June 2, 1999

Mr. Michael B. Roche
Vice President and Director
GPU Nuclear, Inc.
Oyster Creek Nuclear Generating Station
P.O. Box 388
Forked River, NJ 08731

SUBJECT: OYSTER CREEK NUCLEAR GENERATING STATION - ISSUANCE OF AMENDMENT RE: CHANGE REQUEST RELATED TO THE APRM/LPRM SAFETY LIMITS AND SURVEILLANCE REQUIREMENTS (TAC NO. MA4145)

Dear Mr. Roche:

The Commission has issued the enclosed Amendment No. 208 to Facility Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station, in response to your application dated November 5, 1998, as supplemented by your letter dated February 18, 1999.

The amendment will modify the safety limits and surveillances of the local-power range monitor (LPRM) and average-power range monitor (APRM) systems and related Bases pages to ensure the APRM channels respond within the necessary range and accuracy and to verify channel operability. In addition, an unrelated change to the Bases of Specification 2.3 is included to clarify some ambiguous language.

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

Sincerely,

ORIGINAL SIGNED BY:

Ronald B. Eaton, Senior Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-219

Enclosures: 1. Amendment No208to DPR-16
2. Safety Evaluation

cc w/encls: See next page

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

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Ronald B. Eaton, Senior Project Manager, Section 2
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Office of Nuclear Reactor Regulation

Docket No. 50-219

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New Jersey Department of
Environmental Protection
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CN 415
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DATED: June 2, 1999

AMENDMENT NO. 208 TO FACILITY OPERATING LICENSE NO. DPR-16-OYSTER CREEK

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

GPU NUCLEAR, INC.

AND

JERSEY CENTRAL POWER & LIGHT COMPANY

DOCKET NO. 50-219

OYSTER CREEK NUCLEAR GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 208
License No. DPR-16

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

   A. The application for amendment by GPU Nuclear, Inc. et al., (the licensee), dated November 5, 1998, as supplemented by a letter dated February 18, 1999, 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-16 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No208 , are hereby incorporated in the license. GPU Nuclear, Inc. shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

James W. Clifford

James W. Clifford, Chief, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: June 2, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 208

FACILITY OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

Replace the following pages of the Appendix A, Technical Specifications, with the attached revised pages as indicated. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
2.3-1	2.3-1
2.3-2	2.3-2
2.3-3	2.3-3
2.3-4	2.3-4
?.3-5	2.3-5
2.3-6	2.3-6
2.3-7	2.3-7
------	2.3-8
3.1-9	3.1-9
3.1-10	3.1-10
3.1-11	3.1-11
3.1-12	3.1-12
3.1-13	3.1-13
3.1-14	3.1-14
3.1-15	3.1-15
3.1-16	3.1-16
3.1-16a	--------
--------	3.1-17
3.1-18	3.1-18
3.1-19	3.1-19
3.1-20	3.1-20
--------	3.1-21
4.1-1	4.1-1
4.1-2	4.1-2
4,1-3	4.1-3
4.1-4	4.1-4
4.1-5	4.1-5
4.1-6	4.1-6
4.1-7	4.1-7
-------	4.1-8
-------	4.1-9
-------	4.1-10

2.3 LIMITING SAFETY SYSTEM SETTINGS

Applicability: Applies to trip settings on automatic protective devices related to variables on which safety limits have been placed.

Objective: To provide automatic corrective action to prevent the safety limits from being exceeded.

Specification: Limiting safety system settings shall be as follows:

FUNCTION	LIMITING SAFETY SYSTEM SETTINGS

A. Neutron Flux, Scram

A.1 APRM

When the reactor mode switch is in the Run position, the APRM flux scram setting shall be

$$S \leq [(0.90 \times 10^{-6}) W + 60.8] \frac{FRP}{MFLPD}$$

with a maximum setpoint of 115.7% for core flow equal to 61 x 10^6 lb/hr and greater,

where:

S = setting in percent of rated power
W = recirculation flow (lb/hr)

FRP = fraction of rated thermal power is the ratio of core thermal power to rated thermal power

MFLPD = maximum fraction of limiting power density where the limiting power density for each bundle is the design linear heat generation rate for that bundle.

The ratio of FRP/MFLPD shall be set equal to 1.0 unless the actual operating value is less than 1.0 in which case the actual operating value will be used.

This adjustment may be accomplished by increasing the APRM gain and thus reducing the flow reference APRM High Flux Scram Curve by the reciprocal of the APRM gain change.

