 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.

LICENSEE RESPONSE

In response to the Model Technical Specifications paragraph 3.3.6 and Table 3.3.6-1 requirements, the Licensee proposed revising page 3.1-11 of the Oyster Creek Technical Specifications. The revised page 3.1-11 contains Table 3.1.1, "Protective Instrumentation (Contd)" with the following information for function - rod block SDV water level-high:

"1. Trip setting: 18 gallons

2. Reactor Modes in Which Function Must be Operable:

   Refuel (z), Startup (z), Run (z)

3. Min. No. of Operable or Operating (Tripped) Trip System: 1

4. Min. No. of Operable Instrument Channels per Operable Trip System: 1."

[NOTE z: Same as in LICENSEE RESPONSE, Section 4.2 of this report.]

The Licensee responded to the requirements of paragraph 4.3.6 and Table 4.3.6-1 with a proposed revision of page 4.1-6a of the Oyster Creek Technical Specifications which contains Table 4.1.1, "Minimum Check, Calibration and Test Frequency for Protective Instrumentation," with the following information in regard to instrument channel-SDV (rod block).

"(a) Water level high:

1. Calibrate: Each refueling outage

2. Test: Every 3 months

3. Remarks (Applies to test and calibration): By varying level in switch column

(b) Scram trip bypass:

1. Calibrate: NA

2. Test: Each refueling outage"

FRC EVALUATION

The existing Oyster Creek Nuclear Generating Station scram discharge system has six level switches on the scram discharge volume (see "Facility Description and Safety Analysis Report, Oyster Creek Power Plant Unit No. 1," Appendix B, Section 2) 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 18 gallons (see revised page 3.1-11, Table 3.1.1), 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 $\leq$ 37 gallons (see page 3.1-7, Table 3.1-1 of the Oyster Creek 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. Reference 9, page 50, defines Design Criterion 9 ("Instrumentation shall be provided to aid the operator in the detection of water accumulation in the instrumented volume(s) prior to scram initiation"), gives the technical basis for "Long-Term Evaluation of Scram Discharge System," and defines acceptable compliance ("The present alarm and rod block instrumentation meets this criterion given adequate hydraulic coupling with the SDV headers"). Thus, if the Oyster Creek Nuclear Generating Station scram discharge system is modified (long term) so that the hydraulic coupling between scram discharge headers and instrumented volume is adequate and acceptable, then the present alarm and rod block instrumentation consisting of one operable instrument channel with one limit switch for control rod withdrawal block as specified on revised page 3.1-11 is also acceptable.

In the Oyster Creek Nuclear Generating Station, "Scram Discharge Volume Scram Trips" cannot be bypassed while the reactor is in operational conditions of startup and run (see FSAR Section 7), and operational condition "refuel with more than one control rod withdrawn" is not applicable since interlocks are provided which prevent the withdrawal of more than one control rod with the mode switch in the refuel position. Thus, the NRC staff's Model Technical Specifications requirements of paragraph 3.3.6, Table 3.3.6-1, paragraph 4.3.6, and Table 4.3.6-1 are not applicable to the Oyster Creek Nuclear Generating Station for "Trip Function 5, Scram Discharge Volume Scram Trip Bypassed," and "Instrumentation Channel 27b, Scram Discharge Volume (Rod Block) Scram Trip Bypass" should be deleted from revised page 4.1-6a, Table 4.1.1. Otherwise, the proposed revision of page 4.1-6a is acceptable.

The 18-gallon trip setpoint for control rod withdrawal block instrumentation is acceptable (see revised page 3.1-11 of the Oyster Creek Technical Specifications). The Licensee's proposed revision of page 4.1-6a to meet the requirements of paragraph 4.3.6 and Table 4.3.6-1 is also acceptable after deletion of "Instrument Channel 27b" since it prescribes the Channel Functional Test of each control rod withdrawal block instrumentation channel containing a limit switch once per 3 months and Channel Calibration each refueling outage for SDV water level-high.

