April 5, 1990

Docket No. 50-410

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Mr. Lawrence Burkhardt III
Executive Vice President, Nuclear Operations
Niagara Mohawk Power Corporation
301 Plainfield Road
Syracuse, New York 13212

Dear Mr. Burkhardt:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. 69094)

The Commission has issued the enclosed Amendment No. 13 to Facility Operating License No. NPF-69 for the Nine Mile Point Nuclear Station Unit No. 2 (NMP-2). The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated August 3, 1988.

This amendment revises Sections 3/4.3.3, 3/4.3.5 and the associated Bases to change the Nominal Trip Setpoints and Allowable Values pertaining to High Pressure Core Spray and Reactor Core Isolation Cooling pump suction transfer.

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

Sincerely,

Original signed by

Robert E. Martin, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 13 to NPF-69
2. Safety Evaluation

cc: w/enclosures
See next page

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

April 5, 1990

Docket No. 50-410

Mr. Lawrence Burkhardt III
Executive Vice President, Nuclear Operations
Niagara Mohawk Power Corporation
301 Plainfield Road
Syracuse, New York 13212

Dear Mr. Burkhardt:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. 69094)

The Commission has issued the enclosed Amendment No. 13 to Facility Operating License No. NPF-69 for the Nine Mile Point Nuclear Station Unit No. 2 (NMP-2). The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated August 3, 1988.

This amendment revises Sections 3/4.3.3, 3/4.3.5 and the associated Bases to change the Nominal Trip Setpoints and Allowable Values pertaining to High Pressure Core Spray and Reactor Core Isolation Cooling pump suction transfer.

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

Sincerely,

Robert Martin
Robert E. Martin, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 13 to NPF-69
2. Safety Evaluation

cc: w/enclosures
See next page

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

NIAGARA MOHAWK POWER CORPORATION

DOCKET NO. 50-410

NINE MILE POINT NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 13
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 August 3, 1988, 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 I;

   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. 13 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Acting for 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: April 5, 1990

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 13 TO FACILITY OPERATING LICENSE NO. NPF-69

DOCKET NO. 50-410

Revise Appendix A as follows:

Remove Pages	Insert Pages
3/4 3-32	3/4 3-32
3/4 3-37	3/4 3-37
3/4 3-38	3/4 3-38
3/4 3-42	3/4 3-42
3/4 3-55	3/4 3-55
3/4 3-57	3/4 3-57
3/4 3-58	3/4 3-58
B3/4 3-2	B3/4 3-2
B3/4 3-4	B3/4 3-2

TABLE 3.3.3-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP FUNCTION(a)	APPLICABLE OPERATIONAL CONDITIONS	ACTION
C. Division III Trip System			
1. HPCS SYSTEM			
a. Reactor Vessel Water Level - Low, Low, Level 2	4(b)	1, 2, 3, 4*, 5*	36
b. Drywell Pressure - High (d)	4(b)	1, 2, 3	36
c. Reactor Vessel Water Level - High, Level 8	4(e)	1, 2, 3, 4*, 5*	32
d. Pump Suction Pressure - Low (Transfer)	2(f)	1, 2, 3, 4*, 5*	37
e. Suppression Pool Water Level - High	2(f)	1, 2, 3, 4*, 5*	37
f. HPCS System Flow Rate - Low (Bypass)	1	1, 2, 3, 4*, 5*	31
g. Pump Discharge Pressure - High (Bypass)	1	1, 2, 3, 4*, 5*	31
h. Manual Initiation (d)	1/System	1, 2, 3, 4*, 5*	35

	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE OPERATIONAL CONDITIONS	ACTION
D. Loss of Power (Divisions I & II)					
1. 4.16-kV Emergency Bus Under-voltage - Loss of Voltage	3/Bus	2/Bus	2/Bus	1, 2, 3, 4**, 5**	39
2. 4.16-kV Emergency Bus Under-voltage - Degraded Voltage	3/Bus	2/Bus	2/Bus	1, 2, 3, 4**, 5**	39
E. Loss of Power, Division III					
1. 4.16-kV Emergency Bus Under-voltage - Loss of Voltage	3/Bus	2/Bus	2/Bus	1, 2, 3, 4**, 5**	39
2. 4.16-kV Emergency Bus Under-voltage - Degraded Voltage	3/Bus	2/Bus	2/Bus	1, 2, 3, 4**, 5**	39

TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
C. Division III Trip System		
1. HPCS SYSTEM		
a. Reactor Vessel Water Level - Low, Low, Level 2	≥108.8 in.*	≥101.8 in.
b. Drywell Pressure - High	≤1.68 psig	≤1.88 psig
c. Reactor Vessel Water Level - High, Level 8	≤202.3 in.*	≤209.3 in.
d. Pump Suction Pressure - Low (Transfer)	≥97 in. H_2O	≥94.5 in. H_2O
e. Suppression Pool Water Level - High	≤201.0 ft. el	≤201.1 ft. el
f. HPCS System Flow Rate - Low (Bypass)	≥825 gpm	≥750 gpm
g. Pump Discharge Pressure - High (Bypass)	≥240 psig	≥220 psig
h. Manual Initiation	NA	NA
D. Loss of Power (Divisions I & II)		
1. 4.16-kV Emergency Bus Under-voltage - Loss of Voltage	a. 4.16-kV basis - ≥3148	≥3051 volts
	b. ≤3.06-sec time delay	≤3.12-sec time delay
2. 4.16-kV Emergency Bus Under-voltage - Degraded Voltage	a. 4.16-kV basis - ≥3847 volts	≥3770 volts
	b. ≤8.16-sec time delay**	≤8.32-sec time delay **
	c. ≤30.6-sec time delay	≤31.2-sec time delay

TABLE 3.3.3-2 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
E. Loss of Power (Division III)		
1. 4.16-kV Emergency Bus Undervoltage - Loss of Voltage	a. 4.16-kV basis - ≥3148 volts	≥3051 volts
	b. ≤3.06-sec time delay	≤3.12-sec time delay
2. 4.16-kV Emergency Bus Undervoltage - Degraded Voltage	a. 4.16-kV basis - ≥3847 volts	≥3770 volts
	b. ≤12.24-sec time delay	≤12.48-sec time delay

* See Bases Figure B3/4 3-1.

** Alarm only without LOCA signal present; Alarm and trip with LOCA signal present.

TABLE 4.3.3.1-1 (Continued)

EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE IS REQUIRED
B. Division II Trip System (Continued)				
2. Automatic Depressurization System Trip System "B"** (Continued)				
a. Reactor Vessel Water Level - Low, Low, Low, Level 1	S	M	R(c)	1, 2, 3
b. ADS Timer	NA	M	Q	1, 2, 3
c. Reactor Vessel Water Level - Low, Level 3 (Permissive)	S	M	R(c)	1, 2, 3
d. LPCI Pump (B and C) Discharge Pressure - High (Permissive)	S	M	R(c)	1, 2, 3
e. Manual Inhibit	NA	M	NA	1, 2, 3
f. Manual Initiation	NA	M(a)	NA	1, 2, 3
C. Division III Trip System				
1. HPCS System				
a. Reactor Vessel Water Level - Low, Low, Level 2	S	M	R(c)	1, 2, 3, 4*, 5*
b. Drywell Pressure - High(b)	S	M	R(c)	1, 2, 3
c. Reactor Vessel Water Level - High, Level 8	S	M	R(c)	1, 2, 3, 4*, 5*
d. Pump Suction Pressure - Low (Transfer)	S	M	R(c)	1, 2, 3, 4*, 5*
e. Suppression Pool Water Level - High	S	M	R(c)	1, 2, 3, 4*, 5*
f. HPCS System Flow Rate - Low (Bypass)	S	M	R(c)	1, 2, 3, 4*, 5*
g. Pump Discharge Pressure - High (Bypass)	S	M	R(c)	1, 2, 3, 4*, 5*
h. Manual Initiation(b)	NA	M(a)	NA	1, 2, 3, 4*, 5*

TABLE 3.3.5-1

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION

FUNCTIONAL UNITS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM(a)	ACTION
1. Reactor Vessel Water Level - Low, Low, Level 2	2	50
2. Reactor Vessel Water Level - High, Level 8(b)	2	50
3. Pump Suction Pressure - Low (Transfer)	2(c)	51
4. Manual Initiation(d)	1/system(e)	52

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 other OPERABLE channel in the same Trip System is monitoring that parameter.

(b) The RCIC Level 8 trip may be bypassed to perform RCIC 150 psig operational surveillance test in accordance with Specification 4.7.4.c.2.

(c) One Trip System with one-out-of-two logic.

(d) Manual initiation is not required to be OPERABLE with indicated reactor vessel water level on the wide-range instrument greater than the Level 8 setpoint coincident with the vessel pressure less than 600 psig due to the hot calibration/cold operation level error.

