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Docket No. 50-220

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Mr. Donald P. Dise
Vice President - Engineering
Niagara Mohawk Power Corporation
300 Erie Boulevard West
Syracuse, New York 13202

Dear Mr. Dise:

The Commission has issued the enclosed Amendment No. 37 to Facility Operating License No. DPR-63 for the Nine Mile Point Nuclear Station, Unit No. 1. The amendment consists of changes to the Technical Specifications as requested by your letter of February 15, 1980, as supplemented by letters of March 27, 1979 and April 3, 1979.

The amendment revises the Technical Specifications to reflect the installation of an Analog Transmitter Trip Unit System (ATTUS). Prior to your application the staff had reviewed and approved the General Electric Topical Reports NEDO-21617 and NEDO-21617-1, "Analog Transmitter/Trip Unit System." Therefore, our present review covered those site specific design areas as related to the Nine Mile installation.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

/s/

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 37 to License No. DPR-63
2. Safety Evaluation
3. Notice

cc w/encls:
See next page

PSYB D. Tondi	RS HVM	EB
RScholl	HVanderMolen	ALee
4/16/80	4/21/80	4/22/80

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SURNAME	SNorris	PPolk:ms	TAIppolito	WPGammill for	BMBordenick	
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As to Amend format only

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

NIAGARA MOHAWK POWER CORPORATION

DOCKET NO. 50-220

NINE MILE POINT NUCLEAR STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 37
License No. DPR-63

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

   A. The application for amendment by Niagara Mohawk Power Corporation (the licensee) dated February 15, 1979, as supplemented March 27 and April 3, 1979, 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. DPR-63 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 37 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 2, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 37

FACILITY OPERATING LICENSE NO. DPR-63

DOCKET NO. 50-220

Revise Appendix A by removing the following pages and replacing with revised identically numbered pages. Marginal lines indicate area of change.

6
13
188
192-201
203-215
231
232
232a
233
237a

Add page 212a

SAFETY LIMIT

c. The neutron flux shall not exceed its scram setting for longer than 1.5 seconds as indicated by the process computer. When the process computer is out of service, a safety limit violation shall be assumed if the neutron flux exceeds the scram setting and control rod scram does not occur.

To ensure that the Safety Limit established in Specifications 2.1.1a and 2.1.1b is not exceeded, each required scram shall be initiated by its expected scram signal. The Safety Limit shall be assumed to be exceeded when scram is accomplished by a means other than the expected scram signal.

d. Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be more than 7 feet 11 inches (3.88 inches indicator scale) below minimum normal water level (Elevation 302'9"), except as specified "e" below.

e. For the purpose of performing major maintenance (not to exceed 12 weeks in duration) on the reactor vessel; the reactor water level may be lowered 9' below the minimum normal water level (Elevation 302'9"). Whenever the reactor water level is to be lowered below the low-low-low level set point redundant instrumentation will be provided to monitor the reactor water level.

LIMITING SAFETY SYSTEM SETTING

d. The reactor water low level scram trip setting shall be no lower than -12 inches (53 inches indicator scale) relative to the minimum normal water level (302'9").

e. The reactor water low-low level setting for core spray initiation shall be no less than -5 feet (5 inches indicator scale) relative to the minimum normal water level (Elevation 302'9").

f. The flow biased APRM rod block trip settings shall be less than or equal to that shown in Figure 2.1.1.

During periods when the reactor is shut down, consideration must also be given to water level requirements, due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. The core will be cooled sufficiently to prevent clad melting should the water level be reduced to two-thirds of the core height.

The lowest point at which the water level can normally be monitored is approximately 4 feet 8 inches above the top of the active fuel. This is the low-low-low water level trip point, which is 7 feet 11 inches (3.88 inches indicator scale) below minimum normal water level (Elevation 302'9"). The safety limit has been established here to provide a point which can be monitored and also can provide adequate margin. However, for performing major maintenance as specified in Specification 2.1.1.e, redundant instrumentation will be provided for monitoring reactor water level below the low-low-low water level set point. (For example, by installing temporary instrument lines and reference pots to redundant level transmitters, so that the reactor water level may be monitored over the required range.) In addition written procedures, which identify all the valves which have the potential of lowering the water level inadvertently, are established to prevent their operation during the major maintenance which requires the water level to be below the low-low level set point.

