UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 9, 1993

Docket No. 50-410

Mr. B. Ralph Sylvia
Executive Vice President, Nuclear
Niagara Mohawk Power Corporation
301 Plainfield Road
Syracuse, New York 13212

Dear Mr. Sylvia:

SUBJECT: ISSUANCE OF AMENDMENT FOR NINE MILE POINT NUCLEAR STATION, UNIT 2 (TAC NO. M86639)

The Commission has issued the enclosed Amendment No. 51 to Facility Operating License No. NPF-69 for the Nine Mile Point Nuclear Station, Unit 2. The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated May 21, 1993.

The amendment revises TS Table 2.2.1-1, "Reactor Protection System Instrumentation Setpoints," to increase the setpoints for the Average Power Range Monitor (APRM) Flow-Biased Simulated Thermal Power - Upscale scram. The amendment also revises TS 3/4.2.2, "Average Power Range Monitor Setpoints;" TS Table 3.3.6-1, "Control Rod Block Instrumentation;" TS Table 3.3.6-2, "Control Rod Block Instrumentation Setpoints;" TS Table 4.3.6-1, "Control Rod Block Instrumentation Surveillance Requirements;" and TS 6.9.1.9, "Core Operating Limits Report," to delete references to APRM rod block instrumentation. These TS changes are required to facilitate operation in the Extended Load Line Limit region. The amendment also makes a minor editorial correction in parameter 3.a of TS Table 3.3.6-2 and revises TS Bases 3/4.2.2, "APRM Setpoints," to reflect the deletion of references to the APRM rod block instrumentation.

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A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

John E. Menning

John E. Menning, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 51 to NPF-69
2. Safety Evaluation

cc w/enclosures:
See next page

Mr. B. Ralph Sylvia
Niagara Mohawk Power Corporation

Nine Mile Point Nuclear Station
Unit 2

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DATED: November 9, 1993

AMENDMENT NO. 51 TO FACILITY OPERATING LICENSE NO. NPF-69-NINE MILE POINT UNIT 2

Docket File
NRC & Local PDRs
PDI-1 Reading
S. Varga, 14/E/4
J. Calvo, 14/A/4
R. Capra
C. Vogan
J. Menning
OGC
D. Hagan, 3302 MNBB
G. Hill (2), P1-22
C. Grimes, 11/F/23
J. Donoghue, 8/E/23
R. Jones, 8/E/23
ACRS (10)
OPA
OC/LFDCB
PD plant-specific file
C. Cowgill, Region I

cc: Plant Service list

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NIAGARA MOHAWK POWER CORPORATION

DOCKET NO. 50-410

NINE MILE POINT NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 51
License No. NPF-69

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Niagara Mohawk Power Corporation (the licensee) dated May 21, 1993, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter 1;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-69 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 51 are hereby incorporated into this license. Niagara Mohawk Power Corporation shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Capra

Robert A. Capra, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 9, 1993

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 51 TO FACILITY OPERATING LICENSE NO. NPF-69

DOCKET NO. 50-410

Revise Appendix A as follows:

Remove Pages	Insert Pages
2-3	2-3
3/4 2-2	3/4 2-2
3/4 3-60	3/4 3-60
3/4 3-62	3/4 3-62
3/4 3-63	3/4 3-63
3/4 3-64	3/4 3-64
3/4 3-65	3/4 3-65
B3/4 2-1	B3/4 2-1
6-22	6-22

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
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 2) High-Flow-Clamped	≤0.58 (W-ΔW)[a] + 59%, with a maximum of ≤113.5% of RATED THERMAL POWER	≤0.58 (W-ΔW)[a] + 62%, with a maximum of ≤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
3. Reactor Vessel Steam Dome Pressure - High	≤1037 psig	≤1057 psig
4. Reactor Vessel Water Level - Low, Level 3	≥159.3 in. above instrument zero*	≥157.8 in. above instrument zero
5. Main Steam Line Isolation Valve - Closure	≤8% closed	≤12% closed
6. Main Steam Line Radiation[b] - High	≤3.0 x full-power background	≤3.6 x full-power background
7. Drywell Pressure - High	≤1.68 psig	≤1.88 psig

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

(a) The Average Power Range Monitor Scram Function varies as a function of recirculation loop drive flow (W). ΔW is defined as the difference in indicated drive flow (in percent of drive flow which produces rated core flow) between two loop and single loop operation at the same core flow. ΔW=0 for two loop operation. ΔW=5% for single loop operation.

