Docket No. 50-220

JUN 24 1983

Mr. G. K. Rhode
Senior Vice President
Niagara Mohawk Power Corporation
300 Erie Boulevard West
Syracuse, New York 13202

Dear Mr. Rhode:

The Commission has issued the enclosed Confirmatory Order for the Nine Mile Point Nuclear Power Station, Unit No. 1. This Order confirms your commitment, as stated in your letters dated October 15, 1980, April 1, 1981, October 9, 1981 and December 15, 1982 to install permanent Scram Discharge System Modifications by September 30, 1984 or in any case, prior to operation in Cycle 8. The modifications will conform to criteria developed by a BWR Owners Subgroup and endorsed by our Generic Safety Evaluation Report on the BWR Scram Discharge System dated December 1, 1980.

Also enclosed are proposed Model Technical Specifications which are provided as guidance for preparing Technical Specifications changes that will be required to be approved before operation with the modified system resumes.

A copy of this Order is being filed with the Office of the Federal Register for publication.

Sincerely,

Original signed by
D. B. Vassallo

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Order
2. Model Technical Specifications
   (Supersedes our July 7, 1980 Model TS)

cc w/enclosures
See next page

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NRC FORM 318 (10-80) NRCM 0240 **OFFICIAL RECORD COPY** USGPO: 1981—335-960

Mr. G. K. Rhode
Niagara Mohawk Power Corporation
Nine Mile Point Nuclear Station, Unit No. 1

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UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

In the Matter of (NIAGARA MOHAWK POWER CORPORATION) (Nine Mile Point Nuclear Power Station, Unit No. 1)	))))	Docket No. 50-220

CONFIRMATORY ORDER

I.

The Niagara Mohawk Power Corporation (the licensee) is the holder of Facility Operating License No. DPR-63 which authorizes the licensee to operate the Nine Mile Point Nuclear Power Station, Unit No. 1 (the facility), at power levels not in excess of 1850 megawatts thermal. The facility is a boiling water reactor located at the licensee's site in Oswego County, New York.

II.

During a routine shutdown of Browns Ferry Unit No. 3 on June 28, 1980, 76 of 185 control rods failed to fully insert in response to a manual scram from approximately 30% power. All rods were subsequently inserted within 15 minutes and no reactor damage or hazard to the public occurred. However, the event did cause an in-depth review of the current BWR Control Rod Drive Systems which identified design deficiencies requiring both short and long-term corrective measures. These measures were set forth in the "Generic Safety Evaluation Report BWR Scram Discharge System", dated December 1, 1980, prepared by the NRC staff.

To provide reasonable assurance of safe operation pending implementation of long-term corrective measures, the short-term corrective measures have been implemented by IE Bulletin 80-17 (with supplements) and Orders issued on January 9, 1981.

The Generic Safety Evaluation Report (SER) dated December 1, 1980, endorsed the criteria and technical bases that were developed by a BWR Owners Subgroup for use in implementing permanent system modifications to correct identified deficiencies. These criteria were designated as either functional, safety, operating, design, or surveillance, and when taken as a whole, comprise an adequate set of criteria to resolve the issues raised during the Browns Ferry event investigation.

The SER further described an acceptable means of compliance with each criterion. Pre-implementation approval of permanent modifications using the methods described in the SER for compliance with the criteria will not be required. Alternate methods of compliance will require specific NRC approval in advance of implementation.

In addition to the criteria proposed by the BWR Owners Subgroup, the SER added a criterion to address the potential for common cause failures of the scram level instrumentation. An acceptable means of complying with this criterion was the addition of diversity in the design. The addition of diverse instrumentation on the Scram Discharge Instrumented Volume will minimize recurrence of known common cause failures and thus improve system reliability.

Therefore, we have concluded that diverse instrumentation should be provided as required in the SER, with one exception: Alternative 2(d)(ii) has been deleted as a possible means of providing diversity, due to its reliance on prompt operator action. The use of level sensors employing different operating principles, or the use of level sensors made by a different manufacturer, continues to be acceptable means of providing diverse instrumentation.

On October 1, 1980 letters were sent to all BWR licensees requesting a commitment to reevaluate the present scram system and modify it as necessary to meet the design and performance criteria developed by the BWR Owners Subgroup. The letter also requested a schedule for implementation.

III.