The thermal power transient resulting when a scram is accomplished other than by the expected scram signal (e.g., scram from neutron flux following closure of the main turbine stop valves) does not necessarily cause fuel damage. However, for this specification a safety limit violation will be assumed when a scram is only accomplished by means of a backup feature of the plant design. The concept of not approaching a safety limit provided scram signals are operable is supported by the extensive plant safety analysis.

LIMITING CONDITION FOR OPERATION

3.6.2 PROTECTIVE INSTRUMENTATION

Applicability:

Applies to the operability of the plant instrumentation that performs a safety function.

Objective:

To assure the operability of the instrumentation required for safe operation.

Specification:

a. The set points, minimum number of trip systems, and minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as given in Tables 3.6.2a to 3.6.2k.

   If the requirements of a table are not met, the actions listed below for the respective type of instrumentation shall be taken.

   (1) Instrumentation that initiates scram - control rods shall be inserted, unless there is no fuel in the reactor vessel.

SURVEILLANCE REQUIREMENT

4.6.2 PROTECTIVE INSTRUMENTATION

Applicability:

Applies to the surveillance of the instrumentation that performs a safety function.

Objective:

To verify the operability of protective instrumentation.

Specification:

a. Sensors and instrument channels shall be checked, tested and calibrated at least as frequently as listed in Tables 4.6.2a to 4.6.2k.

Table 3.6.2a (cont'd)

INSTRUMENTATION THAT INITIATES SCRAM

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable: Shutdown	Refuel	Startup	Run
(6) Main-Steam-Line Isolation Valve Position	2	4(h)	$\leq$10 percent valve closure from full open		(c)	(c)	X
(7) High Radiation Main-Steam-Line	2	2	$\leq$5 times normal background at rated power		X	X	X
(8) Shutdown Position of Reactor Mode Switch	2	1	- -		(k)	X	X
(9) Neutron Flux							
(a) IRM							
(i) Upscale	2	3(d)	$\leq$96 percent of full scale		(g)	(g)	(g)

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Table 3.6.2a (cont'd)

INSTRUMENTATION THAT INITIATES SCRAM

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable			
				Shutdown	Refuel	Startup	Run
(ii) Inoperative	2	3(d)	- -		X	X	
(b) APRM (i) Upscale	2	3(e)	Figure 2.1.1		X	X	X
(ii) Inoperative	2	3(e)	- -		X	X	X
(iii) Downscale	2	3(e)	$\geq$5 percent of full scale		(g)	(g)	(g)
(10) Turbine Stop Valve Closure	2	4	$\leq$10% valve closure				(i)
(11) Generator Load Rejection	2	2	(j)				(i)

Table 4.6.2a

INSTRUMENTATION THAT INITIATES SCRAM

Surveillance Requirement

	Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(1)	Manual Scram	None	Once per 3 months	None
(2)	High Reactor Pressure	None	Once per month(1)	Once per 3 months(1)
(3)	High Drywell Pressure	None	Once per month(1)	Once per 3 months(1)
(4)	Low Reactor Water Level	Once/day	Once per month(1)	Once per 3 months(1)
(5)	High Water Level Scram Discharge Volume	None	Once per month	Once per 3 months
(6)	Main-Steam-Line Isolation Valve Position	None	Once per 3 months	None
(7)	High Radiation Main-Steam Line	Once/shift	Once per week	Once per 3 months

Table 4.6.2a (cont'd)

INSTRUMENTATION THAT INITIATES SCRAM

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(8) Shutdown Position of Reactor Mode Switch	None	Once during each major refueling outage	None
(9) Neutron Flux			
(a) IRM			
(i) Upscale	(f)	(f)	(f)
(ii) Inoperative	(f)	(f)	(f)
(b) APRM			
(i) Upscale	None	Once/week	Once/week
(ii) Inoperative	None	Once/week	Once/week
(iii) Downscale	None	Once/week	Once/week
(10) Turbine Stop Valve Closure	None	Once per 3 months	None
(11) Generator Load Rejection	None	Once per month	Once per 3 months

NOTES FOR TABLES 3.6.2a AND 4.6.2a

(a) May be bypassed when necessary for containment inerting.

(b) May be bypassed in the refuel and shutdown positions of the reactor mode switch with a keylock switch.

(c) May be bypassed in the refuel and startup positions of the reactor mode switch when reactor pressure is less than 600 psi.

(d) No more than one of the four IRM inputs to each trip system shall be bypassed.