5. CONCLUSIONS

Table 5-1 summarizes results of the final review and evaluation of the Oyster Creek proposed Phase 1 Technical Specifications changes for SDV long-term modification in regard to surveillance requirements for SDV drain and vent valves and LCO/surveillance requirements for reactor protection system and control rod block SDV limit switches. The following conclusions were made:

- The revised page 4.2-2, with the Licensee's agreement to incorporate a revision in the proposed specifications changes that requires cycling each valve at least one complete cycle of full travel at least quarterly, complies with the NRC staff's Model Technical Specifications requirements of paragraphs 4.1.3.1.1a and 4.1.3.1.1b.
- "Instrument Channel 27b, SDV (Rod Block) Scram Trip Bypass" should be deleted from revised page 4.1-6a. It is not applicable to the Oyster Creek Nuclear Generating Station.
- The remaining surveillance requirements are met by revised pages 3.1-7, 3.1-11, 3.1-12a, 4.1-6a, and 4.2-2 of the Oyster Creek Technical Specifications, and by pages 3.2-5 and 4.1-5 without revision.

Table 5-1. Evaluation of Phase 1 Proposed Technical Specifications Changes for Scram Discharge Volume Long-Term Modifications Oyster Creek Nuclear Generating Station

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 31 days (p. 4.2-2, revised)	Acceptable
Cycle each valve one complete cycle	Once per 92 days (4.1.3.1.1b)	Once per 92 days (p. 4.2-2, revised)	Acceptable
REACTOR PROTECTION SYSTEM SDV LIMIT SWITCHES			
Minimum operable channels per trip system	2 (3.3.1, Table 3.3.1-1)	2 (pp. 3.1-7 and 3.1-12a, revised)	Acceptable
SDV water level-high response time	NA (3.3.1, Table 3.3.1-2)	290 ms maximum 210 ms test (p. 3.2-5)	Acceptable
SDV water level-high			
Channel functional test	Monthly (4.3.1.1, Table 4.3.1.1-1)	First monthly, then at 1-3 month intervals (p. 4.1-5)	Acceptable
Channel calibration	Each refueling (4.3.1.1, Table 4.3.1.1-1)	Once per 3 months (p. 4.1-5)	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)	1 (p. 3.1-11, revised)	Acceptable*
SDV scram trip bypassed	1 (3.3.6, Table 3.3.6-1)	NA (p. 3.1-11, revised)	Acceptable*
SDV water level-high			
Trip set point	NA (3.3.6, Table 3.3.6-2)	18 gal (p. 3.1-11, revised)	Acceptable
Channel functional test	Quarterly (4.3.6, Table 4.3.6-1)	Quarterly (p. 4.1-6a, **revised)	Acceptable
Channel calibration	Each refueling (4.3.6, Table 4.3.6-1)	Each refueling (p. 4.1-6a, **revised)	Acceptable
SDV scram trip bypassed			
Channel functional test	Monthly (4.3.6, Table 4.3.6-1)	NA	Acceptable*

* See Reference 9, p. 50, and pp. 18 and 19 of this TER.
**"Instrument Channel 27b" should be deleted.

6. REFERENCES

1. IE Bulletin 80-14, "Degradation of BWR Scram Discharge Volume Capacity"
   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

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 EOC-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. Neutron Flux - Upscale, (15)%	NA
b. Flow Biased Simulated Thermal Power - Upscale	≤ (0.09)**
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

*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, 1978. See Regulatory Guide 1.18, November 1977.)

**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 $\geq$ 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	6	2, 5	61
d. Downscale[e]	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

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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)%*	$\leq$ 0.66 W + (45)%*
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

*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.

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Franklin Research Center
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TER-C5506-58

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.

APPENDIX B

JERSEY CENTRAL POWER AND LIGHT COMPANY LETTER OF MARCH 4, 1981

AND

SUBMITTAL WITH PROPOSED TECHNICAL SPECIFICATIONS CHANGES

FOR

OYSTER CREEK NUCLEAR GENERATING STATION

JCP&L / GPU

Jersey Central Power & Light Company
Madison Avenue at Punchbowl Road
Morristown New Jersey 07960
201 539-6111

March 4, 1981

RECEIVED MAR 11 1981 U.S. NUCLEAR REGULATORY COMMISSION

Director
Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Sir:

Subject: Oyster Creek Nuclear Generating Station
Docket No. 50-219
Technical Specification Change Request No.92

In accordance with 10CFR50.59 and 10CFR50.90, Jersey Central Power & Light Company, owner and operator of the Oyster Creek Nuclear Generating Station, Provisional Operating License No. DPR-16, requests changes to Appendix A of that license.