A.2 IRM — ≤ 38.4 percent of rated neutron flux

A.3 APRM Downscale — ≥ 2% Rated Thermal Power coincident with IRM Upscale (high-high) or Inoperative

	FUNCTION	LIMITING SAFETY SYSTEM SETTINGS
B.	Neutron Flux, Control Rod Block	The Rod Block setting shall be $S \le [(0.90 \times 10^{-6}) W + 53.1] \left[\frac{FRP}{MFLPD}\right]$ with a maximum setpoint of 108% for core flow equal to 61 x 10^6 lb/hr and greater. The definitions of S, W, FRP and MFLPD used above for the APRM scram trip apply. The ratio of FRP to MFLPD shall be set equal to 1.0 unless the actual operating value is less than 1.0, in which case the actual operating value will be used. This adjustment may be accomplished by increasing the APRM gain and thus reducing the flow referenced APRM rod block curve by the reciprocal of the APRM gain change.
C.	Reactor High Pressure, Scram	≤1060 psig
D.	Reactor High Pressure, Relief Valves Initiation	2 @ ≤ 1085 psig 3 @ ≤ 1105 psig
E.	Reactor High Pressure, Isolation Condenser Initiation	≤1060 psig with time delay ≤3 seconds
F.	Reactor High Pressure, Safety Valve Initiation	4 @ 1212 psig ±12 psi 5 @ 1221 psig ±12 psi
G.	Low Pressure Main Steam MSIV Closure	≥ 825 psig (initiated in IRM Line, range 10)
H.	Main Steam Line Isolation Valve Closure, Scram	≤10% Valve Closure from full open

	FUNCTION	LIMITING SAFETY SYSTEM SETTINGS
I.	Reactor Low Water Level, Scram	≥11'5" above the top of the active fuel as indicated under normal operating conditions
J.	Reactor Low-Low Water Level, Main Steam Line Isolation Valve Closure	≥7'2" above the top of the active fuel as indicated under normal operating conditions
K.	Reactor Low-Low Water Level, Core Spray Initiation	≥7'2" above the top of the active fuel
L.	Reactor Low-Low Water Level, Isolation Condenser Initiation	≥7'2" above the top of the active Fuel with time delay ≤3 seconds
M.	Turbine Trip, Scram	10 percent turbine stop valve(s) closure from full open
N.	Generator Load Rejection, Scram	Initiate upon loss of oil pressure from turbine acceleration relay
O.	DELETED	
P.	Loss of Power	
	1) 4.16 KV Emergency Bus Undervoltage (Loss of Voltage)	0 volts with 3 seconds ± 0.5 seconds time delay
	2) 4.16 KV Emergency Bus Undervoltage (Degraded Voltage)	3840 (+20V, -40V) volts 10 ± 10% (1.0) second time delay

2.3 LIMITING SAFETY SYSTEM SETTINGS

Bases:

Safety limits have been established in Specifications 2.1 and 2.2 to protect the integrity of the fuel cladding and reactor coolant system barriers, respectively. Automatic protective devices have been provided in the plant design for corrective actions to prevent the safety limits from being exceeded in normal operation or operational transients caused by reasonably expected single operator error or equipment malfunction. This Specification establishes the trip settings for these automatic protection devices.

The Average Power Range Monitor, APRM[1], trip setting has been established to assure never reaching the fuel cladding integrity safety limit. The APRM system responds to changes in neutron flux. However, near the rated thermal power, the APRM is calibrated using a plant heat balance, so that the neutron flux that is sensed is read out as percent of the rated thermal power. For slow maneuvers, such as those where core thermal power, surface heat flux, and the power transferred to the water follow the neutron flux, the APRM will read reactor thermal power. For fast transients, the neutron flux will lead the power transferred from the cladding to the water due to the effect of the fuel time constant. Therefore, when the neuron flux increases to the scram setting, the percent increase in heat flux and power transferred to the water will be less than the percent increase in neutron flux.

The APRM trip setting will be varied automatically with recirculation flow, with the trip setting at the rated flow of 61.0 x 10^6 lb/hr of greater being 115.7% of rated neutron flux. Based on a complete evaluation of the reactor dynamic performance during normal operation as well as expected maneuvers and the various mechanical failures, it was concluded that sufficient protection is provided by the simple fixed scram setting (2,3). However, in response to expressed beliefs (4) that variation of APRM flux scram with recirculation flow is a prudent measure to ensure safe plant operation, the scram setting will be varied with recirculation flow.