(e) No more than two C or D level LPRM inputs to an APRM shall be bypassed and only four LPRM inputs to an APRM shall be bypassed in order for the APRM to be considered operable. No more than one of the four APRM inputs to each trip system shall be bypassed provided that the APRM in the other instrument channel in the same core quadrant is not bypassed. A Travelling In-Core Probe (TIP) chamber may be used as a substitute APRM input if the TIP is positioned in close proximity to the failed LPRM it is replacing.

(f) Calibrate prior to starting and normal shutdown and thereafter check once per shift and test once per week until no longer required.

(g) IRM's are bypassed when APRM's are onscale. APRM downscale is bypassed when IRM's are onscale.

(h) Each of the four isolation valves has two limit switches. Each limit switch provides input to one of two instrument channels in a single trip system.

(i) May be bypassed when reactor power level is below 45%.

(j) Trip upon loss of oil pressure to the acceleration relay.

(k) May be bypassed when placing the reactor mode switch in the SHUTDOWN position and all control rods are fully inserted.

(l) Only the trip circuit will be calibrated and tested at the frequencies specified in Table 4.6.2a, the primary sensor will be calibrated and tested once per operating cycle.

Table 3.6.2b

INSTRUMENTATION THAT INITIATES
PRIMARY COOLANT SYSTEM OR CONTAINMENT ISOLATION

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable: Shutdown	Refuel	Startup	Run
PRIMARY COOLANT ISOLATION (Main Steam, Cleanup, and Shutdown)							
(1) Low-Low Reactor Water Level	2	2	>5 inches (Indicator Scale)			X	X
(2) Manual	2	1	- -	X	X	X	X
MAIN-STEAM-LINE ISOLATION							
(3) High Steam Flow Main-Steam Line	2	2	$\leq$105 psid			X	X

Table 3.6.2b (cont'd)

INSTRUMENTATION THAT INITIATES
PRIMARY COOLANT SYSTEM OR CONTAINMENT ISOLATION

Limiting Condition for Operation

	Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable			
					Shutdown	Refuel	Startup	Run
(4)	High Radiation Main-Steam Line	2	2	$\leq$5 times normal back-ground at rated power			X	X
(5)	Low Reactor Pressure	2	2	$\geq$850 psig				X
(6)	Low-Low-Low Condenser Vacuum	2	2	$\geq$7 in. mercury vacuum			(a)	X
(7)	High Temperature Main-Steam-Line Tunnel	2	2	$\leq$200F			X	X

Table 3.6.2b (cont'd)

INSTRUMENTATION THAT INITIATES
PRIMARY COOLANT SYSTEM OR CONTAINMENT ISOLATION

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable: Shutdown	Refuel	Startup	Run
CLEANUP SYSTEM ISOLATION							
(8) .High Area Temperature	1	2	≤190	X	X	X	X
SHUTDOWN COOLING SYSTEM ISOLATION							
(9) High Area Temperature	1	1	≤170	X	X	X	X
CONTAINMENT ISOLATION							
(10) Low-Low Reactor Water	2	2	≥5 inches (Indicator Scale)	(c)		X	X

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Table 3.6.2b (cont'd)

INSTRUMENTATION THAT INITIATES
PRIMARY COOLANT SYSTEM OR CONTAINMENT ISOLATION

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable: Shutdown	Refuel	Startup	Run
(11) High Drywell Pressure	2	2	≤3.5 psig	(c)		(b)	(b)
(12) Manual	2	1	- -	X	X	X	X

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Table 4.6.2b

INSTRUMENTATION THAT INITIATES
PRIMARY COOLANT SYSTEM OR CONTAINMENT ISOLATION

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
PRIMARY COOLANT ISOLATION (Main Steam, Cleanup and Shutdown)			
(1) Low-Low Reactor Water Level	Once/day	Once per month(d)	Once per 3 months(d)
(2) Manual	- -	Once during each major refueling outage	- -
MAIN-STEAM-LINE ISOLATION			
(3) High Steam Flow Main-Steam Line	Once/day	Once per month(d)	Once per 3 months(d)
(4) High Radiation Main-Steam Line	Once/shift	Once/week	Once per 3 months
(5) Low Reactor Pressure	Once/day	Once per month(d)	Once per 3 months(d)

Table 4.6.2b (cont'd)

INSTRUMENTATION THAT INITIATES
PRIMARY COOLANT SYSTEM OR CONTAINMENT ISOLATION

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
CONTAINMENT ISOLATION			
(10) Low-Low Reactor Water Level	Once/day	Once per month[(d)]	Once per 3 months[(d)]
(11) High Drywell Pressure	Once/day	Once per month[(d)]	Once per 3 months[(d)]
(12) Manual	- -	Once during each operating cycle	- -

NOTES FOR TABLES 3.6.2b AND 4.6.2b

(a) May be bypassed in the refuel and startup positions of the reactor mode switch when reactor pressure is less than 600 psi.