Pursuant to your correspondence of July 7, 1980 concerning the control rod drive scram discharge volume capability, sections 3.1 4.1 and 4.2 of the Oyster Creek Technical Specifications shall be revised.

The Technical Specification Change Request has been reviewed and approved by the Station Superintendent, the Plant Operations Review Committee, and an Independent Safety Review Group in accordance with Sections 6.5 of the Oyster Creek Technical Specifications.

In accordance with your correspondence of July 22, 1980 which determined that the submittal is Class III per 10 CFR 170.22, a check for $4,000 is enclosed.

Very truly yours,

Ivan R. Finfrock, Jr.

Ivan R. Finfrock, Jr.
Vice President

A001
S
1/1
w/check:
$4000.00

la

Enclosure

8103110703

P

Jersey Central Power & Light Company is a Member of the General Public Utilities System

JERSEY CENTRAL POWER & LIGHT COMPANY
OYSTER CREEK NUCLEAR GENERATING STATION

Provisional Operating
License No. DPR-16

Technical Specification
Change Request No. 92

Docket No. 50-219

Applicant submits by this Technical Specification Change Request No. 92 to the Oyster Creek Nuclear Generating Station Technical Specifications, changes to Specifications 3.1, 4.1 and 4.2.

JERSEY CENTRAL POWER & LIGHT COMPANY

BY Ivan R. Finfrock, Jr.
Vice President

STATE OF NEW JERSEY)
)
COUNTY OF MORRIS)

Sworn and subscribed to before me this 4th day of March, 1981.

[illegible]
Notary Public

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF	)	
	)	DOCKET NO. 50-219
JERSEY CENTRAL POWER & LIGHT COMPANY	)	

CERTIFICATE OF SERVICE

This is to certify that a copy of Technical Specification Change Request No. 92 for the Oyster Creek Nuclear Generating Station Technical Specifications, filed with the United States Nuclear Regulatory Commission on March 4, 1981, has this 4th day of March, 1981 been served on the Mayor of Lacey Township, Ocean County, New Jersey by deposit in the United States mail, addressed as follows:

The Honorable Henry Von Spreckelsen
Mayor of Lacey Township
P. O. Box 475
Forked River, New Jersey 08731

JERSEY CENTRAL POWER & LIGHT COMPANY

BY Ivan R. Finfrock
Vice President

DATED: March 4, 1981

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Jersey Central Power & Light Company
Madison Avenue at Punchbowl Road
Morristown New Jersey 07960
201 539-6111

March 4 , 1981

The Honorable Henry Von Spreckelsen
Mayor of Lacey Township
P. O. Box 475
Forked River, New Jersey 08731

Dear Mayor Von Spreckelsen:

Enclosed herewith is one copy of Technical Specification Change Request No. 92 for the Oyster Creek Nuclear Generating Station Operating License.

This document was filed with the United States Nuclear Regulatory Commission on March 4 , 1981.

Very truly yours,

Ivan R. Finfrock, Jr.

Ivan R. Finfrock, Jr.
Vice President

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Enclosure

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Jersey Central Power & Light Company is a Member of the General Public Utilities System

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Applicant hereby requests the Commission to change Appendix A to the License as follows:

1. Sections to be changed:

   3.1, 4.1, 4.2

2. Extent of Changes:

   Requirements are added for making Scram Discharge Volume (SDV) high level and SDV scram trip bypass rod blocks in Section 3.1 with appropriate surveillance requirements in Section 4.1. Section 4.2 was changed to add requirements for ensuring the operability of the SDV vent and drain valves.

