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.

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.