(b) May be bypassed when necessary for containment inerting.

(c) May be bypassed in the shutdown mode whenever the reactor coolant system temperature is less than 215°F.

(d) Only the trip circuit will be calibrated and tested at the frequencies specified in Table 4.6.2b, the primary sensor will be calibrated and tested once per operating cycle.

Table 3.6.2c

INSTRUMENTATION THAT INITIATES OR ISOLATES EMERGENCY COOLING

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System	Set-Point	Reactor Mode Switch Position in Which Function Must Be Operable: Shutdown	Refuel	Startup	Run
EMERGENCY COOLING INITIATION							
(1) High-Reactor Pressure	2	2	≤1080 psig	(b)		x	x
(2) Low-Low Reactor Water Level	2	2	≥ 5 inches (Indicator Scale)	(b)		x	x
EMERGENCY COOLING ISOLATION (for each of two systems)							
(3) High Steam Flow Emergency Cooling System	2	2 (a)	19 psid			x	x
(4) High Radiation Emergency Cooling System Vent	1	2	25 mr/hr			x	x

Table 4.6.2c

INSTRUMENTATION THAT INITIATES OR ISOLATES EMERGENCY COOLING

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
EMERGENCY COOLING INITIATION			
(1) High Reactor Pressure	None	Once per month[c]	Once per 3 months[c]
(2) Low-Low Reactor Water Level	Once/day	Once per month[c]	Once per 3 months[c]
EMERGENCY COOLING ISOLATION (for each of two systems)			
(3) High Steam Flow Emergency Cooling System	None	Once per 3 months[c]	Once per 3 months[c]
(4) High Radiation Emergency Cooling System Vent	Once/shift	Once during each major refueling outage	Once during each major refueling outage

NOTES FOR TABLES 3.6.2c AND 4.6.2c

(a) Each of two differential pressure switches provide inputs to one instrument channel in each trip system.

(b) May be bypassed in the cold shutdown condition.

(c) Only the trip circuit will be calibrated and tested at the frequencies specified in Table 4.6.2c, the primary sensor will be calibrated and tested once per operating cycle.

Table 3.6.2d

INSTRUMENTATION THAT INITIATES CORE SPRAY[e]

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System	Set-Point	Reactor Mode Switch Position in Which Function Must Be Operable: Shutdown	Refuel	Startup	Run
START CORE SPRAY PUMPS							
(1) High Drywell Pressure	2	2	$\leq$3.5 psig	(d)	x	(a)	(a)
(2) Low-Low Reactor Water Level	2	2	$\geq$5 inches (Indicator Scale)	(b)	X	X	X
OPEN CORE SPRAY DISCHARGE VALVES							
(3) Reactor Pressure and either (1) or (2) above.	2	2	$\geq$365 psig	X	X	X	X

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Table 4.6.2d

INSTRUMENTATION THAT INITIATES CORE SPRAY

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
START CORE SPRAY PUMPS			
(1) High Drywell Pressure	Once/day	Once per month(c)	Once per 3 months(c)
(2) Low-Low Reactor Water Level	Once/day	Once per month(c)	Once per 3 months(c)
OPEN CORE SPRAY DISCHARGE VALVES			
(3) Reactor Pressure and either (1) or (2) above	None	Once per month(c)	Once per 3 months(c)

NOTES FOR TABLES 3.6.2d AND 4.6.2d

(a) May be bypassed when necessary for containment inerting.

(b) May be bypassed when necessary for performing major maintenance as specified in Specification 2.1.1.e.

(c) Only the trip circuit will be calibrated and tested at the frequencies specified in Table 4.6.2d, the primary sensor will be calibrated and tested once per operating cycle.

(d) May be bypassed when necessary for integrated leak rate testing.

(e) The instrumentation that initiates the Core Spray System is not required to be operable, if there is no fuel in the reactor vessel.