An increase in the APRM scram trip setting would decrease the margin present before the fuel cladding integrity safety limit is reached. The APRM scram trip setting was determined by an analysis of margins required to provide a reasonable range for maneuvering during operation. Reducing this operating margin would increase the frequency of spurious scrams, which have an adverse effect on reactor safety because of the resulting thermal stresses. Thus, the APRM scram trip setting was selected because it provides adequate margin for the fuel cladding integrity safety limit and yet allows operating margin that reduces the possibility of unnecessary scrams.

The scram trip setting must be adjusted to ensure that the LHGR transient peak is not increased for any combination of maximum fraction of limiting power density (MFLPD) and reactor core thermal power. The scram setting is adjusted in accordance with the formula in Specification 2.3.A, when the MFLPD is greater than the fraction of the rated power (FRP). the adjustment may be accomplished by increasing the APRM gain and thus reducing the flow referenced APRM High Flux Scram Curve by the reciprocal of the APRM gain change.

For operation in the Startup mode while the reactor is at low pressure, the IRM range 9 High Flux scram setting of 12% of the rated power provides adequate thermal margin between the maximum power and the safety limit of 18.3% of rated power to accommodate anticipated maneuvers associated with power plant startup. There are a few possible sources of rapid reactivity input to the system in the low power/low flow condition. Effects of increasing pressure at zero or low void content are minor, because cold water from sources available during the startup is not much colder than that already in the system, temperature coefficients are small, and control rod sequences are constrained by operating procedures backed up by the rod worth minimizer. Worth of individual rods is very low in a constrained rod pattern. In a sequenced rod withdrawal approach to the scram level, the rate of power rise is no more than five percent of the rated per minute, and the IRM system would be more than adequate to assure a scram before the power could exceed the safety limit.

To continue operation beyond 12% of rated power, the IRMs must be transferred into range 10. The Reactor Protection System is designed such that reactor pressure must be above 825 psig to successfully transfer the IRMs into range 10, thus assuring protection for the fuel cladding safety limit. The IRM scram remains active until the mode switch is placed in the RUN position at which time the trip becomes a coincident IRM upscale, APRM downscale scram.

The adequacy of the IRM scram was determined by comparing the scram level on the IRM range 10 to the scram level on the APRMs at 30% of rated flow. The IRM scram is at 38.4% of rated power while the APRM scram is at 52.7% of rated power. The minimum flow for Oyster Creek is at 30% of rated power and this would be the lowest APRM scram point. The increased recirculation flow to 65% of flow will provide additional margin to CPR Limits. The APRM scram at 65% of rate flow is 87.1% of rated power, while the IRM range 10 scram remains at 38.4% of rated power. Therefore, transients requiring a scram based on flux excursion will be terminated sooner with a IRM range 10 scram than with an APRM scram. The transients requiring a scram by nuclear instrumentation are the loss of feedwater heating and the improper startup of an idle recirculation loop. The loss of feedwater heating transient is not affected by the range 10 IRM since the feedwater heaters will not be put into service until after the LPRM downscales have cleared, thus insuring the operability of the APRM system. This will be administratively controlled. The improper startup of an idle recirculation loop becomes less severe at lower power level and the IRM scram would be adequate to terminate the flux excursion.

The Rod Worth Minimizer is not required beyond 10% of rated power. The ability of the IRMs to terminate a rod withdrawal transient is limited due to the number and location of IRM detectors. An evaluation was performed that showed by maintaining a minimum recirculation flow of 39.65×10^6 lb/hr in range 10 a complete rod withdrawal initiated at 35% of rated power or less would not result in violating the fuel cladding safety limit. Therefore, a rod block on the IRMs at less than 35% of rated power would be adequate protection against a rod withdrawal transient.

Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent gross rod withdrawal at constant recirculation flow rate to protect against grossly exceeding the MCPR Fuel Cladding Integrity Safety Limit. This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the safety limit increases as the flow decreases for the specified trip setting versus flow relationship. Therefore, the worst-case MCPR, which could occur during steady-state operation, is at 108% of the rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored continuously by the incore LPRM system. As with APRM scram trip setting, the APRM rod block trip setting is adjusted downward if the maximum fraction of limiting power density exceeds the fraction of the rated power, thus preserving the APRM rod block safety margin. As with the scram setting, this may be accomplished by adjusting the APRM gains.