Because the implementation of modifications to meet the criteria proposed by the BWR Owners Subgroup and endorsed by the NRC staff will restore the margins of safety in the BWR scram system, we have determined that these modifications should be completed on an expeditious schedule. In response to our letter of October 1, 1980 and additional discussions with the NRC staff, the licensee committed, by letters dated October 15, 1980, April 1, 1981, October 9, 1981 and December 15, 1982 to install the long term modifications before reactor operation in Cycle 8. These commitments were confirmed in a June 8, 1983 telephone conversation with the licensee's staff. In view of the foregoing, we have determined that these commitments are required in the interest of public health and safety and should, therefore, be confirmed by an immediately effective order.

IV.

Accordingly, pursuant to sections 103, 161i, and 182 of the Atomic Energy Act of 1954, as amended, and the Commission's regulations in 10 CFR Parts 2 and 50, IT IS HEREBY ORDERED EFFECTIVE IMMEDIATELY THAT:

1. The licensee shall install the long term BWR scram discharge system modifications in conformance with the staff's Generic SER, which incorporates the BWR Owners Subgroup criteria, before reactor operation in Cycle 8 or, in the alternative, the licensee shall place and maintain the facility in a cold shutdown or refueling mode of operation until such modifications are made. Extensions of time for installation may be granted for good cause shown by the licensee. The modifications shall include diverse instrumentation as provided in the SER with the exception that alternative 2(d)(ii) will not be accepted.

2. For those cases in which a different method of complying with the criteria than that described in the SER is chosen, the licensee shall submit the design details and supporting analyses for approval to the Director, Division of Licensing, Washington, D. C. 20555 with a copy to the Regional Administrator of the appropriate NRC regional office, at least 3 months prior to the required implementation date.

3. Technical Specification changes required for operation with the modified system shall be submitted at least 3 months prior to the required implementation date.

V.

The licensee may request a hearing on this Order within 25 days of the date of publication of this Order in the Federal Register. A request for hearing shall be submitted to the Director, Division of Licensing, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. A copy of the request shall also be sent to the Executive Legal Director at the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. A REQUEST FOR HEARING SHALL NOT STAY THE IMMEDIATE EFFECTIVENESS OF THIS ORDER.

If a hearing is requested by the licensee, the Commission will issue an order designating the time and place of any such hearing. If a hearing is held, the issue to be considered at such a hearing shall be whether the licensee should comply with the conditions set forth in Section IV of this Order.

The request for information made in this Order was approved by OMB under clearance number 3150-0083 which expires on December 31, 1983. Comments on burden and duplication may be directed to the Office of Management and Budget, Reports Management, Room 3208, New Executive Office Building, Washington, D.C.

This Order is effective upon issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation

Dated at Bethesda, Maryland
this 24th day of June 1983.

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor protection system instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Tablè 2.2.1-1.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

With a reactor protection system instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2.1-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High	≤ (120)/(125) divisions of full scale	≤ (122)/(125) divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-Upscale, Setdown	≤ (15)% of RATED THERMAL POWER	≤ (20)% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power-Upscale		
1) Flow Biased	≤ 0.66 W+(51)%, with a maximum of	≤ 0.66 W+(54)%, with a maximum of
2) High Flow Clamped	≤ (113.5)% of RATED THERMAL POWER	≤ (115.5)% of RATED THERMAL POWER
c. Fixed Neutron Flux-Upscale	≤ (118)% of RATED THERMAL POWER	≤ (120)% of RATED THERMAL POWER
d. Inoperative	NA	NA
(e. Downscale	≥ (5)% of RATED THERMAL POWER	≥ (3)% of RATED THERMAL POWER)
3. Reactor Vessel Steam Dome Pressure - High	≤ (1045) psig	≤ (1065) psig
4. Reactor Vessel Water Level - Low, Level 3	≥ (12.5) inches above instrument zero*	≥ (11.0) inches above instrument zero
5. Main Steam Line Isolation Valve - Closure	≤ (6)% closed	≤ (7)% closed
6. Main Steam Line Radiation - High	≤ (2.5) x full power background	≤ (3.0) x full power background
7. (Primary Containment) (Drywell) Pressure - High	≤ (1.69) psig	≤ (1.89) psig
8. Scram Discharge Volume Water Level - High	≤ (36)% of full scale	≤ (39)% of full scale
9. Turbine Stop Valve - Closure	≤ (5)% closed	≤ (7)% closed
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≥ (500) psig	≥ ( ) psig
11. Reactor Mode Switch Shutdown Position	NA	NA
12. Manual Scram	NA	NA

*See Bases Figure B 3/4 3-1.