Table 3.6.2e

INSTRUMENTATION THAT INITIATES CONTAINMENT SPRAY

Limiting Condition for Operation

Parameter		Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System	Set-Point	Reactor Mode Switch Position in Which Function Must Be Operable			
					Shutdown	Refuel	Startup	Run
(1) a.	High Drywell Pressure	2	2	$\leq$ 3.5 psig	(a)		x	x
	and							
b.	Low-Low Reactor Water Level	2	2	$\geq$ 5 inches (Indicator Scale)	(a)		x	x

Table 4.6.2e

INSTRUMENTATION THAT INITIATES CONTAINMENT SPRAY

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(1)a. High Drywell Pressure	Once/day	Once per month[b]	Once per 3 months[b]
b. Low-Low Reactor Water Level	Once/day	Once per month[b]	Once per 3 months[b]

NOTES FOR TABLES 3.6.2e AND 4.6.2e

(a) May be bypassed in the shutdown mode whenever the reactor coolant temperature is less than 215^{o} F.

(b) Only the trip circuit will be calibrated and tested at the frequencies specified in Table 4.6.2e, the primary sensor will be calibrated and tested once per operating cycle.

Table 3.6.2f

INSTRUMENTATION THAT INITIATES AUTO DEPRESSURIZATION

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System	Set-Point	Reactor Mode Switch Position in Which Function Must Be Operable			
				Shutdown	Refuel	Startup	Run
INITIATION							
(1) a. Low-Low-Low Reactor Water Level	2 (a)	2 (a)	$\geq$ 3.88 inches (Indicator scale)	(b)		(b)	x
and							
b. High Drywell Pressure	2 (a)	2 (a)	$\leq$ 3.5 psig	(b)		(b)	x

Table 4.6.2f

INSTRUMENTATION THAT INITIATES AUTO DEPRESSURIZATION

Surveillance Requirement

Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
INITIATION			
(1) a. Low-Low-Low Reactor Water	None	Once per month$^{(c)}$	Once per 3 months$^{(c)}$
and			
b. High Drywell Pressure	Once/day	Once per month$^{(c)}$	Once per 3 months$^{(c)}$

NOTES FOR TABLES 3.6.2f AND 4.6.2f

(a) Both instrument channels in either trip system are required to be energized to initiate auto depressurization. One trip system is powered from power board 102 and the other trip system from power board 103.

(b) May be bypassed when the reactor pressure is less than 110 psig and the reactor coolant temperature is less than the corresponding saturation temperature.

(c) Only the trip circuit will be calibrated and tested at the frequencies specified in Table 4.6.2f, the primary sensor will be calibrated and tested once per operating cycle.

Table 3.6.2k

HIGH PRESSURE COOLANT INJECTION

Limiting Condition for Operation

Parameter	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System	Set-Point	Reactor Mode Switch Position in Which Function Must Be Operable: Shutdown	Refuel	Startup	Run
(1) Low Reactor Water Level	2	2	$\geq$ 53 inches (Indicator scale)	(a)		(a)	X
(2) Automatic Turbine Trip	1	1	---	(a)		(a)	X

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Table 4.6.2k

HIGH PRESSURE COOLANT INJECTION

Surveillance Requirement

	Parameter	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(1)	Low Reactor Water Level	Once per day	Once per month[(b)]	Once per 3 months[(b)]
(2)	Automatic Turbine Trip	None	Once during each operating cycle	None

NOTES FOR TABLES 3.6.2k AND 4.6.2k

(a) May be bypassed when the reactor pressure is less than 110 psig and the reactor coolant temperature is less than the corresponding saturation temperature.

(b) Only the trip circuit will be calibrated and tested at the frequencies specified in Table 4.6.2k, the primary sensor will be calibrated and tested once per operating cycle.

The reactor protection system automatically initiates a reactor scram to prevent exceeding established limits. In addition, other protective instrumentation is provided to initiate action which mitigates the consequences of accidents or terminates operator error.

The reactor protection system is a dual channel type (Table 3.6.2.a). Each trip system except the manual scram has two independent instrument channels. Operation of either channel will trip the trip system, i.e., the trip logic of the channel is one-out-of-two. A simultaneous trip of both trip systems will cause a reactor scram, i.e., the tripping logic of the trip systems is two-out-of-two. The tripping logic of the total system is referred to as one-out-of-two taken twice. This system will accommodate any single failure and still perform its intended function and in addition, provide protection against spurious scrams. The reliability of the dual channel system or probability that it will perform its intended function is less than that of a one-out-of-two system and somewhat greater than that of a two-out-of-three system (Section VIII-A.1.0 of the FSAR).