The settings on the reactor high pressure scram, anticipatory scrams, reactor coolant system relief valves and isolation condenser have been established to assure never reaching the reactor coolant system pressure safety limit as well as assuring the system pressure does not exceed the range of the fuel cladding integrity safety limit. In addition, the APRM neutron flux scram and the turbine bypass system also provide protection for these safety limits, e.g., turbine trip and loss of electrical load transients (5). In addition to preventing power operation above 1060 psig, the pressure scram backs up the other scrams for these transients and other steam line isolation type transients. Actuation of the isolation condenser during these transients removes the reactor decay heat without further loss of reactor coolant thus protecting the reactor water level safety limit.

The reactor coolant system safety valves offer yet another protective feature for the reactor coolant system pressure safety limit since these valves are sized assuming no credit for other pressure relieving devices. In compliance with Section I of the ASME Boiler and Pressure Vessel Code, the safety valve must be set to open at a pressure no higher than 103% of design pressure, and they must limit the reactor pressure to no more than 110% of design pressure. The safety valves are sized according to the Code for a condition of main steam isolation valve closure while operating at 1930 MWt, followed by (1) a reactor scram on high neutron flux, (2) failure of the recirculation pump trip on high pressure, (3) failure of the turbine bypass valves to open, and (4) failure of the isolation condensers and relief valves to operate. Under these conditions, a total of 9 safety valves are required to turn the pressure transient. The ASME B&PV Code allows a ±1% of working pressure (1250 psig) variation in the lift point of the valves. This variation is recognized in Specification 4.3.

The low pressure isolation of the main steam line at 825 psig was provided to give protection against fast reactor depressurization and the resulting rapid cool-down of the vessel. The low-pressure isolation protection is enabled with entry into IRM range 10 or the RUN mode. In addition, a scram on 10% main steam isolation valve (MSIV) closure anticipates the pressure and flux transients which occur during normal or inadvertent isolation valve closure. Bypass of the MSIV closure scram function below 600 psig is permitted to provide sealing steam and allow the establishment of condenser vacuum. Advantage is taken of the MSIV scram feature to provide protection for the low-pressure portion of the fuel cladding integrity safety limit. To continue operation beyond 12% of rated power, the IRM's must be transferred into range 10. Reactor pressure must be above 825 psig to successfully transfer the IRM's into range 10. Entry into range 10 at less than 825 psig will result in main steam line isolation valve closure and MSIV closure scram. This provides automatic scram protection for the fuel cladding integrity safety limit which allows a maximum power of 25% of rated at pressures below 800 psia. Below 600 psig, when the MSIV closure scram is bypassed, scram protection is provided by the IRMs.

Operation of the reactor at pressure lower than 825 psig requires that the mode switch be in the STARTUP position and the IRMs be in range 9 or lower. The protection for the fuel clad integrity safety limit is provided by the IRM high neutron flux scram in each IRM range. The IRM range 9 high flux scram setting at 12% of rated power provides adequate thermal margin to the safety limit of 25% of rated power. There are few possible significant sources of rapid reactivity input to the system through IRM range 9: effects of increasing pressure at zero and low void content are minor; reactivity excursions from colder makeup water, will cause an IRM high flux trip; and the control rod sequences are constrained by operating procedures backed up by the rod worth minimizer. In the unlikely event of a rapid or uncontrolled increase in reactivity, the IRM system would be more than adequate to ensure a scram before power could exceed the safety limit. Furthermore, a mechanical stop on the IRM range switch requires an operator to pull up on the switch handle to pass through the stop and enter range 10. This provides protection against an inadvertent entry into range 10 at low pressures. The IRM scram remains active until the mode switch is placed in the RUN position at which time the trip becomes a coincident IRM upscale, APRM downscale scram.

The low level water level trip setting of 11'5" above the top of the active fuel has been established to assure that the reactor is not operated at a water level below that for which the fuel cladding integrity safety limit is applicable. With the scram set at this point, the generation of steam, and thus the loss of inventory is stopped. For example, for a loss of feedwater flow a reactor scram at the value indicated and isolation valve closure at the low-low water level set point results in more than 4 feet of water remaining above the core after isolation (6).

During periods when the reactor is shut down, decay heat is present and adequate water level must be maintained to provide core cooling. Thus, the low-low level trip point of 7'2" above the core is provided to actuate the core spray system (when the core spray system is required as identified in Section 3.4) to provide cooling water should the level drop to this point.*

The turbine stop valve(s) scram is provided to anticipate the pressure, neutron flux, and heat flux increase caused by the rapid closure of the turbine stop valve(s) and failure of the turbine bypass system.