(e) One Trip System with one channel.

Amendment No. 13

TABLE 3.3.5-2

REACTOR CORE ISOLATION COOLING SYSTEM ACTUATION INSTRUMENTATION SETPOINTS

FUNCTIONAL UNITS	TRIP SETPOINT	ALLOWABLE VALUE
1. Reactor Vessel Water Level - Low, Low, Level 2	$\geq$108.8 in.*	$\geq$101.8 in.
2. Reactor Vessel Water Level - High, Level 8	$\leq$202.3 in.*	$\leq$209.3 in.
3. Pump Suction Pressure - Low (Transfer)	$\geq$102 in. H_2O	$\geq$101 in. H_2O
4. Manual Initiation	NA	NA

* See Bases Figure B3/4 3-1.

Amendment No. 13

TABLE 4.3.5.1-1

REACTOR CORE ISOLATION COOLING SYSTEM

ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNITS	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Reactor Vessel Water Level - Low, Low, Level 2	S	M	R*
2. Reactor Vessel Water Level - High, Level 8	S	M	R*
3. Pump Suction Pressure - Low (Transfer)	S	M	R*
4. Manual Initiation **	NA	M+	NA

---

* Perform the calibration procedure for the trip unit setpoint at least once per 31 days.

** Manual initiation is not required to be OPERABLE with indicated reactor vessel water level on the wide range instrument greater than Level 8 setpoint coincident with the vessel pressure less than 600 psig because of the hot calibration/cold operation level error.

+ Manual initiation switches shall be tested at least once per 18 months during shutdown. All other circuitry associated with manual initiation shall receive a CHANNEL FUNCTIONAL TEST at least once per 31 days as part of circuitry required to be tested for automatic system actuation.

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION (Continued)

high or low end of the setting has a direct bearing on safety, are established at a level away from the normal operating range to prevent inadvertent actuation of the systems involved.

Except for the MSIVs, the FSAR Chapter 15 safety analysis does not address individual sensor response times or the response times of the logic systems to which the sensors are connected. For AC-operated valves, it is assumed that the AC power supply is lost and is restored by startup of the emergency diesel generators. In this event, a time of 13 seconds is assumed before the valve starts to move. In addition to the pipe break, the failure of the DC-operated valve is assumed; thus the signal delay (sensor response) is concurrent with the 13-second diesel startup. The safety analysis considers an allowable inventory loss in each case which in turn determines the valve speed in conjunction with the 13-second delay. It follows that checking the valve speeds and the 13-second time for establishing emergency power will establish the response time for the isolation functions.

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 an allowance for instrument drift specifically allocated for each trip in the safety analysis. The Trip Setpoint and Allowable Value also contain additional margin for instrument accuracy and calibration capability.

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

The emergency core cooling system actuation instrumentation is provided to initiate actions to mitigate the consequences of accidents that are beyond the ability of the operator to control. This specification provides the OPERABILITY requirements, Trip Setpoints, and response times that will ensure effectiveness of the systems to provide the design protection. Although the instruments are listed by system, in some cases the same instrument may be used to send the actuation signal to more than one system at the same time.

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 an allowance for instrument drift specifically allocated for each trip in the safety analysis. The Trip Setpoint and Allowable Value also contain additional margin for instrument accuracy and calibration capability.

The HPCS pump suction pressure-low represents an analytical transfer level in the condensate storage tank of 14 feet at maximum flow and 3.0 feet at minimum flow. This is above the corresponding minimum tank level of 10.2 feet at maximum flow and 2.9 feet at minimum flow required to prevent vortexing.

3/4.3.4 RECIRCULATION PUMP TRIP ACTUATION INSTRUMENTATION (Continued)

between each Trip Setpoint and the Allowable Value is an allowance for instrument drift specifically allocated for each trip in the safety analyses. The Trip Setpoint and Allowable Value also contain additional margin for instrument accuracy and calibration capability.

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.

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 an allowance for instrument drift specifically allocated for each trip in the safety analyses. The Trip Setpoint and Allowable Value also contain additional margin for instrument accuracy and calibration capability.

The RCIC pump suction pressure-low represents an analytical transfer level in the condensate storage tank of 13.1 feet at maximum flow and 2.53 feet at minimum flow. This is above the corresponding minimum tank level of 5.0 feet at maximum flow and 2.5 feet at minimum flow required to prevent vortexing.