(b) See footnote (**) to Table 3.3.2-2 for trip setpoint during hydrogen addition test.

POWER DISTRIBUTION LIMITS

3/4.2.2 AVERAGE POWER RANGE MONITOR SETPOINTS

LIMITING CONDITIONS FOR OPERATION

3.2.2 The Average Power Range Monitor (APRM) flow-biased simulated thermal power-upscale scram trip setpoint (S) shall be established according to the relationship specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With the APRM flow-biased simulated thermal power-upscale scram trip setpoint less conservative than the value shown in the Allowable Value column for S, as above determined, initiate corrective action within 15 minutes and adjust S to be consistent with the Trip Setpoint value* within 6 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.2 The FRACTION OF RATED THERMAL POWER (FRTP) and the CORE MAXIMUM FRACTION OF LIMITING POWER DENSITY (CMFLPD) shall be determined, the value of T** calculated, and the most recent actual APRM flow-biased simulated thermal power-upscale scram trip setpoint verified to be within the above limit or adjusted, as required:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with CMFLPD greater than or equal to FRTP.

d. The provisions of Specification 4.0.4 are not applicable.

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* With CMFLPD greater than the FRTP rather than adjusting the APRM setpoints, the APRM gain may be adjusted so that APRM readings are greater than or equal to 100% times CMFLPD provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER and a notice of adjustment is posted on the reactor control panel.

** Definition of T is specified in the CORE OPERATING LIMITS REPORT.

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. Source Range Monitor			
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(f)	5	61
3. Intermediate Range Monitor			
a. Detector Not Full In	6	2, 5	61
b. Upscale	6	2, 5	61
c. Inoperative	6	2, 5	61
d. Downscale(e)	6	2, 5	61
4. Scram Discharge Volume			
Water Level - High, Float Switch	2	1, 2, 5**	62
5. Reactor Coolant System Recirculation Flow			
a. Upscale	2	1	62
b. Inoperative	2	1	62
c. Comparator	2	1	62
6. Reactor Mode Switch			
a. Shutdown Mode	2	3, 4	62
b. Refuel Mode	2	5	62

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION			TRIP SETPOINT	ALLOWABLE VALUE
1.	Rod Block Monitor			
	a.	Upscale	*	*
	b.	Inoperative	NA	NA
	c.	Downscale	≥5% of RATED THERMAL POWER	≥3% of RATED THERMAL POWER
2.	Source Range Monitor			
	a.	Detector Not Full In	NA	NA
	b.	Upscale	≤1 x 10^5 cps	≤1.6 x 10^5 cps
	c.	Inoperative	NA	NA
	d.	Downscale	≥3 cps**	≥1.8 cps**
3.	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
4.	Scram Discharge Volume			
	Water Level - High, Float Switch		≤16.5 in.	≤39.75 in.

Table 3.3.6-2 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION			TRIP SETPOINT	ALLOWABLE VALUE
5.	Reactor Coolant System Recirculation Flow			
	a.	Upscale	≤108% rated flow	≤111% rated flow
	b.	Inoperative	NA	NA
	c.	Comparator	≤10% flow deviation	≤11% flow deviation
6.	Reactor Mode Switch			
	a.	Shutdown Mode	NA	NA
	b.	Refuel Mode	NA	NA

* Specified in the CORE OPERATING LIMITS REPORT

** For fuel loading and startup from refueling the count rate may be less than 3 cps if the following conditions are met: the signal to noise ratio is greater than or equal to 5, and the signal is greater than 1.3 cps.