*Correction 11/30/87

The generator load rejection scram is provided to anticipate the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves to a load rejection and failure of the turbine bypass system. This scram is initiated by the loss of turbine acceleration relay oil pressure. The timing for this scram is almost identical to the turbine trip.

The undervoltage protection system is a 2 out of 3 coincident logic relay system designed to shift emergency buses C and D to on-site power should normal power be lost. There is a separate 2 out of 3 coincident logic relay designed to shift emergency buses to on-site power should normal power be degraded to an unacceptable level. The trip points and time delay settings have been selected to assure an adequate power source to emergency safeguards systems in the event of a total loss of normal power or degraded conditions which would adversely affect the functioning of engineered safety features connected to the plant emergency power distribution system.

The APRM downscale signal insures that there is adequate Neutron Monitoring System protection if the reactor mode switch is placed in the run position prior to APRMs coming on scale. With the reactor mode switch in run, an APRM downscale signal coincident with an associate IRM Upscale (High-High) or Inoperative signal generates a trip signal. This function is not specifically credited in the accident analyses but it is retained for overall redundancy and diversity of the RPS.

References

(1) FDSAR, Volume 1, Section VII-4.2.4.2

(2) FDSAR, Amendment 28, Item III.A-12

(3) FDSAR, Amendment 32, Question 13

(4) Letters, Peter A. Morris, Director, Division of Reaction Licensing, USAEC, to John E. Logan, Vice President, Jersey Central Power and Light Company

(5) FDSAR, Amendment 65, Section B.XI

(6) FDSAR, Amendment 65, Section B.IX

TABLE 3.1.1
Sheet 1 of 13

PROTECTIVE INSTRUMENTATION REQUIREMENTS

		Reactor Modes in Which Function Must Be Operable				Minimum Number of OPERABLE or OPERATING [tripped]	Minimum Number of Instrument Channels Per OPERABLE	
Function	Trip Setting	Shutdown	Refuel	Startup	Run	Trip Systems	Trip System	Action Required*
A. Scram								
1. Manual Scram		X	X	X	X	2	1	Insert control rods
2. High Reactor Pressure	**		X(s)	X(ll)	X	2	2(nn)	
3. High Drywell Pressure	≤ 3.5 psig		X(u)	X(u)	X	2	2(nn)	
4. Low Reactor Water Level	**		X	X	X	2	2(nn)	
5. a. High Water Level in Scram Discharge Volume North Side	≤ 29 gal.		X(a)	X(z)	X(z)	2	2(nn)	
b. High Water Level in Scram Discharge Volume South Side	≤ 29 gal.		X(a)	X(z)	X(z)	2	2(nn)	
6. Low Condenser Vacuum	≥ 20 in. hg.			X(b)	X	1	3(mm)(nn)	
7. DELETED								

PROTECTIVE INSTRUMENTATION REQUIREMENTS

Function	Trip Setting	Reactor Modes in Which Function Must Be Operable				Minimum Number of OPERABLE or OPERATING [tripped] Trip Systems	Minimum Number of Instrument Channels Per OPERABLE Trip System	Action Required*
		Shutdown	Refuel	Startup	Run			
8. Average Power Range Monitor (APRM)	**		X(c,s)	X(c)	X(c)	2	3(nn)	
9. Intermediate Range Monitor (IRM)	**		X(d)	X(d)		2	3(nn)	
10. Main Steamline Isolation Valve Closure	**		X(b,s)	X(b)	X	2	4(nn)	
11. Turbine Trip Scram	**				X(j)	2	4(nn)	
12. Generator Load Rejection Scram	**				X(j)	2	2(nn)	
13. APRM Downscale/IRM Upscale	**				X(c)	2	3(nn)	
Reactor Isolation								
1. Low-Low Reactor Water Level	**	X	X	X	X	2	2(oo)	Close Main Steam Isolation Valves and Close Isolation Condenser Vent Valve or,
2. High Flow in Main Steamline A	≤120% rated	X(s)	X(s)	X	X	2	2(oo)	PLACE IN COLD SHUTDOWN