The instrumentation used to initiate action other than scram is generally similar to the reactor protection system. There are usually two trip systems required or available for each function. There are usually two instrument channels for each trip system. Either channel can trip the trip system but both trip systems are required to initiate the respective action. Where only one trip system is provided only one instrument channel is required to trip the trip system. All instrument channels except those for automatic depressurization are normally energized. De-energizing causes a trip. Power to the trip systems for each function is from reactor protection system buses 11 and 12.

The signals for initiating automatic blowdown and rod block differ from other initiating signals in that only one of the two trip systems is required to start blowdown or initiate rod block. Both instrument channels in the trip system must trip to initiate automatic blowdown. This difference is due to the requirement that automatic depressurization be prevented unless A.C. power is available to the emergency core cooling systems. The instrument channels in the trip system for automatic depressurization are normally de-energized. In order to cause a trip both instrument channels must be energized. Power to energize the instrument channels is from power boards 102 and 103. If A.C. power is lost to one power board, one trip system becomes inoperable

b. The control rod block functions are provided to prevent excessive control rod withdrawal so that MCPR is maintained greater than 1.06. The trip logic for this function is 1 out of n; e.g., any trip on one of the eight APRM's, eight IRM's or four SRM's will result in a rod block. The minimum instrument channel requirements provide sufficient instrumentation to assure the single failure criteria is met. The minimum instrument channel requirements for the rod block may be reduced by one for a short period of time to allow maintenance, testing, or calibration. This time period is only ~3% of the operating time in a month and does not significantly increase the risk of preventing an inadvertent control rod withdrawal.

The APRM rod block trip is flow biased and prevents a significant reduction in MCPR especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCPR is maintained greater than 1.06.

The APRM rod block also provides local protection of the core; i.e., the prevention of critical heat flux in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern. The trip point is flow biased. The worst case single control rod withdrawal error has been analyzed and the results show that with the specified trip settings rod withdrawal is blocked before the MCPR reaches 1.06, thus allowing adequate margin. Below ~60% power the worst case withdrawal of a single control rod results in a MCPR > 1.06 without rod block action, thus below this level it is not required.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level. Analysis of the worst case accident results in rod block action before MCPR approaches 1.06.

A downscale indication on an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and the control rod motion is prevented. The downscale rod blocks are set at 5 percent of full scale for IRM and 2 percent of full scale for APRM (APRM signal is generated by averaging the output signals from eight LPRM flux monitors).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 37 TO FACILITY OPERATING LICENSE NO. DPR-63

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR STATION, UNIT NO. 1

DOCKET NO. 50-220

I. INTRODUCTION

The licensee, Niagara Mohawk Power Corporation, in its submittals as listed under reference of this report has proposed certain modifications to the Reactor Protection Systems (RPS)/Engineered Safety Features (ESF). These modifications, developed by the General Electric Company (GE) involve installing a new design for safety systems instrumentation in the RPS/ESF of Boiling Water Reactors (BWR). The new design, referred to as the Analog Transmitter/Trip Unit System (ATTUS), is being supplied as original equipment in the GE/BWR 6 and has been made available to BWR owners as a backfit. The ATTUS is a replacement for mechanical sensor switches at the parameter sensor level and does not involve the logic levels of RPS/ESF systems. GE developed ATTUS to offset operating disadvantages of the direct pressure and differential pressure actuated switches of the original safety system instrumentation.

The new ATTUS is comprised of an analog transmitter and trip unit/calibration system (Model 510DU). GE presented ATTUS to the NRC staff for licensing under topical report NEDO-21617 of April 1977 and NEDO-21617-1 of January 1978. The staff reviewed and found acceptable ATTUS in its letter to GE dated June 27, 1978.

The staff in its approval of ATTUS required from those licensees who are backfitting their nuclear units certain plant specific information in order to interface the review with the staff's review of the topical report on the subject. The particular information required of the licensee is the environmental qualification and the divisional separation of the hardware installed for the plant backfit.