LIMITING SAFETY SYSTEM SETTING

BASES

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

8. Scram Discharge Volume Water Level-High

The scram discharge volume receives the water displaced by the motion of the control rod drive pistons during a reactor scram. Should this volume fill up to a point where there is insufficient volume to accept the displaced water at pressures below 65 psig, control rod insertion would be hindered. The reactor is therefore tripped when the water level has reached a point high enough to indicate that it is indeed filling up, but the volume is still great enough to accommodate the water from the movement of the rods at pressures below 65 psig when they are tripped. The trip setpoint for each scram discharge volume is equivalent to a contained volume of ( ) gallons of water.

9. Turbine Stop Valve-Closure

The turbine stop valve closure trip anticipates the pressure, neutron flux, and heat flux increases that would result from closure of the stop valves. With a trip setting of (5)% of valve closure from full open, the resultant increase in heat flux is such that adequate thermal margins are maintained during the worst case transient (assuming the turbine bypass valves (fail to) operate).

10. Turbine Control Valve Fast Closure, Trip Oil Pressure-Low

The turbine control valve fast closure trip anticipates the pressure, neutron flux, and heat flux increase that could result from fast closure of the turbine control valves due to load rejection coincident with failure of the turbine bypass valves. The Reactor Protection System initiates a trip when fast closure of the control valves is initiated by the fast acting solenoid valves and in less than (30) milliseconds after the start of control valve fast closure. This is achieved by the action of the fast acting solenoid valves in rapidly reducing hydraulic trip oil pressure at the main turbine control valve actuator disc dump valves. This loss of pressure is sensed by pressure switches whose contacts form the one-out-of-two-twice logic input to the Reactor Protection System. This trip setting, a faster closure time, and a different valve characteristic from that of the turbine stop valve, combine to produce transients which are very similar to that for the stop valve. Relevant transient analyses are discussed in Section (15.1.0) of the Final Safety Analysis Report.

11. Reactor Mode Switch Shutdown Position

The reactor mode switch Shutdown position is a redundant channel to the automatic protective instrumentation channels and provides additional manual reactor trip capability.

12. Manual Scram

The Manual Scram is a redundant channel to the automatic protective instrumentation channels and provides manual reactor trip capability.

3/4.1.3 CONTROL RODS

CONTROL ROD OPERABILITY

LIMITING CONDITION FOR OPERATION

3.1.3.1 All control rods shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

a. With one control rod inoperable due to being immovable, as a result of excessive friction or mechanical interference, or known to be untrippable:

   1. Within one hour:

      a) Verify that the inoperable control rod, if withdrawn, is separated from all other inoperable control rods by at least two control cells in all directions.

      b) Disarm the associated directional control valves** either:
         1) Electrically, or
         2) Hydraulically by closing the drive water and exhaust water isolation valves.

      c) Comply with Surveillance Requirement 4.1.1.c.

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

   2. Restore the inoperable control rod to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

b. With one or more control rods trippable but inoperable for causes other than addressed in ACTION a, above:

   1. If the inoperable control rod(s) is withdrawn, within one hour:

      a) Verify that the inoperable withdrawn control rod(s) is separated from all other inoperable control rods by at least two control cells in all directions, and

      b) Demonstrate the insertion capability of the inoperable withdrawn control rod(s) by inserting the control rod(s) at least one notch by drive water pressure within the normal operating range*.

      Otherwise, insert the inoperable withdrawn control rod(s) and disarm the associated directional control valves** either:

      a) Electrically, or
      b) Hydraulically by closing the drive water and exhaust water isolation valves.

*The inoperable control rod may then be withdrawn to a position no further withdrawn than its position when found to be inoperable.

**May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

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.

   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. At least once per 31 days verifying each valve to be open,* and

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

4.1.3.1.2 When above the (preset power level) (low power setpoint) 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.

**May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

SURVEILLANCE REQUIREMENTS (Continued)

4.1.3.1.4 The scram discharge volume shall be determined OPERABLE by demonstrating:

a. The scram discharge volume drain and vent valves OPERABLE, when control rods are scram tested from a normal control rod configuration of less than or equal to (50)% ROD DENSITY at least once per 18 months, by verifying that the drain and vent valves:

    1. Close within (30) seconds after receipt of a signal for control rods to scram, and
    2. Open when the scram signal is reset.

b. Proper (float) (level sensor) response by performance of a CHANNEL FUNCTIONAL TEST of the scram discharge volume scram and control rod block level instrumentation (ΔT level measuring system) (after each scram from a pressurized condition) (at least once per 31 days).