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 an allowance for instrument drift specifically allocated for each trip in the safety analyses. The Trip Setpoint and Allowable Value also contain additional margin for instrument accuracy and calibration capability. The scram discharge volume water level-high setpoint is referenced to a scram discharge volume instrument zero level at elevation 263 feet 10 inches.

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 10 CFR 50, Appendix A, General Design Criteria (GDC) 19, 41, 60, 61, 63 and 64.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 13 TO FACILITY OPERATING LICENSE NO. NPF-69

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR POWER STATION, UNIT NO. 2

DOCKET NO. 50-410

INTRODUCTION

By letter dated August 3, 1988, Niagara Mohawk Power Corporation, the licensee, proposed changes to Nine Mile Point Unit 2 Technical Specifications and associated Bases to revise the Nominal Trip Setpoints and Allowable Values pertaining to High Pressure Core Spray (HPCS) and Reactor Core Isolation Cooling (RCIC) pump suction transfer. In the existing Technical Specification calculations, velocity head was not a parametric consideration in pump suction transfer, and hence the existing calculations do not accurately reflect the levels at which transfer should occur.

The licensee has checked and corrected these calculations by introducing the velocity head consideration. The actual transfer level in the tank at a given pressure switch setpoint will vary as a function of flow rate due to the effects of resistance and velocity head losses in the suction piping. In order to provide a parameter with a constant value and to reduce potential confusion over what the CST level in the Technical Specification actually represents, the setpoints for the pressure switches will be specified in the Technical Specification in lieu of tank level.

EVALUATION

Pump Suction Transfer Indication

The licensee proposes to change the existing indication at which HPCS and RCIC pump suction transfers from the condensate storage tank to the suppression pool.

Instead of specifying the tank level at which transfer occurs, the licensee intends to reflect the setpoint of the pressure switch which controls pump suction. The pressure switches are presently located on the safety related portion of the HPCS and RCIC pump suction lines inside of secondary containment. There are no hardware changes involved in the pressure switches modification; only the setpoints will be changed.

Changing the transfer indication of the pressure switches from level to pressure, increases the licensee's flexibility with regard to switchover. During low flow conditions, the licensee may maximize use of the available water volume in the CST by delaying transfer to the suppression pool. The licensee has stated that the proposed change is also intended to promote greater operator convenience. These changes are acceptable.

High Pressure Core Spray (HPCS) and Reactor Coolant Isolation System (RCIC) Switchover Setpoint

The revised calculations account for the velocity head component and replace condensate tank level with the corresponding actuation pressure for both high and low flow conditions. The derived analytical limits are above the required corresponding minimum submergence level in order to prevent vortexing at both high and low flow conditions. Instrument and calibration accuracies and drift have been allowed for.

The actuation pressure associated with the new pressure setpoint remains constant at all flow conditions while the tank level varies. This provides a better means of addressing suction transfer of the HPCS and RCIC systems than does specifying a tank level because it establishes a parameter with a constant value. Incorporating the actuation pressure in the pump suction lines will result in a Technical Specification that is easier to implement and to understand; the potential confusion over what the tank level in the Technical Specifications actually represents will be eliminated. The Allowable Values and Nominal Trip Setpoints proposed in the amendment should assure actuation occurs above the minimum submergence levels under all flow conditions. Therefore, we find this change to be acceptable.

Technical Specification Changes

The following are the Technical Specification changes associated with the proposal:

(1) Table 3.3.3-1/ Item C.1.d

Replace CST level Indication with Pump Suction Pressure Transfer Indication.

(2) Table 3.3.3-2

Replace CST level Trip Setpoint and Allowable Value with Pump Suction Pressure Transfer Trip Setpoint and Allowable Value.

(3) Table 4.3.3.1-1/Item C.1.d

Replace CST level Indication with Pump Suction Pressure Transfer Indication

(4) Table 3.3.5-1

Replace CST level Indication with Pump Suction Transfer Indication

(5) Table 3.3.5-2

Replace CST level Trip Setpoint and Allowable Value with Pump Suction Pressure Transfer Trip Setpoints and Allowable Value.

(6) Table 4.3.5.1-1

Replace CST level Indication with Pump Suction Pressure Transfer Indication

The above Technical Specification changes are consistent with the licensee's supporting analyses and are, therefore, acceptable.

ENVIRONMENTAL CONSIDERATION

This amendment involves a change in a requirement with respect to the installation or use of the facility components located within the restricted areas as defined in 10 CFR 20 and changes surveillance requirements. The staff has determined that this 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

CONCLUSION

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: April 5, 1990

PRINCIPAL CONTRIBUTOR:
A. Massey