TABLE 3.1.1
Sheet 3 of 13

PROTECTIVE INSTRUMENTATION REQUIREMENTS

Function	Trip Setting	Reactor Modes in which Function Must Be OPERABLE: Shutdown	Refuel	Startup	Run	Minimum Number of OPERABLE or OPERATING [tripped] Trip Systems	Minimum Number of Instrument Channels Per OPERABLE Trip System	Action Required*
3. High Flow in Main Steam-line B	≤120% rated	X(s)	X(s)	X	X	2	2(oo)	
4. High Temperature in Main Steamline Tunnel	≤Ambient at Power + 50°F	X(s)	X(s)	X	X	2	2(oo)	
5. Low Pressure in Main Steamline	**			X(cc)	X	2	2(oo)	
6. DELETED								
C. Isolation Condenser Initiation								
1. High Reactor Pressure	**	X(s)	X(s)	X(ll)	X	2	2(pp)	PLACE IN COLD
2. Low-Low Reactor Water Level	≥7'2" above TOP of ACTIVE FUEL	X(s)	X(s)	X	X	2	2(pp)	SHUTDOWN CONDITION
D. Core Spray								
1. Low-Low Reactor Water Level	**	X(t)	X(t)	X(t)	X	2	2(pp)	Consider the respective core spray loop
2. High Drywell Pressure	≤ 3.5 psig	X(t)	X(t)	X(t)	X	2(k)	2(k)(pp)	inoperable and comply with
3. Low Reactor Pressure (valve permissive)	≥ 285 psig	X(t)	X(t)	X(t)	X	2	2(pp)	Spec 3.4

TABLE 3.1.1 PROTECTIVE INSTRUMENTATION REQUIREMENTS
Sheet 4 of 13

Function	Trip Setting	Reactor Modes in which Function Must Be OPERABLE: Shutdown	Refuel	Startup	Run	Minimum Number of OPERABLE or OPERATING [tripped] Trip Systems	Minimum Number of Instrument Channels Per OPERABLE Trip System	Action Required*
E. Containment Spray								
Comply with Technical Specification 3.4								
F. Primary Containment Isolation								
1. High Drywell Pressure	≤ 3.5 psig	X(u)	X(u)	X(u)	X	2(k)	2(k)(oo)	Isolate containment or PLACE IN COLD SHUTDOWN CONDITION
2. Low-Low Reactor Water Level	≥ 7'2" above TOP of ACTIVE FUEL	X(u)	X(u)	X(u)	X	2	2(oo)	
G. Automatic Depressurization								
1. High Drywell Pressure	≤ 3.5 psig	X(v)	X(v)	X(v)	X	2(k)	2(k)	See note h
2. Low-Low-Low Reactor Water Level	≥4'8" above TOP of ACTIVE FUEL	X(v)	X(v)	X(v)	X	2	2	See note h
3. Core Spray Booster Pump d/p Permissive	>21.2 psid	X(v)	X(v)	X(v)	X	Note i	Note i	See note I
H. Isolation Condenser Isolation (See Note hh)								
1. High Flow Steam Line	≤ 20 psig P	X(s)	X(s)	X	X	2	2(oo)	Isolate affected Isolation Condenser comply with Spec 3.8. See note dd
2. High Flow Condensate Line	≤ 27" P H_20	X(s)	X(s)	X	X	2	2(oo)	

Function	Trip Setting	Reactor Modes in which Function Must Be OPERABLE: Shutdown	Refuel	Startup	Run	Minimum Number of OPERABLE or OPERATING [tripped] Trip Systems	Minimum Number of Instrument Channels Per OPERABLE Trip System	Action Required*
I. Offgas System Isolation								
1. High Radiation In Offgas Line (e)	≤ 2000 mRem/hr	X(s)	X(s)	X	X	1 (ii)	2(ii)	See note jj
J. Reactor Building Isolation and Standby Gas Treatment System Initiation								
1. High Radiation Reactor Building Operating Floor	≤ 100 mR/hr	X(w)	X(w)	X	X	1	1	Isolate Reactor Building and Initiate Standby Gas Treatment System or Manual Surveillance for not more than 24 Hours (Total for all instruments under J) in any 30-day period.
2. Reactor Building Ventilation Exhaust	≤ 17 mR/hr	X(w)	X(w)	X	X	1	1	
3. High Drywell Pressure	≤ 3.5 psig	X(u)	X(u)	X	X	1(k)	2(k)	
4. Low-Low Reactor Water Level	≥ 7'2" above TOP of ACTIVE FUEL	X	X	X	X	1	2	
K. Rod Block								
1. SRM Upscale	≤ 5x10[5] cps		X	X(1)		1	2	No control rod withdrawals permitted
2. SRM Downscale	≥ 100cps(f)		X	X(1)		1	2	
3. IRM Downscale	≥ 5/125 fullscale (g)		X	X		2	3	