ROD BLOCK MONITOR

LIMITING CONDITION FOR OPERATION

3.1.4.3 Both rod block monitor (RBM) channels shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to (30)% of RATED THERMAL POWER.

ACTION:

a. With one RBM channel inoperable, restore the inoperable RBM channel to OPERABLE status within 24 hours and verify that the reactor is not operating on a LIMITING CONTROL ROD PATTERN; otherwise, place the inoperble rod block monitor channel in the tripped condition within the next hour.

b. With both RBM channels inoperable, place at least one inoperable rod block monitor channel in the tripped condition within one hour.

SURVEILLANCE REQUIREMENTS

4.1.4.3 Each of the above required RBM channels shall be demonstrated OPERABLE by performance of a:

a. CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION at the frequencies and for the OPERATIONAL CONDITIONS specified in Table 4.3.6-1.

b. CHANNEL FUNCTIONAL TEST prior to control rod withdrawal when the reactor is operating on a LIMITING CONTROL ROD PATTERN.

3/4.1.3 CONTROL RODS

The specification of this section ensure that (1) the minimum SHUTDOWN MARGIN is maintained, (2) the control rod insertion times are consistent with those used in the accident analysis, and (3) limit the potential effects of the rod drop accident. The ACTION statements permit variations from the basic requirements but at the same time impose more restrictive criteria for continued operation. A limitation on inoperable rods is set such that the resultant effect on total rod worth and scram shape will be kept to a minimum. The requirements for the various scram time measurements ensure that any indication of systematic problems with rod drives will be investigated on a timely basis.

Damage within the control rod drive mechanism could be a generic problem, therefore with a control rod immovable because of excessive friction or mechanical interference, operation of the reactor is limited to a time period which is reasonable to determine the cause of the inoperability and at the same time prevent operation with a large number of inoperable control rods.

Control rods that are inoperable for other reasons are permitted to be taken out of service provided that those in the nonfully-inserted position are consistent with the SHUTDOWN MARGIN requirements.

The number of control rods permitted to be inoperable could be more than the eight allowed by the specification, but the occurrence of eight inoperable rods could be indicative of a generic problem and the reactor must be shutdown for investigation and resolution of the problem.

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent the MCPR from becoming less than (1.06) during the limiting power transient analyzed in Section (15.__) of the FSAR. This analysis shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the specifications, provide the required protection and MCPR remains greater than (1.06). The occurrence of scram times longer then those specified should be viewed as an indication of a systemic problem with the rod drives and therefore the surveillance interval is reduced in order to prevent operation of the reactor for long periods of time with a potentially serious problem.

The scram discharge volume is required to be OPERABLE so that it will be available when needed to accept discharge water from the control rods during a reactor scram and will isolate the reactor coolant system from the containment when required.

Control rods with inoperable accumulators are declared inoperable and Specification 3.1.3.1 then applies. This prevents a pattern of inoperable accumulators that would result in less reactivity insertion on a scram than has been analyzed even though control rods with inoperable accumulators may still be inserted with normal drive water pressure. Operability of the accumulator ensures that there is a means available to insert the control rods even under the most unfavorable depressurization of the reactor.

CONTROL RODS (Continued)

Control rod coupling integrity is required to ensure compliance with the analysis of the rod drop accident in the FSAR. The overtravel position feature provides the only positive means of determining that a rod is properly coupled and therefore this check must be performed prior to achieving criticality after completing CORE ALTERATIONS that could have affected the control rod coupling integrity. The subsequent check is performed as a backup to the initial demonstration.

In order to ensure that the control rod patterns can be followed and therefore that other parameters are within their limits, the control rod position indication system must be OPERABLE.

The control rod housing support restricts the outward movement of a control rod to less than (3) inches in the event of a housing failure. The amount of rod reactivity which could be added by this small amount of rod withdrawal is less than a normal withdrawal increment and will not contribute to any damage to the primary coolant system. The support is not required when there is no pressure to act as a driving force to rapidly eject a drive housing.

The required surveillance intervals are adequate to determine that the rods are OPERABLE and not so frequent as to cause excessive wear on the system components.

3/4.1.4 CONTROL ROD PROGRAM CONTROLS

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to result in a peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. The specified sequences are characterized by homogeneous, scattered patterns of control rod withdrawal. When THERMAL POWER is greater than (20)% of RATED THERMAL POWER, there is no possible rod worth which, if dropped at the design rate of the velocity limiter, could result in a peak enthalpy of 280 cal/gm. Thus requiring the RSCS and RWM to be OPERABLE when THERMAL POWER is less than or equal to (20)% of RATED THERMAL POWER provides adequate control.