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), Q(c)	Q	1*
b. Inoperative	NA	S/U(b)(c), Q(c)	NA	1*
c. Downscale	NA	S/U(b)(c), Q(c)	Q	1*
2. 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
3. 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
4. Scram Discharge Volume				
Water Level - High, Float Switch	NA	Q	R	1, 2, 5**

TABLE 4.3.6-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION		CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
5.	Reactor Coolant System Recirculation Flow				
	a. Upscale	NA	S/U(b), Q	Q	1
	b. Inoperative	NA	S/U(b), Q	NA	1
	c. Comparator	NA	S/U(b), Q	Q	1
6.	Reactor Mode Switch				
	a. Shutdown Mode	NA	R	NA	3, 4
	b. Refuel Mode	NA	R	NA	5

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section assure that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10 CFR 50.46.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod-to-rod power distribution within an assembly. The peak clad temperature is calculated assuming an LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times 1.02 is used in the heatup code along with the exposure-dependent steady-state gap conductance and rod-to-rod local peaking factor. The limiting value for APLHGR is specified in the CORE OPERATING LIMITS REPORT for two-recirculation-loop operation.

The calculational procedure used to establish the APLHGR specified in the CORE OPERATING LIMITS REPORT is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10 CFR 50. A complete discussion of each code employed in the analysis is presented in Reference 1.

A list of the significant plant input parameters to the loss-of-coolant accident analysis is presented in Bases Table B3.2.1-1.

For plant operations with single recirculation loop the MAPLHGR limits are specified in the CORE OPERATING LIMITS REPORT. The constant factor is derived from LOCA analyses initiated from single loop operation to account for earlier boiling transition at the limiting fuel node compared to the standard LOCA evaluations.

3/4.2.2 APRM SETPOINTS

The fuel cladding integrity Safety Limits of Specification 2.1 were based on a power distribution which would yield the design LHGR at RATED THERMAL POWER. The flow-biased simulated thermal power-upscale scram setting of the APRM instruments must be adjusted to ensure that the MCPR does not become less than the fuel cladding integrity safety limit or that greater than or equal to 1% plastic strain does not occur in the degraded situation. The scram setpoint is adjusted in accordance with the formula in this specification when the combination of THERMAL POWER and CMFLPD indicates a higher peaked power distribution to ensure that an LHGR transient would not be increased in the degraded condition.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

6.9.1.8 (Continued)

The Semiannual Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the OFFSITE DOSE CALCULATION MANUAL (ODCM), pursuant to Specifications 6.13 and 6.14, respectively, as well as any major change to liquid, gaseous, or solid radwaste treatment systems pursuant to Specification 6.15. It shall also include a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2.

The Semiannual Radioactive Effluent Release Reports shall also include the following: an explanation of why the inoperability of liquid or gaseous effluent monitoring instrumentation was not corrected within the time specified in Specification 3.3.7.9 or 3.3.7.10, respectively, and a description of the events leading to liquid holdup tanks exceeding the limits of Specification 3.11.1.4.

CORE OPERATING LIMITS REPORT

6.9.1.9

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle for the following:

    1) The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for Specification 3.2.1.

    2) The Average Power Range Monitor (APRM) flow-biased simulated thermal power-upscale scram trip setpoint for Specification 3.2.2.

    3) The K_f core flow adjustment factor for Specification 3.2.3.

    4) The MINIMUM CRITICAL POWER RATIO (MCPR) for Specification 3.2.3.

    5) The LINEAR HEAT GENERATION RATE (LHGR) for Specification 3.2.4.

    6) Control Rod Block Instrumentation Setpoint for the rod block monitor upscale trip setpoint and allowable value for Specification 3.3.6.

    and shall be documented in the CORE OPERATING LIMITS REPORT.

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following document.

    1) General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE-20566 latest approved revision.