Function	Trip Setting	Reactor Modes in which Function Must Be OPERABLE: Shutdown	Refuel	Startup	Run	Minimum Number of OPERABLE or OPERATING [tripped] Trip Systems	Minimum Number of Instrument Channels Per OPERABLE Trip System	Action Required*
K. Rod Block (Cont'd.)								
4. APRM Upscale	**		X(s)	X	X	2	3(c)	
5. APRM Downscale	≥ 2/150 fullscale				X	2	3 (c)	
6. IRM Upscale	≤ 108/125 fullscale		X	X		2	3	
7. a) Water Level High Scram Discharge Volume North	≤ 14 gallons		X(z)	X(z)	X(z)	1	1 per Instrument Volume	
b) Water Level High Scram Discharge Volume South	≤ 14 gallons		X(z)	X(z)	X(z)	1	1 per Instrument Volume	
L. Condenser Vacuum Pump Isolation								
Deleted								
M. Diesel Generator Load Sequence Timers	Time Delay after energization of relay							
1. CRD Pump	60 sec ± 15%	X	X	X	X	2(m)	1(n)(kk)	Consider the pump inoperable and comply with Spec 3.4.D (See note q)

TABLE 3.1.1 PROTECTIVE INSTRUMENTATION REQUIREMENTS

Function	Trip Setting	Reactor Modes in which Function Must Be OPERABLE: Shutdown	Refuel	Startup	Run	Minimum Number of OPERABLE or OPERATING [tripped] Trip Systems	Minimum Number of Instrument Channels Per OPERABLE Trip System	Action Required*
M. Diesel Generator Load Sequence Timers (Cont'd.)								
2. Service Water Pump	120 sec ± 15% (SK1A) (SK2A) 10 sec. ± 15% (SK7A) (SK8A)	X	X	X	X	2(o)	2(p)(kk)	Consider the pump inoperable and comply within 7 days (See note q)
3. Reactor Building Closed Cooling Water Pump (bb)	166 sec ± 15%	X	X	X	X	2(m)	1(n)(kk)	Consider the pump inoperable and comply within 7 days (See note q)
N. Loss of Power								
a. 4.16 KV Emergency Bus Undervoltage (Loss of Voltage)	**	X(ff)	X(ff)	X(ff)	X(ff)	2	1(kk)	
b. 4.16 KV Emergency Bus Undervoltage (Degraded Voltage)	**	X(ff)	X(ff)	X(ff)	X(ff)	2	3(kk)	See note ee
O. Containment Vent and Purge Isolation								
1. Drywell High Radiation	≤ 74.6 R/hr	X(u)	X(u)	X(u)	X	1	1	Isolate vent & purge pathways or PLACE IN COLD SHUTDOWN CONDITION

* Action required when minimum conditions for operation are not satisfied. Also permissible to trip inoperable trip system. A channel may be placed in an inoperable status for up to six hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE instrument channel in the same trip system is monitoring that parameter.

** See Specification 2.3 for Limiting Safety System Settings.

Notes:

a. Permissible to bypass, with control rod block, for reactor protection system reset in REFUEL MODE.

b. Permissible to bypass below 600 psig in REFUEL and STARTUP MODES.

c. One (1) APRM in each OPERABLE trip system may be bypassed or inoperable provided the requirements of Specification 3.1.C and 3.10.C are satisfied. Two APRM's in the same quadrant shall not be concurrently bypassed except as noted below or permitted by note.

Any one APRM may be removed from service for up to six hours for test or calibration without inserting trips in its trip system only if the remaining OPERABLE APRM's meet the requirements of Specification 3.1.B.1 and no control rods are moved outward during the calibration or test. During this short period, the requirements of Specifications 3.1.B.2, 3.1.C and 3.10.C need not be met.

d. The IRMs shall be inserted and OPERABLE until the APRMs are OPERABLE and reading at least 2/150 full scale.

e. Offgas system isolation trip set at ≤2,000 mRem/hr. Air ejector isolation valve closure time delay shall not exceed 15 minutes.

f. Unless SRM chambers are fully inserted.

g. Not applicable when IRM on lowest range.

h. With one or more instrument channel(s) inoperable in one ADS trip system, place the relay contact(s) for the inoperable initiation signal in the tripped condition within 4 days, or declare ADS inoperable and take the action required by Specification 3.4.B.3.