The RSCS and RWM provide automatic supervision to assure that out-of-sequence rods will not be withdrawn or inserted.

The analysis of the rod drop accident is presented in Section (15.__) of the FSAR and the techniques of the analysis are presented in a topical report, Reference 1, and two supplements, References 2 and 3.

The RBM is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided. Tripping one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the written sequence used by the operator for withdrawal of control rods.

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 the inoperable channel(s) and/or that trip system in the tripped condition* within one hour. The provisions of Specification 3.0.4 are not applicable.

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 trip system** in the tripped condition within one hour and take the ACTION required by Table 3.3.1-1.

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 functional unit 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 channel per trip system 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 system.

*An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.3.1-1 for that Trip Function shall be taken.

**If more channels are inoperable in one trip system than in the other, place the trip system with more inoperable channels in the tripped condition, except when this would cause the Trip Function to occur.

TABLE 3.3.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
1. Intermediate Range Monitors(b):			
a. Neutron Flux - High	2 3, 4 5(c)	3 2 3(d)	1 2 3
b. Inoperative	2 3, 4 5	3 2 3(d)	1 2 3
2. Average Power Range Monitor(e):			
a. Neutron Flux - Upscale, Setdown	2 3 5(c)	2 2 2(d)	1 2 3
b. Flow Biased Simulated Thermal Power - Upscale	1	2	4
c. Fixed Neutron Flux - Upscale	1	2	4
d. Inoperative	1, 2 3 5(c)	2 2 2(d)	1 2 3
(e. Downscale	1(g)	2	4)
3. Reactor Vessel Steam Dome Pressure - High	1, 2(f)	2	1
4. Reactor Vessel Water Level - Low, Level 3	1, 2	2	1
5. Main Steam Line Isolation Valve - Closure	1(g)	4	4

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
6. Main Steam Line Radiation - High	1, $2^{(f)}$	2	5
7. (Primary Containment) (Drywell) Pressure - High	1, $2^{(h)}$	2	1
8. Scram Discharge Volume Water Level - High	1, 2 $5^{(i)}$	2 2	1 3
9. Turbine Stop Valve - Closure	$1^{(j)}$	$4^{(k)}$	6
10. Turbine Control Valve Fast Closure, Valve Trip System Oil Pressure - Low	$1^{((g))(j)}$	$2^{(k)}$	6
11. Reactor Mode Switch Shutdown Position	1, 2 3, 4 5	1 1 1	1 7 3
12. Manual Scram	1, 2 3, 4 5	2 2 2	1 8 9

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION

ACTION 1 - Be in at least HOT SHUTDOWN within 12 hours.

ACTION 2 - Verify all insertable control rods to be inserted in the core and lock the reactor mode switch in the Shutdown position within one hour.

ACTION 3 - Suspend all operations involving CORE ALTERATIONS* and insert all insertable control rods within one hour.

ACTION 4 - Be in at least STARTUP within 6 hours.

ACTION 5 - Be in STARTUP with the main steam line isolation valves closed within 6 hours or in at least HOT SHUTDOWN within 12 hours.

ACTION 6 - 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 7 - Verify all insertable control rods to be inserted within one hour.

ACTION 8 - Lock the reactor mode switch in the Shutdown position within one hour.

ACTION 9 - Suspend all operations involving CORE ALTERATIONS*, and insert all insertable control rods and lock the reactor mode switch in the SHUTDOWN position 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) This function shall be automatically bypassed when the reactor mode switch is in the Run position.

(c) 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.

(d) The non-coincident NMS reactor trip function logic is such that all channels go to both trip systems. Therefore, when the "shorting links" are removed, the Minimum OPERABLE Channels Per Trip System is 4 APRMS and 6 IRMS.

(e) 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.

(f) This function is not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed per Specification 3.10.1.

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

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

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

(j) This function shall be automatically bypassed when turbine first stage pressure is $\leq$ (250) psig, equivalent to THERMAL POWER less than (30)% of RATED THERMAL POWER.

(k) 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 - High	NA
b. Inoperative	NA
2. Average Power Range Monitor*:	
a. Neutron Flux - Upscale, Setdown	NA
b. Flow Biased Simulated Thermal Power - Upscale	≤ (0.09)(**)
c. Fixed Neutron Flux - Upscale	≤ (0.09)
d. Inoperative	NA
(e. Downscale	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) (Drywell) 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 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, 6 ± 1 seconds.