UNITED STATES
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WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 51 TO FACILITY OPERATING LICENSE NO. NPF-69

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR STATION, UNIT 2

DOCKET NO. 50-410

1.0 INTRODUCTION

By letter dated May 21, 1993 (Reference 1), Niagara Mohawk Power Corporation (the licensee) submitted a request for changes to the Nine Mile Point Nuclear Station, Unit 2 (NMP2), Technical Specifications (TS). The requested changes would (1) revise the slope of the line on the power-to-flow map representing the average power range monitor (APRM) flow-biased simulated thermal power scram setpoint, and (2) relocate references to the APRM rod block instrumentation and setpoints from the TS to the updated safety analysis report (USAR).

The APRM scram setpoint modifications are requested to remove power to flow restrictions imposed at low flows where the APRM flow-biased scram limit encroaches on the extended load line limit (ELLLA) region. Analyses supporting operation in the ELLLA region appear in Appendix 15G of the NMP2 USAR, as well as in Appendix A of the USAR, "Reload Analysis," which presents cycle specific analyses.

2.0 EVALUATION

2.1 APRM Flow-Biased Simulated Thermal Power Upscale Scram Setpoint

The APRM simulated thermal power (STP) scram function furnishes protection against exceeding thermal limits during slow transients. In those transients considered 'fast,' neutron flux leads thermal flux due to the fuel time constant, so setting a relatively high fixed neutron flux trip is acceptable since the fuel will not heat up sufficiently to challenge thermal margins. This APRM fixed neutron flux trip would need to be lowered to respond adequately to slow transients. In these slow cases, the neutron and thermal fluxes are matched, so allowing the neutron flux to rise to the fixed flux level could lead to thermal limit violations.

To provide protection for the slow transients, the APRM signal is modified using a time delay which simulates the fuel time constant. The STP signal is compared with a flow-biased reference that decreases approximately parallel to the flow control lines of the power-to-flow map.

The current APRM STP trip setpoint uses an equation [0.66(W-ΔW) + 51%], with a maximum value of 113.5%. The high value, or clamp, is set below the APRM fixed neutron flux trip. The proposed change uses a new equation to establish the STP setpoint, [0.58(W-ΔW) + 59%], but maintains the maximum clamped value of 113.5%.

The Bases for NMP2 TS 3/4.2.2 state that the APRM flow biased STP upscale scram setpoint is adjusted to ensure that the minimum critical power ratio (MCPR) does not decrease to less than the fuel cladding integrity safety limit or that greater than or equal to 1% plastic strain does not occur during degraded plant conditions. Transient analyses discussed in NMP2 USAR Appendix A, "Reload Analysis" use a fixed value of 117% as the APRM STP trip setpoint, demonstrating that the fuel safety requirements are maintained even with the new setpoint. An example is the loss of feedwater heating event (manual flow control case) which depends on the APRM STP scram for protection. The change of critical power ratio listed in Table A.15.0-1 of the USAR remains at 0.11. This and the other analyses remain conservative with the new setpoint, since the actual APRM STP trip would occur at a lower power setpoint than used in the analyses.

The Bases for NMP2 TS 2.2.1 explains that the APRM setpoints were selected to provide adequate margin for the safety limits while allowing operating margin from unnecessary shutdowns. Further, the difference between the setpoint and allowable value accounts for instrument accuracy and calibration capability. The proposed change serves part of the stated objective, avoiding unnecessary shutdowns, by furnishing greater margin between the operating envelope and the setpoint at lower flows. The margin between the allowable value and trip point is maintained with this change. Thus, the only real alteration is to the margin provided between the setpoint and the analysis point of 117%. Examination of the difference in the available margin shows that safety margins are not unduly reduced. At 40% flow, the setpoint change (from 77.4% power to 82.2%) results in a reduction of approximately 5% to the available margin. This maintains a nearly 35% power margin between the analysis and actual setpoints. At higher flows the margin is smaller, but the change is smaller as well. At no value of flow is the difference between the actual setpoint and the analysis value ever less than the original difference of 3.5% at the clamped value.