3. Changes Requested:

   The requested changes are on the attached revised Technical Specification page

4. Discussion and Safety Evaluation:

   The addition of the requirement for making the SDV high level and scram trip bypass rod blocks do not add any additional likelihood of an accident or increase the severity of any accident since the present plant design has these functions as rod blocks, presently there are no technical specification requirements.

   The additional requirement to verifying the SDV vent and drain valves are open at least every 31 days does not change any plant design considerations and only serves to verify a condition that should exist.

   The requirement for demonstrating the function of the SDV vent and drain valves when control rod scram testing is conducted does not introduce any new testing and only serves to verify the proper functioning of these valves during the scram tests.

TABLE 3.1.1 PROTECTIVE INSTRUMENTATION REQUIREMENTS

Function	Trip Setting	Reactor Modes in which Function Must Be Operable: Shutdown	Refuel	Startup	Run	Min. No. of Operable or Operating (Tripped) Trip Systems	Min. No. of Operable Instrument Channels Per Operable Trip Systems	Action Requir
A. Scram								Insert contr rods
1. Manual Scram		X	X	X	X	2	1	
2. High Reactor Pressure	**		X(s)	X	X	2	2	
3. High Drywell Pressure	≤ 2 psig		X(u)	X(u)	X	2	2	
4. Low Reactor Water Level	**		X	X	X	2	2	
5. High Water Level in Scram Discharge Volume	≤ 37 gal.		X(a)	X(z)	X (z)	2	2	
6. Low Condenser Vacuum	≥ 23" Hg		X(b)	X(b)	X	2	2	
7. High Radiation in Main Steam-line Tunnel	≤ 10 x normal background		X(s)	X	X	2	2	
8. Average Power Range Monitor (APRM)	**		X(c,s)	X(c)	X(c)	2	3	
9. Intermediate Range Monitor (IRM)	**		X(d)	X(d)		2	3	

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TABLE 3.1.1 PROTECTIVE INSTRUMENTATION REQUIREMENTS (CONTD)

Function	Trip Setting	Reactor Modes In Which Function Must be Operable: Shutdown	Refuel	Startup	Run	Min.No. of Operable or Operating (Tripped) Trip Systems	Min.No.of Operable Instrument Channels Per Operable Trip Systems	Action Required*
Rod Block								No control rod withdrawals permitted
1. SRM Upscale	5×10^5 cps		X	X(1)		1	3(y)	
2. SRM Downscale	$100\ cps^{(f)}$		X	X(1)		1	3(y)	
3. IRM Downscale	5/125 fullscale(g)		X	X		2	3	
4. APRM Upscale	**		X(s)	X	X	2	3(c)	
5. APRM Downscale	2/150 fullscale				X	2	3(c)	
6. IRM Upscale	108/125 fullscale		X	X		2	3	
7. Scram Discharge Volume								
a) Water level high	18 gallons		X (z)	X (z)	X (z)	1	1	
Condenser Vacuum Pump Isolation								Insert control rods
1. High Radiation in Main Steam Tunnel	10 x Normal Background	During Startup and run when vacuum pump 1 operating				2	2	
Diesel Generator Load Sequence Timers	Time delay after energiz. of relay							
1. Containment Spray Pump	40 sec ± 15%	X	X	X	X	2(m)	1(n)	Consider containment spray loop inoperable and comply with spec. 3.4.C (see Note

TABLE 3.1.1 (CON'D)

i. The interlock is not required during the start-up test program and demonstration of plant electrical output but shall be provided following these actions.

j. Not required below 40% of turbine rated steam flow.

k. All four (4) drywell pressure instrument channels may be made inoperable during the integrated primary containment leakage rate test (See Specification 4.5), provided that primary containment integrity is not required and that no work is performed on the reactor or its connected systems which could result in lowering the reactor water level to less than 4'8" above the top of the active fuel.

l. Bypassed in IRM Ranges 8, 9, & 10.

m. There is one time delay relay associated with each of two pumps.

n. One time delay relay per pump must be operable.

o. There are two time delay relays associated with each of two pumps.

p. Two time delay relays per pump must be operable.

q. Manual initiation of affected component can be accomplished after the automatic load sequencing is completed.

r. Time delay starts after closing of containment spray pump circuit breaker.