#Measured from start of turbine control valve fast closure.

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors:				
a. Neutron Flux - High	S/U,S,(b)	S/U[c], W	R	2
	S	W	R	3, 4, 5
b. Inoperative	NA	W	NA	2, 3, 4, 5
2. Average Power Range Monitor[f]:				
a. Neutron Flux - Upscale, Setdown	S/U,S,(b)	S/U[c], W	SA	2
	S	W	SA	3, 5
b. Flow Biased Simulated Thermal Power - Upscale	S,D((g))	S/U[c], W	W[d][e],SA,(R[h])	1
c. Fixed Neutron Flux - Upscale	S	S/U[c], W	W[d], SA	1
d. Inoperative	NA	W	NA	1, 2, 3, 5
(e. Downscale	S	W	SA	1)
3. Reactor Vessel Steam Dome Pressure - High	(S)	M	(R)	1, 2
4. Reactor Vessel Water Level - Low, Level 3	(S)	M	(R)	1, 2
5. Main Steam Line Isolation Valve - Closure	NA	M	R	1
6. Main Steam Line Radiation - High	S	M	R	1, 2[i]
7. (Primary Containment) (Drywell) Pressure - High	(S)	M	(R)	1, 2

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
8. Scram Discharge Volume Water Level - High	(S)	M	(R)	1, 2, 5(j)
9. Turbine Stop Valve - Closure	(S)	M	(R)	1
10. Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low	(S)	M	(R)	1
11. Reactor Mode Switch 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) The IRM and SRM channels shall be determined to overlap for at least (½) decades during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for at least (½) decades during each controlled shutdown, if not performed within the previous 7 days.

(c) Within 24 hours prior to startup, 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 is greater than 2% of RATED THERMAL POWER. 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 flow biased channel 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.

(g) Verify measured core flow to be greater than or equal to established core flow at the existing pump speed.

((h) This calibration shall consist of (the adjustment, as required, of) (verifying) the 6 ± 1 second simulated thermal power time constant.)

(i) This function is not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed per Specification 3.10.1.

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

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6. The control rod 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 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.

SURVEILLANCE REQUIREMENTS

4.3.6 Each of the above required control rod 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 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 Neutron Flux - 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 Bypass	(2)	(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 BLOCK INSTRUMENTATION

ACTION

ACTION 60 - Declare the RBM inoperable and 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 channelto 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 $\geq$ (30)% 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 (or the reference APRM channel indicates less than (30)% of RATED THERMAL POWER).

b. This function shall be automatically bypassed if detector count rate is > 100 cps or the IRM channels are on range (3) 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 BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale	≤ 0.66 W + (40)%	≤ 0.66 W + (43)%
b. Inoperative	NA	NA
c. Downscale	≥ (5)% of RATED THERMAL POWER	≥ (3)% of RATED THERMAL POWER
2. APRM		
a. Flow Biased Neutron Flux - Upscale	≤ 0.66 W + (42)%*	≤ 0.66 W + (45)%*
b. Inoperative	NA	NA
c. Downscale	≥ (5)% of RATED THERMAL POWER	≥ (3)% of RATED THERMAL POWER
d. Neutron Flux - Upscale, Startup	≤ (12)% of RATED THERMAL POWER	≤ (14)% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ (2×10^5) cps	≤ (5×10^5) cps
c. Inoperative	NA	NA
d. Downscale	≥ (3) cps	≥ (2) cps
4. INTERMEDIATE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ (108/125) divisions of full scale	≤ (110/125) divisions of full scale
c. Inoperative	NA	NA
d. Downscale	≥ (5/125) divisions of full scale	≥ (3/125) divisions of full scale
5. SCRAM DISCHARGE VOLUME		
a. Water Level-High	≤ ( ) inches	≤ ( ) inches
b. Scram Trip Bypass	NA	NA
6. REACTOR COOLANT SYSTEM RECIRCULATION FLOW		
a. Upscale	≤ (108/125) divisions of full scale	≤ (111/125) divisions of full scale
b. Inoperative	NA	NA
c. (Comparator) (Downscale)	≤ (10)% flow deviation	≤ (11)% 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.

TABLE 4.3.6-1

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. ROD BLOCK MONITOR				
a. Upscale	NA	S/U[b][c],M[c]	Q	1*
b. Inoperative	NA	S/U[b][c],M[c]	NA	1*
c. Downscale	NA	S/U[b][c],M[c]	Q	1*
2. APRM				
a. Flow Biased Neutron Flux - 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	NA	2, 5
b. Upscale	NA	S/U[b],W	Q	2, 5
c. Inoperative	NA	S/U[b],W	NA	2, 5
d. Downscale	NA	S/U[b],W	Q	2, 5
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in	NA	S/U[b],W	NA	2, 5
b. Upscale	NA	S/U[b],W	Q	2, 5
c. Inoperative	NA	S/U[b],W	NA	2, 5
d. Downscale	NA	S/U[b],W	Q	2, 5
5. SCRAM DISCHARGE VOLUME				
a. Water Level-High	NA	(M) (Q)	R	1, 2, 5**
b. Scram Trip Bypass	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 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. Includes reactor manual control multiplexing system input.

* With THERMAL POWER $\geq$ (30)% 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.

3/4.9.6 REFUELING PLATFORM

The OPERABILITY requirements ensure that (1) the refueling platform will be used for handling control rods and fuel assemblies within the reactor pressure vessel, (2) each crane and hoist has sufficient load capacity for handling fuel assemblies and control rods, and (3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE POOL

The restriction on movement of loads in excess of the nominal weight of a fuel assembly over other fuel assemblies in the storage pool ensures that in the event this load is dropped 1) the activity release will be limited to that contained in a single fuel assembly, and 2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses.

3/4.9.8 and 3/4.9.9 WATER LEVEL - REACTOR VESSEL and WATER LEVEL -SPENT FUEL STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove (99)% of the assumed (10)% iodine gap activity released from the rupture of an irradiated fuel assembly. This minimum water depth is consistent with the assumptions of the accident analysis.

3/4.9.10 CONTROL ROD REMOVAL

These specifications ensure that maintenance or repair of control rods or control rod drives will be performed under conditions that limit the probability of inadvertent criticality. The requirements for simultaneous removal of more than one control rod are more stringent since the SHUTDOWN MARGIN specification provides for the core to remain subcritical with only one control rod fully withdrawn.

3/4.9.11 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal loop be OPERABLE or that an alternate method capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation ensures that 1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during REFUELING, and 2) sufficient coolant circulation would be available through the reactor core to assure accurate temperature indication and to distribute and prevent stratification of the poison in the event it becomes necessary to actuate the standby liquid control system.

The requirement to have two shutdown cooling mode loops OPERABLE when there is less than (23) feet of water above the reactor vessel flange ensures that a single failure of the operating loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and (23) feet of water above the reactor vessel flange, a large heat sink is available for core cooling. Thus, in the event a failure of the operating RHR loop, adequate time is provided to initiate alternate methods capable of decay heat removal or emergency procedures to cool the core.

3/4.9.10 CONTROL ROD REMOVAL

SINGLE CONTROL ROD REMOVAL

LIMITING CONDITION FOR OPERATION

3.9.10.1 One control rod and/or the associated control rod drive mechanism may be removed from the core and/or reactor pressure vessel provided that at least the following requirements are satisfied until a control rod and associated control rod drive mechanism are reinstalled and the control rod is fully inserted in the core.

a. The reactor mode switch is OPERABLE and locked in the Shutdown position or in the Refuel position per Table 1.2 and Specification 3.9.1.

b. The source range monitors (SRM) are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied, except that the control rod selected to be removed;

   1. May be assumed to be the highest worth control rod required to be assumed to be fully withdrawn by the SHUTDOWN MARGIN test, and

   2. Need not be assumed to be immovable or untrippable.

d. All other control rods in a five-by-five array centered on the control rod being removed are inserted and electrically or hydraulically disarmed or the four fuel assemblies surrounding the control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

e. All other control rods are inserted.

APPLICABILITY: OPERATIONAL CONDITIONS 4 and 5.

ACTION:

With the requirements of the above specification not satisfied, suspend removal of the control rod and/or associated control rod drive mechanism from the core and/or reactor pressure vessel and initiate action to satisfy the above requirements.

SURVEILLANCE REQUIREMENTS

4.9.10.1 Within 4 hours prior to the start of removal of a control rod and/or the associated control rod drive mechanism from the core and/or reactor pressure vessel and at least once per 24 hours thereafter until a control rod and associated control rod drive mechanism are reinstalled and the control rod is inserted in the core, verify that:

a. The reactor mode switch is OPERABLE and locked in the Shutdown position or in the Refuel position with the "one rod out" Refuel position interlock OPERABLE per Specification 3.9.1.

b. The SRM channels are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied per Specification 3.9.10.1.c.

d. All other control rods in a five-by-five array centered on the control rod being removed are inserted and electrically or hydraulically disarmed or the four fuel assemblies surrounding the control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

e. All other control rods are inserted.