II. EVALUATION

The ATTUS, as stated above, is a replacement for the mechanical type sensor switches at the sensor level and not the logic level. Since the dual channel design (with two trip systems) of the RPS is not being altered, the safe and reliable operation of the trip system is not compromised. The automatic and manual initiation and protective action of essential systems remain unchanged. The parameter sensors being replaced with ATTUS along with the safety systems they actuate are listed below:

Parameters Sensors	Safety Systems
Main Steam Flow - High	RPS/Engineered Safety Features
Drywell Pressure - High	RPS/Engineered Safety Features
Reactor Vessel Level - High Reactor Vessel Level - Low	RPS/Engineered Safety Features
Reactor Vessel Level - Low, Low, Low	Engineered Safety Features
Reactor Vessel Pressure - High Reactor Vessel Pressure - Low	RPS/Engineered Safety Features
Reactor Vessel Pressure - High (Emergency Condenser Initiation) and Reactor Vessel Pressure Low (Opens Core Spray Discharge Valves)	Engineered Safety Features
Emergency Condenser Flow High	Engineered Safety Features

The new transmitters replacing the existing mechanical switches are Rosemount Model #1151DP for differential pressure indication and Rosemount Model #1151GP for pressure indication. Differential pressure transmitters are used for level and flow indication and pressure transmitters are used for all other pressure indication.

The four channel sensor system and local trip unit cabinets for the modifications to the RPS/ESF systems satisfies the single failure criteria and the applicable separation criteria in force when the plant was constructed. The new differential pressure and pressure transmitters sensors for the seven parameters being monitored are to be mounted in the four existing transmitter support racks located in the east, west, and north instrument rooms of the reactor building. The existing racks have been modified to accommodate the new transmitters and the rack supports have been modified to meet site seismic requirements.

Basically, the modification to the existing RPS is the rerouting of cables from the four transmitter support racks to the four new local trip unit cabinets (supplied with ATTUS) also located in the reactor building. These units at the 281' level are separated from one another by a minimum distance between any two of approximately 52 feet. The cabling returns from these cabinets to the existing raceways between the reactor building and control room to connect to the RPS cabinets. The physical independence, separation, and isolation of the system have not been changed from the initial construction criteria. We find this acceptable.

The new ATTUS equipment, as indicated above, is located in the reactor building. The accident environmental conditions for the area where the equipment is located is: temperature 150°F, pressure 0.28 psig, relative humidity 100%, and radiation 1×10^5 R. The transmitters are environmental qualified to: 212°F, 50 psi steam, and 1.7×10^5 R. The local trip unit cabinets and components are environmentally qualified to: 156°F, 8" WC, 99% relative humidity and 1.7×10^5 R. The normal conditions in this area maintained by the ventilation system are: temperature 70/80°F; pressure - .25" WC, relative humidity 20-80% and radiation 5-15 mr/hr. Credit is not taken for post accident monitoring for this instrumentation. We find the environmental qualifications for the equipment to be acceptable.

Based on our review of the licensee's submittals, we conclude that the modifications to the reactor protection system satisfies the requirements for single failure, electrical isolation and physical separation, and environmental qualifications; and, therefore, are acceptable.

III. ENVIRONMENTAL CONSIDERATIONS

We have determined that this amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that this amendment involves an action which is insignificant from the standpoint of environmental impact, and pursuant to 10 CFR §51.5(d)(4) that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

IV. CONCLUSION

We have concluded that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: May 2, 1980

REFERENCES

1. LeBoeuf, Lamb, Leiby & MacRae, letter dated February 15, 1979.

2. Niagara Mohawk Power Corporation, letter dated March 27, 1979.

3. Niagara Mohawk Power Corporation, letter dated April 3, 1979

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UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-220

NIAGARA MOHAWK POWER CORPORATION

NOTICE OF ISSUANCE OF FACILITY LICENSE AMENDMENT

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 37 to Facility Operating License No. DPR-63 to Niagara Mohawk Power Corporation (the licensee) which revised the Technical Specifications for operation of the Nine Mile Point Nuclear Station, Unit No. 1 (the facility) located in Oswego County, New York. The amendment is effective as of its date of issuance.

The amendment revises the Technical Specifications to reflect the installation of an Analog Transmitter Trip Unit System (ATTUS).

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated February 15, 1979 supplemented by letters dated March 27 and April 3, 1979, (2) Amendment No. 37 to License No. DPR-63, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW., Washington, D. C. and at the Oswego County Office Building, 46 E. Bridge Street, Oswego, New York 13126. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland this 2nd day of May 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

TA Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors