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Docket No. 50-220

OCT 1 1975

Niagara Mohawk Power Corporation
ATTN: Mr. Gerald K. Rhode
Vice President - Engineering
300 Erie Boulevard West
Syracuse, New York 13202

Gentlemen:

The Commission has issued the enclosed Amendment No. 1 to Facility Operating License No. DPR-63 for the Nine Mile Point Nuclear Station, Unit 1. The amendment also includes Change No. 1 to the Technical Specifications in accordance with your applications dated January 21 and February 27, 1975, and supplement dated March 19, 1975.

The amendment modifies the Technical Specifications relating to the water level instrumentation, the low-low-low water level setpoint, description of the access penetrations in reactor building railroad bay, changes in position titles in the station operating organization, and to correct typographical errors.

Copies of the related Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

/S/

George Lear, Chief
Operating Reactors Branch #3
Division of Reactor Licensing

Enclosures:
1. Amendment No. 1
2. Safety Evaluation
3. Federal Register Notice

cc w/encls:
See next page

OFFICE	RL:ORB-3	RL:ORB-3	OELD	RL:ORB-3		
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DATE	8/26/75	8/26/75	9/24/75	10/1/75		

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

Niagara Mohawk Power Corporation

cc: w/enclosures

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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 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 1
License No. DPR-63

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

   A. The applications for amendment by Niagara Mohawk Power Corporation (the licensee) dated January 21, 1975, February 27, 1975, and supplement dated March 19, 1975, comply 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; and

   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.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility 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, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 1."

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

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Reactor Licensing

Attachment:
Change No. 1 to the
Technical Specifications

Date of Issuance: **OCT 1 1975**

ATTACHMENT TO LICENSE AMENDMENT NO. 1

CHANGE NO. 1 TO TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-63

DOCKET NO. 50-220

A. Replace the following pages of Appendix A, "Radiological Technical Specifications": cover page, 6, 8, 13, 16, 17, 49, 52, 59, 138, 159, 162, 171, 191, 197, 199, 205, 208, 211, 213, 231, 247, and 248 with correspondingly numbered, revised pages.

B. Replace page 82 of Appendix B, "Environmental Technical Specifications", with revised page 82.

RADIOLOGICAL TECHNICAL SPECIFICATIONS

APPENDIX A

TO

FACILITY OPERATING LICENSE NO. DPR-63

FOR THE

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR STATION UNIT 1

DOCKET NO. 50-220

DECEMBER 26, 1974

SAFETY LIMIT

b. When the reactor pressure is less than 585 psig or reactor recirculation flow is less than 5% of design, the reactor thermal power shall not exceed 333 Mwt.

c. The neutron flux shall not exceed its scram setting for longer than 1.7 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.

d. Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be more than seven feet eleven inches (127.1 inches indicator scale) below minimum normal water level (Elevation 302'9").

LIMITING SAFETY SYSTEM SETTING

b. The IRM scram trip setting shall not exceed 12 percent of rated neutron flux.

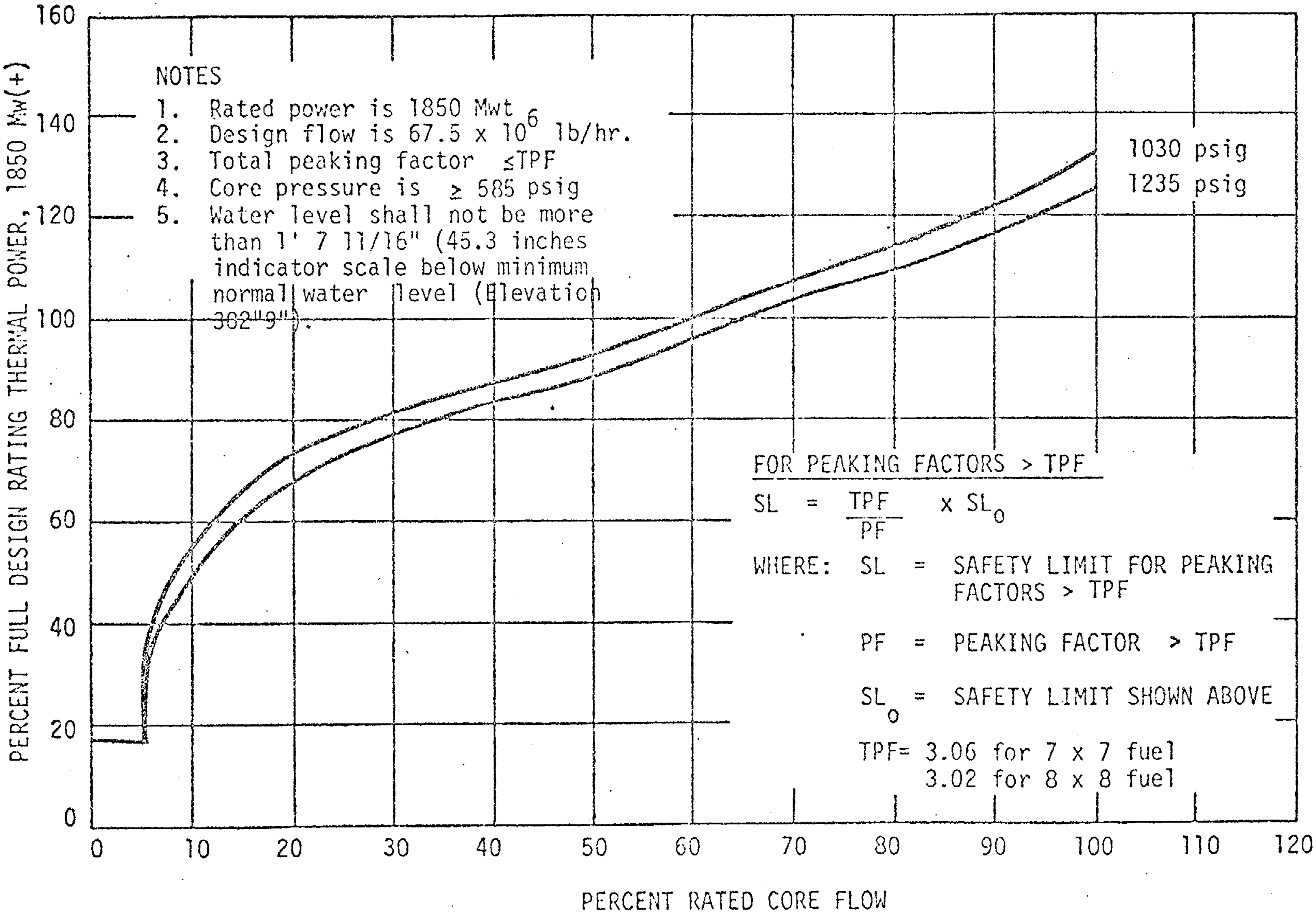
c. The reactor high-pressure scram trip setting shall be $\leq$ 1080 psig.