MULTIPLE CONTROL ROD REMOVAL

LIMITING CONDITION FOR OPERATION

3.9.10.2 Any number of control rods and/or control rod drive mechanisms may be removed from the core and/or reactor pressure vessel provided that at least the following requirements are satisfied until all control rods and control rod drive mechanisms are reinstalled and all control rods are inserted in the core.

a. The reactor mode switch is OPERABLE and locked in the Shutdown position or in the Refuel position per Specification 3.9.1, except that the Refuel position "one-rod-out" interlock may be bypassed, as required, for those control rods and/or control rod drive mechanisms to be removed, after the fuel assemblies have been removed as specified below.

b. The source range monitors (SRM) are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied.

d. All other control rods are either inserted or have the surrounding four fuel assemblies removed from the core cell.

e. The four fuel assemblies surrounding each control rod or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

APPLICABILITY: OPERATIONAL CONDITION 5.

ACTION:

With the requirements of the above specification not satisfied, suspend removal of control rods and/or control rod drive mechanisms from the core and/or reactor pressure vessel and initiate action to satisfy the above requirements.

SURVEILLANCE REQUIREMENTS

4.9.10.2.1 Within 4 hours prior to the start of removal of control rods and/or control rod drive mechanisms from the core and/or reactor pressure vessel and at least once per 24 hours thereafter until all control rods and control rod drive mechanisms are reinstalled and all control rods are inserted in the core, verify that:

a. The reactor mode switch is OPERABLE and locked in the Shutdown position or in the Refuel position per Specification 3.9.1.

b. The SRM channels are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied.

d. All other control rods are either inserted or have the surrounding four fuel assemblies removed from the core cell.

e. The four fuel assemblies surrounding each control rod and/or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

4.9.10.2.2 Following replacement of all control rods and/or control rod drive mechanisms removed in accordance with this specification, perform a functional test of the "one-rod-out" Refuel position interlock, if this function had been bypassed.

3/4.3.5 REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

The reactor core isolation cooling system actuation instrumentation is provided to initiate actions to assure adequate core cooling in the event of reactor isolation from its primary heat sink and the loss of feedwater flow to the reactor vessel without providing actuation of any of the emergency core cooling equipment.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

The control rod block functions are provided consistent with the requirements of the specifications in Section 3/4.1.4, Control Rod Program Controls and Section 3/4.2 Power Distribution Limits. The trip logic is arranged so that a trip in any one of the inputs will result in a control rod block.

Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

3/4.3.7 MONITORING INSTRUMENTATION

3/4.3.7.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring instrumentation ensures that; (1) the radiation levels are continually measured in the areas served by the individual channels; (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and (3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of (NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980).

3.4.3.7.2 SEISMIC MONITORING INSTRUMENTATION

The OPERABILITY of the seismic monitoring instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the unit. (This instrumentation is consistent with the recommendations of Regulatory Guide 1.12 "Instrumentation for Earthquakes", April 1974.)

3/4.3.7.3 METEOROLOGICAL MONITORING INSTRUMENTATION

The OPERABILITY of the meteorological monitoring instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public. (This instrumentation is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs," February 1972.)

BASES

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

The reactor protection system automatically initiates a reactor scram to:

a. Preserve the integrity of the fuel cladding.

b. Preserve the integrity of the reactor coolant system.

c. Minimize the energy which must be adsorbed following a loss-of-coolant accident, and

d. Prevent inadvertent criticality.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required surveillance.

The reactor protection system is made up of two independent trip systems. There are usually four channels to monitor each parameter with two channels in each trip system. The outputs of the channels in a trip system are combined in a logic so that either channel will trip that trip system. The tripping of both trip systems will produce a reactor scram. The system meets the intent of IEEE-279 for nuclear power plant protection systems. The bases for the trip settings of the RPS are discussed in the bases for Specification 2.2.1.

The measurement of response time at the specified frequencies provides assurance that the protective functions associated with each channel are completed within the time limit assumed in the accident analysis. No credit was taken for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurement, provided such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) inplace, onsite or offsite test measurements, or (2) utilizing replacement sensors with certified response times.