The ELLLA analyses contained in NMP2 USAR, Appendix 15G considered the impact of limiting oscillations based upon published General Electric guidance as well as institution of requirements set forth in NRC Bulletin 88-07 (Reference 2). The setpoint change does not alter these measures.

Thus, the proposed setpoint modification adequately accounts for the required margin of safety as shown by analyses and satisfies the considerations set forth in the TS Bases. The change to the APRM STP flow biased scram setpoint is acceptable.

2.2 APRM Flow-Biased Neutron Flux Upscale Rod Block Instrumentation System

The basis for the APRM flow-biased neutron flux upscale rod block setpoints are discussed along with the scram setpoint in the Bases for NMP2 TS 2.2.1 in terms of protecting against violation of the safety limit. However, safety analyses for NMP2 take no credit for the operation of the APRM rod block instrumentation. Although in certain situations the rod block would prevent the need for a protective action, it is not necessary to ensure that safety limits are not violated. For instance, the rod block monitor (RBM), not the APRM rod block, provides a safety function for a control rod withdrawal error. The APRM rod block acts only as a backup function.

Removal of references to the APRM rod block instrumentation is consistent with the criteria detailed in the Commission's Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors (Reference 3), which states that design or operation constraints should satisfy four criteria in order to be located in the TS: (1) Installed instrumentation that is used to detect, and indicate in the control room a significant abnormal degradation of the reactor coolant pressure boundary. (2) A process variable, design feature or operating restriction that is the initial condition of a design basis accident or transient analysis that either assumes failure of or presents a challenge to the integrity of a fission product barrier. (3) A structure, system or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier. (4) A structure, system or component which operating experience or probabalistic safety assessment has shown to be significant to public health and safety.

The application provided justification for the conclusion that the APRM rod block instrumentation for NMP2 did not meet the first three criteria. The analyses of the USAR show that the last criterion is not met since the safety analyses do not rely on the system, therefore, it is not needed to meet safety margins and furnish protection.

Thus, the relocation of the APRM rod block instrumentation to the USAR will permit design changes in accordance with 10 CFR 50.59 and the safety functions will be adequately controlled by the regulatory requirements that apply to the design control process. Further, the change is consistent with the Commission's Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors.

3.0 TECHNICAL SPECIFICATIONS

The following changes to the NMP2 TS are proposed in the application:

a. TS Table 2.2.1-1, "Reactor Protection System Instrumentation Setpoints," is revised to include the new flow biased simulated thermal power upscale setpoint equation.

b. TS 3/4.2.2, "Average Power Range Monitor Setpoints," and associated Bases are revised to remove references to the APRM flow-biased neutron flux upscale control rod block trip setpoint.

c. TS Tables 3.3.6-1, 3.3.6-2 and 4.3.6-1 of TS 3/4.3.6, "Control Rod Block Instrumentation" are modified, removing parameters associated with the APRM flow-biased neutron flux upscale control rod block setpoints. An editorial change is included for TS Table 3.3.6-2.

d. TS 6.9.1.9, "Core Operating Limits Report" references to the APRM flow-biased neutron flux upscale control rod block trip are deleted.

As discussed in the previous section, these changes are considered acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The amendment also relates to changes in recordkeeping, reporting, or administrative procedures or requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (58 FR 34080). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (10). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

7.0 REFERENCES

1. Letter (NMP2L 1388) from B. Ralph Sylvia (NMPC) to USNRC dated May 21, 1993 transmitting an Application for Amendment to Nine Mile Point Unit 2 Operating License.

2. NRC Bulletin 88-07, Supplement 1, "Power Oscillations in Boiling Water Reactors (BWRs)," December 30, 1988.

3. USNRC, Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors, 58 FR 39132, July 22, 1993.

Principal Contributor:
J. Donoghue

Date: November 9, 1993

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

Original signed by:

John E. Menning, Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 51 to NPF-69
2. Safety Evaluation

cc w/enclosures:
See next page

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P.3 SER