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 (Elevation 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 2.1.2 Revised.

Figure 2.1.1 Revised
FUEL CLADDING INTEGRITY SAFETY LIMIT



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Should a power transient occur, the process computer would show the time interval the neutron flux is over its scram setting. When the process computer is out of service, a safety limit violation will be assumed if the neutron flux exceeds the scram setting and control rod scram does not occur. The process computer shall be returned to an operable condtion as soon as practical.

When the reactor is shut down, consideration must alse be given to water level requirements due to the effect of decay heat. Specification 2.1.1.d provides a limit for the shutdown water level. If the reactor water level dropped below the top of the active fuel, the ability to cool the core would be reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. But with a water level above the top of the active fuel, adequate cooling would be maintained and the decay heat could easily be accommodated.

The lowest point at which the water level can be monitored is approximately four feet eight inches above the top of the active fuel. This is the low-low-low water level trip point, which is seven feet eleven inches (127.1 inches indicator scale) below minimum normal water level (Elevation 302'9"). The safety limit has
1 been established here to provide a point which can be monitored and also can provide adequate margin.

system in the low power flow condition. Effects of increasing pressure at zero or low void content are minor, cold water from sources available during startup is not much colder than that already in the system, temperature coefficients are small, and control rod patterns are constrained to be uniform by operating procedures backed up by the rod worth minimizer. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than five percent of rated per minute, and the IRM system would be more than adequate to assure a scram before the power could exceed the safety limit.

Procedural controls will assure that the IRM scram is maintained up to 20 percent flow. This is accomplished by keeping the reactor mode switch in the startup position unitl 20 percent flow is exceeded and the APRM's are on scale. Then the reactor mode switch may be switched to the run mode, thereby switching scram protection from the IRM to the APRM system.

c. As demonstrated in Appendix E-I* and the Technical Supplement to Petition to Increase Power Level, the reactor high pressure scram is a backup to the neutron flux scram, turbine stop valve closure scram, generator load rejection scram, and main steam isolation-valve closure scram, for various reactor isolation incidents. However, rapid isolation at lower power levels generally results in high pressure scram preceding other scrams because the transients are slower and those trips associated with the turbine-generator are bypassed.

The operator will set the trip setting at 1080 psig or lower. However, the actual set point can be as much as 15.8 psi above the 1080 psig indicated set point due to the deviations discussed above.

1 d. A reactor water low-level scram trip setting -12 inches (53 inches indicator scale) relative to the minimum normal water level (Elevation 302'9") will assure that power production will be terminated with adequate coolant remaining

*FSAR

in the core. The analysis of the feedwater pump loss in the Technical Supplement to Petition to Increase Power Level, dated April 1970 has demonstrated that approximately four feet of water remains above the core following the low-level scram.

The operator will set the low-level trip setting no lower than -12 inches (53 inches indicator scale)
1 relative to the minimum normal water level (Elevation 302'9"). However, the actual set point can be as
much as 2.6 inches lower due to the deviations discussed above.

e. A reactor water low-low level signal - 5 feet (5 inches indicator scale) relative to the minimum normal
1 water level (Elevation 302'9") will assure that core cooling will continue even if level is dropping.
Core spray cooling will adequately cool the core, as discussed in LCO 3.1.4.