s. These functions not required to be operable with the reactor temperature less than 212°F and the vessel head removed or vented.

t. These functions may be inoperable or bypassed when corresponding portions in the same core spray system logic train are inoperable per Specification 3.4.A.

u. These functions are not required to be operable when primary containment integrity is not required to be maintained.

v. These functions not required to be operable when the ADS is not required to be operable.

w. These functions must be operable only when irradiated fuel is in the fuel pool or reactor vessel and secondary containment integrity is required per specification 3.5.B.

y. The number of operable channels may be reduced to 2 per Specification 3.9-E and F.

z. The bypass function to permit scram reset in the shutdown or refuel mode with control rod block must be operable in this mode.

Amendment No. 44

TABLE 4.1.1

MINIMUM CHECK, CALIBRATION AND TEST FREQUENCY FOR PROTECTIVE INSTRUMENTATION

	Instrument Channel	Check	Calibrate	Test	Remarks (Applies to Test and Calibration)
1.	High Reactor Pressure	N A	1/3 mo.	Note 1	By application of test pressure
2.	High Drywell Pressure (scram)	N A	1/3 mo.	Note 1	" " " " "
3.	Low Reactor Water Level	1/d	1/3 mo.	Note 1	" " " " "
4.	Low-Low Water Level	1/d	1/3 mo.	Note 1	" " " " "
5.	High Water Level in Scram Discharge Volume (Scram)	NA	1/3 mo.	Note 1	By varying level in switch columns
6.	Low-Low-Low Water Level	NA	1/3 mo.	Note 1	By application of test pressure
7.	High Flow in Main Steamline	1/d	1/3 mo.	Note 1	" " " " "
8.	Low Pressure in Main Steamline	N A	1/3 mo.	Note 1	" ." " " "
9.	High drywell Pressure (Core Cooling)	1/d		Note 1	" " " " "
10.	Main Steam Isolation Valve (Scram)	N A	N A	1/3 mo.	By exercising valve
11.	APRM Level	N A	1/3d	N A	Output adjustment using operational type heat balance during power operation

NOTE 1: Initially once/mo., thereafter according to Fig. 4.1.1, with an interval not less than one month nor more than three months.

NOTE 2: At least daily during reactor power operation, the reactor neutron flux peaking factor shall be estimated and the flow-referenced APRM scram and rod block settings shall be adjusted, if necessary, as specified in Section 2.3, Specifications (1) (a) and (2) (a).

	Instrument Channel	Check	Calibrate	Test	Remarks (Applies to Test & Calibration)
19.	Manual Scram Buttons	N A	NA	1/3 mo	
20.	High Temperature Main Steamline Tunnel	N A	Each refueling outage	Each refueling outage	Using heat source box
21.	SRM	*	*	*	Using built-in calibration equipment
22.	Isolation Condenser High Flow P (Steam and Water)	N A	1/3 mo	1/3 mo	By application of test pressure
23.	Turbine Trip Scram	N A		Every 3 months	
24.	Generator Load Rejection Scram	N A	Every 3 months	Every 3 months	
25.	Recirculation Loop Flow	N A	Each Refueling Outage	NA	By application of test pressure
26.	Low Reactor Pressure Core Spray Valve Permissive	N A	Every 3 months	Every 3 months	By application of test pressure
27.	Scram Discharge Volume (Rod Block)				
	a) Water level high	N A	Each Refueling Outage	Every 3 months	By varying level in switch column.
	b) Scram trip bypass	N A	N A	Each refueling outage	

*Calibrate prior to startup and normal shutdown and thereafter check 1/s and test 1/wk until no longer required.

F. At specific power operating conditions, the actual control rod configuration will be compared with the expected configuration based upon appropriately corrected past data. This comparison shall be made every equivalent full power month. The initial rod inventory measurement performed when equilibrium conditions are established after a refueling or major core alteration will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle.

G. At power operating conditions, the actual control rod density will be compared with the 3.5 percent control rod density included in Specification 3.3.B.5. This comparison shall be made every equivalent full power month.

H. The scram discharge volume drain and vent valves shall be verified open at least once per 31 days, except in shutdown mode.*

I. All withdrawn control rods shall be determined OPERABLE by demonstrating the scram discharge volume drain and vent valves OPERABLE. This will be done at least once per refueling cycle by placing the mode switch in shutdown and by verifying that:

a. **The drain and vent valves close within 60 seconds after receipt of a signal for control rods to scram, and**

b. **The scram signal can be reset and the drain and vent valves open when the scram discharge volume trip is bypassed.**

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

Basis: The core reactivity limitation (Specification 3.2.A) requires that core reactivity be limited such that the core could be made subcritical at any time during the operating cycle, with the strongest operable control rod fully withdrawn and all other operable rods fully inserted. Compliance with this requirement can be demonstrated conveniently only at the time of refueling. Therefore, the demonstration must be such that it will apply to the entire subsequent fuel cycle. The demonstration is performed with the reactor core in the cold, xenon-free condition and will show that the reactor is sub-critical at that time by at least $R + 0.25\% \Delta k$ with the highest worth operable control rod fully withdrawn.

The value of R is the difference between two calculated values of reactivity of the cold, xenon-free core with the strongest operable control rod fully withdrawn. The reactivity value at the beginning of life is subtracted from the maximum reactivity value anytime later in life to determine R, which must be a positive quantity or its value is conservatively taken as zero. The value of R shall include the potential shutdown margin loss assuming full B_4C settling in all

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inventory to be present in the core. The value 0.25% Δk in the expression R + 0.25% Δk serves at the beginning of life as a finite, demonstrable shutdown margin. This margin is demonstrated by full withdrawal of the strongest rod and partial withdrawal of a diagonally adjacent rod to a position calculated to insert an R + 0.25% Δk reactivity. Observation of subcriticality in this condition assures subcriticality with not only the strongest rod fully withdrawn but at least an R + 0.25% Δk margin beyond this.

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the specified limits, provide the required protection. In the analytical treatment of the transients, 190 milliseconds are allowed between a neutron sensor reaching the scram point and the start of motion of the control rods. This is adequate and conservative when compared to the typical time delay of about 210 mil seconds estimated from scram test results. Approximately the first 90 millisec of each of these time intervals result from the sensor and circuit delays; then the pil scram solenoid de-energizes. Approximately 120 milliseconds later, the control rod motion is estimated to actually begin. However, 200 milliseconds is conservatively assumed for this time interval in the transient analyses and this is also included in t allowable scram insertion times of specification of 3-2-3. The specified limits provide sufficient scram capability to accommodate failure to scram of any one operable rod. This failure is in addition to any inoperable rods that exist in the core, provided that those inoperable rods met the core reactivity Specification 3.2.A.

Control rods (8) which cannot be moved with control rod drive pressure are clearly indicative of an abnormal operating condition on the affected rods and are, therefore, considered to be inoperable. Inoperable rods are valved out of service to fix their position in the core and assure predictable behavior. If the rod is fully inserted and then valved out of service, it is in a safe position of maximum contribution to shutdown reactivity. If it is valved out of service in a non-fully inserted position, that position is required to be consistent with the shutdown reactivity limitation stated in Specification 3.2.A, which assures the core can be shutdown at all times with control rods.

Although there are many possible patterns of inoperable control rods which would meet this specification, the operator will be provided with only a limited number of predetermined patterns which allow him to continue operation with inoperable rods. The availability of allowable patterns to the operator assures that information for determining compliance with the specification is immediately available to him at the time a control rod becomes inoperable and does not require reliance on calculations at that time before compliance can be determined.

The allowable inoperable rod patterns will be determined using information obtained in the startup test program supplemented by calculations. During initial startup, the reactivity condition of the as-built core will be determined. Also, sub-critical patterns of widely separated withdrawn control rods will be observed in the control rod sequences being used. The observations, together with calculated strengths of the strongest control rods in these patterns will comprise a set of allowable separations of malfunctioning rods. During the fuel cycle, similar observations made during any cold shutdown can be used to update and/or increase the allowable patterns.

The number of rods permitted to be valved out of service could