The operator will set the low-low level core spray initiation point at no less than -5 feet (5 inches indicator
1 scale) relative to the minimum normal water level (Elevation 302'9"). However, the actual set point can be as
much as 2.6 inches lower due to the deviations discussed above.

f. The APRM rod block trip setting will be varied automatically with recirculation flow with the trip setting at rated flow being $\leq$ 105 percent of rated neutron flux. Although the operator will set the rod block setting at less than or equal to that shown in Figure 2.1.2 Revised, the actual neutron flux setting can be as much as 2.7 percent of rated neutron flux above the line. This includes the errors discussed above. The flow bias can vary as much as one percent of rated recirculation flow above or below the indicated point.

g-h. The low pressure isolation of the main steam lines at 850 psig was provided to give protection against fast reactor depressurization and the resulting rapid cool-down of the vessel. Advantage was taken of the scram feature which occurs when the main-steam-line isolation valves are closed, to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation of the reactor at pressures lower than 850 psig requires that the reactor mode switch be in the startup position where protection of the

The turbine main condenser is normally available. The emergency cooling system (Section V-E*) is provided as a redundant backup for core decay heat removal following reactor isolation and scram. One emergency condenser system has a heat removal capacity at normal pressure of 19.0 x 10^7 Btu/hr, which is approximately three percent of maximum reactor steam flow. This capacity is sufficient to handle the decay heat production at 100 seconds following a scram. If only one of the emergency cooling systems is available, 2000 pounds of water will be lost from the reactor vessel through the relief valves in the 100 seconds following isolation and scram. This represents a minor loss relative to the vessel inventory of about 450,000 pounds (Section V-E.3.1*).

The required heat removal capability is based on the data of Table V-1* adjusted to normal operating pressures. The only difference is manual system initiation rather than automatic initiation.

The system may be manually initiated at any time. The system is automatically initiated on high reactor pressure in excess of 1080 psig sustained for 10 seconds. The time delay is provided to prevent unnecessary actuation of the system during anticipated turbine trips (Appendix E-I.3.13*). Automatic initiation is provided to minimize the coolant loss following isolation from the main condenser.** To assist in depressurization for small line breaks the system is initiated on low-low reactor water level five feet (5 inches indicator scale)
1 below the minimum normal water level (Elevation 302'9") sustained for 10 seconds. The timers for initiation of the emergency condensers will be set at 10 seconds delay based on the analysis (Appendix E-I.3.13*) although they can be set anywhere between 10 and 15 seconds.

The initial water volume in each emergency condenser is 21,360 ±1500 gallons which keeps the level within ±6 inches of the normal water level. About 72,000 gallons are available from the two gravity feed condensate storage tanks. To assure this gallonage, a level check shall be done at least once per day.

This is sufficient to provide about eight hours of continuous system operation. This time is sufficient to restore additional heat sinks or pump makeup water from the two-200,000 gallon condensate storage tanks. The fire protection is also available as a makeup water supply.

* FSAR
** Technical Supplement to Petition to Increase Power Level

LIMITING CONDITION FOR OPERATION

c. If a redundant component in each of the core spray systems becomes inoperable, both systems shall be considered operable provided that the component is returned to an operable condition within 7 days and the additional surveillance required is performed.

d. If a core spray system becomes inoperable and all the components are operable in the other system, the reactor may remain in operation for a period not to exceed 7 days.

e. If Specifications a, b, c and d are not met, a normal orderly shutdown shall be initiated within one hour and the reactor shall be in the cold shutdown condition within ten hours.

If both core spray systems become inoperable the reactor shall be in the cold shutdown condition within ten hours and no work (except as specified in "f" below) shall be performed on the reactor or its connected systems which could result in lowering the reactor water level to more than seven feet eleven inches below minimum
1 normal water level (127.1 inches indicator scale).

SURVEILLANCE REQUIREMENT

d. Core spray header ΔP instrumentation

check	Once/day
calibrate	Once/3 months
test	Once/3 months

e. <u>Surveillance with Inoperable Components</u>

When a component or system becomes inoperable its redundant component or system shall be demonstrated to be operable immediately and daily thereafter.

f. Surveillance during control rod drive maintenance which is simultaneous with the suppression chamber unwatered shall include at least hourly checks that the conditions listed in 3.1.4f are met.

Pressure Blowdown

In the event of a small line break, substantial coolant loss could occur from the reactor vessel while it was still at relatively high pressures. A pressure blowdown system is provided which in conjunction with the core spray system will prevent significant fuel damage for all sized line breaks (Appendix E-11.2.0*).

Operation of three solenoid-actuated pressure relief valves is sufficient to depressurize the primary system to 110 psig which will permit full flow of the core spray system within required time limits (Appendix E-11.2*). Requiring all six of the relief valves to be operable, therefore, provides twice the minimum number required. Prior to or following refueling at low reactor pressure, each valve will be manually opened to verify valve operability. The malfunction analysis (Section II.XV, "Technical Supplement to Petition to Increase Power Level," dated April 1970) demonstrates that no serious consequences result if one valve fails to close since the resulting blowdown is well within design limits.

In the event of a small line break, considerable time is available for the operator to permit core spray operation by manually depressurizing the vessel using the solenoid-actuated valves. However, to ensure that the depressurization will be accomplished, automatic features are provided. The relief valves shall be capable of automatic initiation from simultaneous low-low-low water level (7'-11" below minimum normal water level
1 at Elevation 302'9", 127.1 inches indicator scale) and high containment pressure (3.5 psig). The system response to small breaks requiring depressurization is discussed in Section VII-A.3.3* and the time available to take operator action is summarized in Table VII-1.* Additional information is included in the answers to Questions III-1 and III-5 of the First Supplement.

Steam from the reactor vessel is discharged to the suppression chamber during valve testing. Conducting the tests with the reactor at low pressure such as just prior to or just after refueling minimizes the stress on the reactor coolant system.

The test interval of once per operating cycle results in a system failure probability of 7.0×10^{-7} (Fifth Supplement, p. 115)* and is consistent with practical consideration.

*FSAR

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

d. Frequency

Three integrated leak rate tests shall be performed at approximately equal intervals during each 10-year service period with the third test in each ten-year interval corresponding with the ten-year scheduled in-service inspection shutdown.

e. Local Leak Rate Tests

(1) Primary containment testable penetrations and isolation valves shall be tested at a pressure of 35 psig each major refueling outage except bolted double-gasketed seals shall be tested whenever the seal is closed after being opened, and at least at each refueling outage. 1

(2) Personnel air lock door seals shall be tested once within 24 hours after opening when the reactor is in a power operating condition, at a pressure of 10 psig and the leak rate extrapolated to 35 psig. Air lock seals shall also be leak rate tested at a pressure of 35 psig at the beginning of each operating cycle. An additional 35 psig leak rate test shall be performed near the middle of the operating cycle should a shutdown requiring de-inerting arise. If the above shutdown does not occur or is not anticipated, the air lock seals will be

LIMITING CONDITION FOR OPERATION

c. If a redundant component in each of the containment spray systems or their associated raw water systems become inoperable, both systems shall be considered operable provided that the component is returned to an operable condition within 7 days and that the additional surveillance required is performed.

d. If a containment spray system or its associated raw water system becomes inoperable and all the components are operable in the other systems, the reactor may remain in operation for a period not to exceed 7 days.

e. If Specifications a or b are not met, shutdown shall begin within one hour and the reactor coolant shall be below 215F within ten hours.

If both containment spray systems
become inoperable the reactor
shall be in the cold shutdown
condition within ten hours and
no work (except as specified in
"f" below) shall be performed on
the reactor which could result in
lowering the reactor water level
to more than seven feet eleven
1 inches (127.1 inches indicator
scale) below minimum normal water
level (Elevation 302'9").

SURVEILLANCE REQUIREMENT

c. Raw Water Cooling Pumps

At least once per quarter manual startup and operability of the raw water cooling pumps shall be demonstrated.

d. Surveillance with Inoperable Components

When a component or system becomes inoperable its redundant component or system shall be demonstrated to be operable immediately and daily thereafter.

e. Surveillance during control rod drive maintenance which is simultaneous with the suppression chamber unwatered shall include at least hourly checks that the conditions listed in 3.3.7.f are met.

BASES FOR 3.3.7 AND 4.3.7 CONTAINMENT SPRAY SYSTEM

For reactor coolant temperatures less than 215 F not enough steam is generated during a loss-of-coolant accident to pressurize the containment. In fact, for coolant temperatures up to 312 F, the resultant loss-of coolant accident pressure would not exceed the design pressure of 35 psig.

Operation of only one containment spray pump is sufficient to provide the required containment spray flow. The specified flow of 3000 gpm (approximately 95 percent to the drywell and the balance to the suppression chamber) is sufficient to remove post accident core energy released including a substantial chemical re-action involving hydrogen generation and will also limit pressure and temperature rises in the pressure suppression system to below design values (Appendix E-II 2.2.3 p.E-78 and the Fifth Supplement).* Each containment spray system is considered operable when both pumps are capable of delivering at least 3000 gpm at a pump developed head of 375 feet of water at 60 F. Requiring both pumps in both systems operable (400 percent redundancy) will assure the availability of the containment spray system.

Allowable outages are specified to account for components that become inoperable in both systems and for more than one component in a system.

The corresponding raw water cooling system is designed to maintain containment spray water temperature no greater than 140 F under the most limiting operating conditions. The containment spray raw water cooling system is considered operable when the flow rate is not less than 3000 gpm and the pressure on the raw water
1 side of the containment spray heat exchangers is not less than 160 psig. The higher pressure on the raw water side will assure that any leakage is into the containment spray system.

Electrical power for all system components is normally available from the reserve transformer. Upon loss of this service the pumping requirement will be supplied from the diesel generator. At least one diesel generator shall always be available to provide backup electrical power for one containment spray system, corresponding raw water cooling system and associated electronic equipment required for automatic system initiation.

Automatic initiation of the containment spray system assures that the containment will not be overpres-surized due to hydrogen generation. This automatic feature would only be required if all core spray system malfunctioned and significant metal-water reaction occurred. For the normal operation condition of 90F suppression chamber water and two psig containment pressure, containment spray actuation would not be necessary for about 15 minutes. Raw water cooling affects the temperature of the spray water and the

*FSAR

LIMITING CONDITION FOR OPERATION

3.4.3 ACCESS CONTROL

Applicability:

Applies to the access control to the reactor building.

Objective:

To specify the requirements necessary to assure the integrity of the secondary containment system.

Specification:

a. Only one door in each of the double-doored access ways shall be opened at one time.

1 b. Only one door or closeup of the railroad bay shall be opened at one time.

c. The core spray and containment spray pump compartments' doors shall be closed at all times except during passage in order to consider the core spray system and the containment spray system operable.

SURVEILLANCE REQUIREMENT

4.4.3 ACCESS CONTROL

Applicability:

Applies to the periodic checking of the condition of portions of the reactor building.

Objective:

To assure that pump compartments are properly closed at all times.

Specification:

a. The core and containment spray pump compartments shall be checked once per week and after each entry.

Table 3.6.2a

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
(1) Manual Scram	2	1			X	X	X
(2) High Reactor Pressure	2	2	≤1080 psig		X	X	X
(3) High Drywell Pressure	2	2	≤3.5 psig		X	(a)	(a)
1 (4) Low Reactor Water Level	2	2	≥53 inches. (Indicator scale).		X	X	X
(5) High Water Level Scram Discharge Volume	2	2	≤37 gal.		(b)	X	X

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 (1) Low-Low Reactor Water Level	2	2	≥5 inches. (Indicator scale)	X	X	X	X
(2) Manual	2	1	- -	X	X	X	X
MAIN-STEAM-LINE ISOLATION							
(3) High Steam Flow Main-Steam Line	2	2	≤ 105 psid	X	X	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	$\leq$190F	X	X	X	X
SHUTDOWN COOLING SYSTEM ISOLATION							
(9) High Area Temperature	1	1	$\leq$170F	X	X	X	X
CONTAINMENT ISOLATION							
1 (10) Low-Low Reactor Water Level	2	2	$\geq$5 inches. (Indicator scale)	X	X	X	X

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-High Reactor Pressure	2	2	≤1080 psig	X	X	X	X
1 (2) Low-Low Reactor Water Level	2	2	>5 inches. (Indicator scale).	X	X	X	X
EMERGENCY COOLING ISOLATION (for each of two systems)							
(3) High Steam Flow Emergency Cooling System	2	2 (a)	19 psid	X	X	X	X
(4) High Radiation Emergency Cooling System Vent	1	2	25 mr/hr	X	X	X	X

Table 3.6.2d

INSTRUMENTATION THAT INITIATES CORE 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
	START CORE SPRAY PUMPS							
	(1) High Drywell Pressure	2	2	$\leq$ 3.5 psig	(a)	X	(a)	(a)
1	(2) Low-Low Reactor Water Level	2	2	$\geq$5 inches. (Indicator scale)	X	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

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 and	2	2	$\leq$ 3.5 psig	X	X	X	X
1 b. Low-Low Reactor Water Level	2	2	>5 inches. (Indicator scale)	X	X	X	X

Table 3.6.2f

INSTRUMENTATION THAT INITIATES AUTO DEPRESSURIZATION

Limiting Condition for Operation

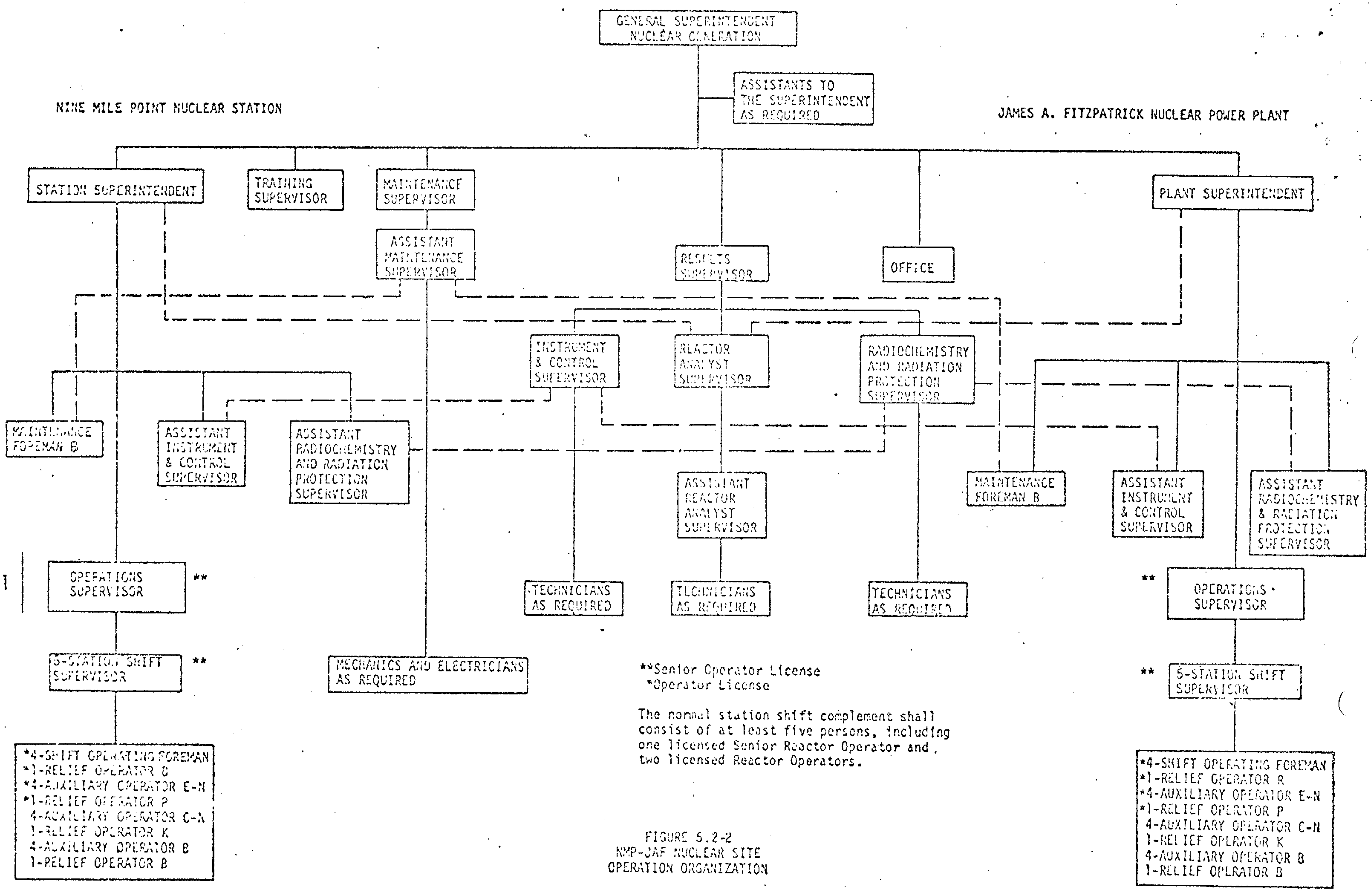
	Parameter	Minimum No. of Operable Trip Systems	Minimum No. of Operable Instrument Channels Per Operable Trip System	Set Point	Reactor Mode Switch Position in Which Function Must Be Operable			
					Shutdown	Refuel	Startup	Run
	INITIATION							
1	(1)a. Low-Low-Low Reactor Water Level	2 (a)	2 (a)	>127.1 inches (Indicator scale)	X	X	X	X
	and							
	b. High Drywell	2 (a)	2 (a)	3.5 psig	X	X	X	X

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	(1) Low Reactor Water Level	2	2	$\geq$53 inches. (Indicator scale)	X	X	X	X
	(2) Automatic Turbine Trip	1	1	- -			X	X



**Senior Operator License
*Operator License

The normal station shift complement shall consist of at least five persons, including one licensed Senior Reactor Operator and two licensed Reactor Operators.

FIGURE 6.2-2
NMP-JAF NUCLEAR SITE
OPERATION ORGANIZATION

Table 6.2-1

MINIMUM SHIFT CREW COMPOSITION [1]

	License	Normal Operation	Shutdown Condition	Operation W/O Process Computer [3]	Reactor [4] Startups
	Senior Operator	1	1	1	1
1	Operator	2	1	2	3
	Unlicensed [2]	2	1	3	2

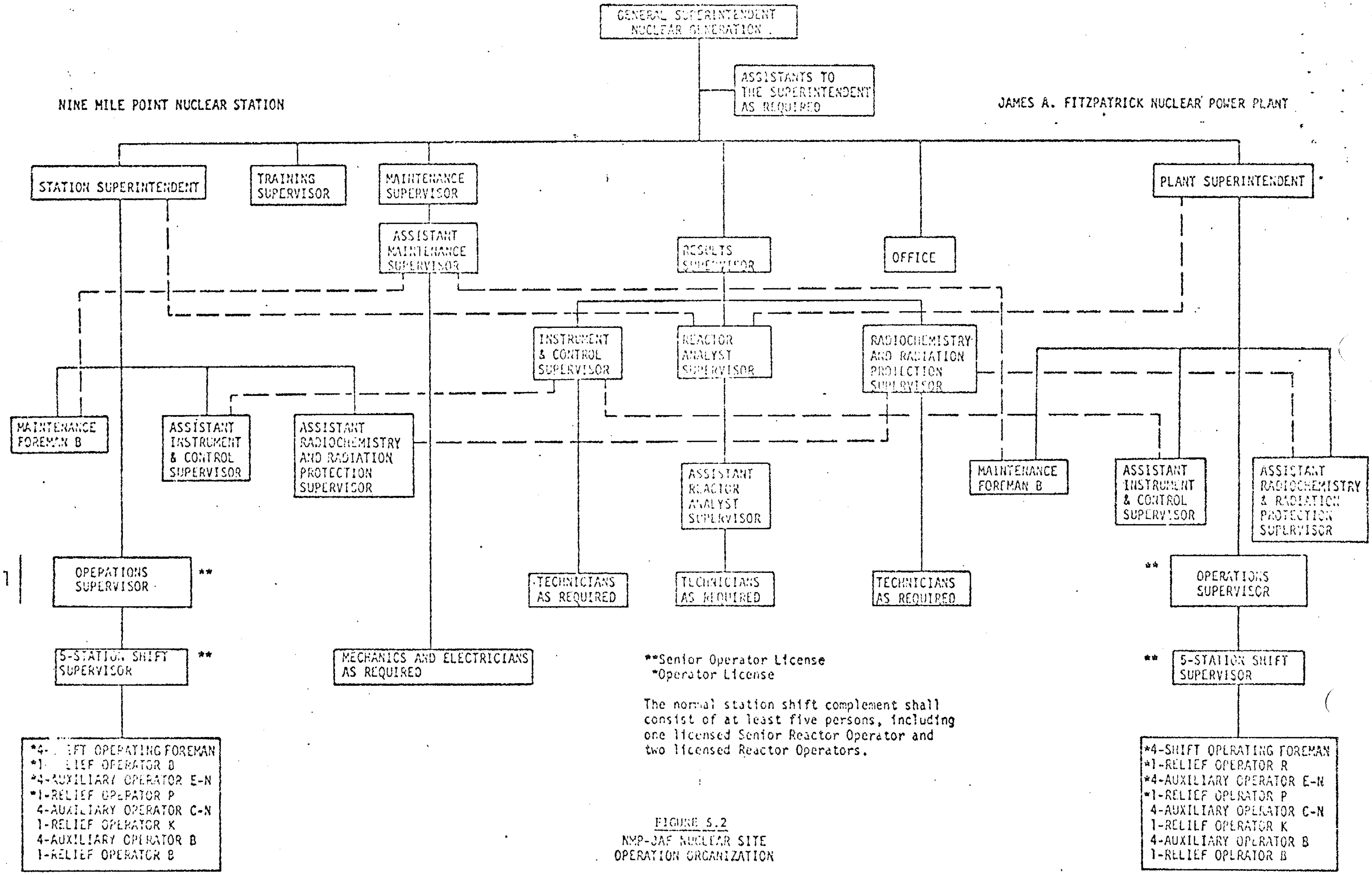
Notes:

(1) At any one time more licensed or unlicensed operating people could be present for maintenance, repairs, fuel outages, etc.

(2) Those operating personnel not holding an "Operator" or "Senior Operator" License.

(3) For operation longer than eight hours without process computer.

(4) For reactor startups, except a scram recovery where the reason for scram is both clearly understood and corrected.



**Senior Operator License
*Operator License

The normal station shift complement shall consist of at least five persons, including one licensed Senior Reactor Operator and two licensed Reactor Operators.

FIGURE 5.2
NMP-JAF NUCLEAR SITE
OPERATION ORGANIZATION

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 1 TO FACILITY OPERATING LICENSE NO. DPR-63

(CHANGE NO. 1 TO TECHNICAL SPECIFICATIONS)

NINE MILE POINT UNIT 1

NIAGARA MOHAWK POWER CORPORATION

DOCKET NO. 50-220

Introduction

By letters dated January 21 and February 27, 1975 and by a supplemental letter dated March 19, 1975, Niagara Mohawk Power Corporation requested changes to the Technical Specifications appended to Facility Operating License No. DPR-63 for the Nine Mile Point Nuclear Station Unit 1. The proposed changes include:

1. Modification of the method currently used in determining reactor water level from the installed instrumentation to provide for direct use of the instrumentation reading rather than for conversion of the instrumentation reading to a frame of reference based on elevation above sea level.

2. A change in the low-low-low water level setpoint from 128" to 127.1" as indicated on the installed instrumentation.

3. Revision of the statements in the Basis regarding the performance requirements for the containment spray raw water cooling system.

4. A change in the position title of the "Assistant to the Superintendent for Operation" to "Operations Supervisor", and the addition of an "Assistant Maintenance Supervisor" position to the Station organization. These proposed changes would be made in both Appendices A and B to Facility Operating License No. DPR-63.

5. A change in the description of the access penetrations in the reactor building railroad bay to reflect the current structural design of the reactor building.

6. Correction of typographical errors made in the new Technical Specifications which were reissued concurrently with the conversion of the operating license from the Provisional Operating License DPR-17 to the Full Term Operating License DPR-63.

Discussion

The proposed changes identified in the first four items above were submitted by the licensee in a letter dated January 21, 1975. Clarification of the wording in the submittal was provided by a subsequent letter dated March 19, 1975. The remaining changes, identified as items 5 and 6 above, were submitted by the licensee in a letter dated February 27, 1975.

In addition to the licensee's proposed changes, we have included with this license amendment a new cover page to better identify Appendix A to Facility Operating License No. DPR-63 as "Radiological Technical Specifications".

Evaluation

Our evaluation of the proposed changes is as follows:

1. Reactor Water Level Instrumentation

   The various set points for reactor water level that are presently included in the Technical Specifications use Elevation 302'9" above sea level as a frame of reference or datum plane for measuring reactor water level. The proposed change would use water level indicator settings that are read directly from the installed water level instrumentation without reference to a specific elevation above sea level as a datum. Since this proposal involves only the method of using currently available and installed instrumentation and does not change any of the specified requirements for the amount of water to be present in the reactor vessel for the various limiting conditions of operation, we have concluded that the proposed change does not alter any of the safety provisions of the Technical Specifications and is, therefore, acceptable.

2. Reactor Water Level Low-Low-Low Setpoint

   A fuel cladding safety limit exists at the low-low-low reactor water level setpoint to assure that sufficient water level is maintained above the top of the active fuel to provide adequate cooling and decay heat removal. The reactor water level at the low-low-low level setpoint is approximately 4'8" above the top of the active fuel. The proposed change to the actual setting of the low-low-low level setpoint would correct an error which was made in the original setting. The initial setpoint was 7'11" below elevation 302'9" which is equivalent to 128" on the installed instrumentation indicator scale. The corrected setpoint would be 127.1" on the indicator scale. Since the actual water level in the reactor vessel is inversely proportional to the magnitude of the indicator scale reading (i.e., the higher the indicator scale reading, the lower the actual water level in the reactor vessel and vice versa), a change in the setpoint from 128" to 127.1" effectively increases the amount of water present in the reactor vessel at the low-low-low water level setpoint. Therefore, the proposed change would correct an existing error and would slightly increase the margin of safety. In addition, the change does not invalidate the results of current transient analyses since an allowable deviation of ± 2.6 inches in set point error is assumed in the calculations.

3. Containment Spray Pumping System

Although the proposed change to the Bases is not associated with a related change to the limiting conditions for operation or the surveillance requirements for the containment spray raw water cooling system, the pumping characteristics used to define the system operability requirements, which the licensee must adhere to, appear only in the Bases.

The proposed change would revise the information in the Bases related to the minimum pump performance requirements for the containment spray raw water cooling system by replacing the requirement for a system flow rate of 3000 gpm at a developed pump head of 540 feet of water with a requirement for a 3000 gpm minimum flow rate and a minimum system pressure of 160 psig on the raw water side of the containment spray heat exchangers. As stated in the Safety Analysis in the FSAR, the design of the containment spray raw water cooling system requires a system flow rate of 3000 gpm and a minimum raw water pressure of 160 psig at the heat exchangers (1) to maintain the containment spray water temperature below 140°F under the most limiting conditions, and (2) to assure that any leakage at the heat exchanger tube to shell interface would be into the containment spray system, thereby precluding potential radioactive contamination of the raw water system. Since the proposed change assures that the system design requirements are satisfied, we conclude that it is acceptable.

4. Station Organization

The proposed change in the Nine Mile Point Nuclear Station Unit 1 organization involves a change in a personnel position title and an addition of a position to the personnel roster. The proposed change in the position title of the "Assistant to the Superintendent for Operation" to "Operations Supervisor" does not change the function of the position and is consistent with ANSI N18.1-1971, "Selection and Training of Nuclear Power Plant Personnel".

The other proposed change involves the addition of an "Assistant Maintenance Supervisor" position to the Station operating organization. The establishment of this additional position reflects the increase in the maintenance workload resulting from the operation of two reactor facilities at the same site.

These changes involve upgrading and editing of the Technical Specifications and are acceptable.

5. Reactor Building Access Control

The current Technical Specification for access control to the reactor building railroad bay reflects the original design of the reactor building which provided a door and a hatch arrangement to meet the minimum requirement for two access penetrations in series to assure the integrity of the secondary containment system. The proposed change reflects a recent modification of the reactor building design, (i.e., the addition of a reactor building extension), which provided an additional access door to the railroad bay. We have previously reviewed this modification of the reactor building and found it acceptable as set forth in our letter to NMPC dated April 12, 1974. This new access door, in combination with the original access door, assures compliance with the requirement for two reactor building railroad bay penetrations in series; and, as a result, access control requirements associated with the railroad bya hatch are no longer necessary. Therefore, since the hatch has been functionally replaced by an additional access door, we find the proposed change to be acceptable.

6. Typographical Error Corrections

Two typographical errors have been identified within Appendix A; these errors were made with the reissuance of the Technical Specifications during the conversion of Provisional Operating License No. DPR-17 to Full-Term Operating License No. DPR-63. These errors and corrections are as follows: (1) the pressure at which the primary containment testable penetrations and isolation valves are tested should have read 35 psig instead of 36 psig on page 138 of Appendix A, and (2) on Table 6.2-1, page 248 of Appendix A, "Minimum Shift Crew Composition", in the column headed by "Operation Without Process Computer", the numbers 2 and 3 should be transposed to require two operators and three unlicensed personnel.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the changes do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the changes do 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.

OCT 1 1975

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-220

NIAGARA MOHAWK POWER CORPORATION

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY

OPERATING LICENSE

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

The amendment modifies the Technical Specifications relating to the reactor water level instrumentation, the low-low-low water level setpoints, description of the access penetrations in reactor building railroad bay, changes in position titles in the station operating organization, and to correct typographical errors.

The applications for the amendment comply 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 is not required since the amendment does not involve a significant hazards consideration.

For further details with respect to this action, see (1) the applications for amendment dated January 21 and February 27, 1975, and supplement dated March 19, 1975, (2) Amendment No. 1 to License No. DPR-63, with Change No. 1, 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, N. W., Washington, D. C. and at the Oswega City Library, 120 E. Second 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 Reactor Licensing.

Dated at Bethesda, Maryland, this 1st day of October, 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

George Lear

George Lear, Chief
Operating Reactors Branch #3
Division of Reactor Licensing