With one or more instrument channel(s) inoperable in both ADS trip systems, restore ADS initiation capability in at least one trip system within 1 hour, or declare ADS inoperable and take the action required by Specification 3.4.B.3.

Relief valve controllers shall not be bypassed for any more than 3 hours (total time for all controllers) in any 30-day period and only one relief valve controller may be bypassed at a time.

i. With two core spray systems OPERABLE:

1. A maximum of two core spray booster pump differential pressure (d/p) switches may be inoperable provided that the switches are in opposing ADS trip systems [i.e., only: either RV-40 A&D or RV-40 B&C]. Place the relay contacts associated with the inoperable d/p switch(es) in the de-energized position, within 24 hours. Restore the inoperable d/p switch(es) within 8 days, or declare ADS inoperable and take the action required by Specification 3.4.B.3;

or,

2. If two inoperable d/p switches are in the same ADS trip system [i.e., RV-40 A&B or RV-40 C&D], place the relay contacts associated with the inoperable d/p switch(es) in the de-energized position, within 24 hours. Restore the inoperable d/p switches within 4 days, or declare ADS inoperable and take the action required by Specification 3.4.B.3.

With only one core spray system OPERABLE:

If one or more d/p switches become inoperable in the OPERABLE core spray system, declare ADS inoperable and take the action required by Specification 3.4.B.3.

j. Not required below 40% of rated reactor thermal power.

k. All four (4) drywell pressure instrument channels may be made inoperable during the integrated primary containment leakage rate test (See Specification 4.5), provided that the plant is in the COLD SHUTDOWN condition and that no work is performed on the reactor or its connected systems which could result in lowering the reactor water level to less than 4'8" above the TOP OF THE ACTIVE FUEL.

l. Bypass in IRM Ranges 8,9, and 10.

m. There is one time delay relay associated with each of two pumps.

n. One time delay relay per pump must be OPERABLE.

o. There are two time delay relays associated with each of two pumps. One timer per pump is for sequence starting (SK1A, SK2A) and one timer per pump is for tripping the pump circuit breaker (SK7A, SK8A).

p. Two time delay relays per pump must be OPERABLE.

q. Manual initiation of affected component can be accomplished after the automatic load sequencing is completed.

r. Time delay starts after closing of containment spray pump circuit breaker.

s. These functions not required to be OPERABLE with the reactor temperature less than 212°F and the vessel head removed or vented or during REACTOR VESSEL PRESSURE TESTING.

t. These functions may be inoperable or bypassed when corresponding portions in the same core spray system logic train are inoperable per Specification 3.4.A.

u. These functions not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required to be maintained.

v. These functions not required to be OPERABLE when the ADS is not required to be OPERABLE.

w. These functions must be OPERABLE only when irradiated fuel is in the fuel pool or reactor vessel and SECONDARY CONTAINMENT INTEGRITY is required per Specification 3.5.B.

y. Deleted

z. The bypass function to permit scram reset in the SHUTDOWN or REFUEL MODE with control rod block must be OPERABLE in this mode.

aa. Pump circuit breakers will be tripped in 10 seconds ± 15% during a LOCA by relays SK7A and SK8A.

bb. Pump circuit breakers will trip instantaneously during a LOCA.

cc. Only applicable during STARTUP MODE while OPERATING in IRM range 10.

dd. If an isolation condenser inlet (steam side) isolation valve becomes or is made inoperable in the open position during the RUN MODE comply with Specification 3.8.E. If an AC motor-operated outlet (condensate return) isolation valve becomes or is made inoperable in the open position during the RUN MODE comply with Specification 3.8.F.

ee. With the number of OPERABLE channels one less than the Minimum Number of OPERABLE Instrument Channels per OPERABLE Trip System, operation may proceed until performance of the next required Channel Functional Test provided the inoperable channel is placed in the tripped condition within 1 hour.

ff. This function is not required to be OPERABLE when the associated safety bus is not required to be energized or fully OPERABLE as per applicable sections of these Technical Specifications.

gg. Deleted

hh. The high flow trip function for "B" Isolation Condenser is bypassed upon initiation of the alternate shutdown panel. This prevents a spurious trip of the Isolation Condenser in the event of fire induced circuit damage.

ii. Instrument shall be OPERABLE during main condenser air ejector operation except that a channel may be taken out-of-service for the purpose of a check, calibration, test, or maintenance without declaring it inoperable.

jj. With no channel OPERABLE, main condenser offgas may be released to the environment for as long as 72 hours provided the stack radioactive noble gas monitor is OPERABLE. Otherwise, be in at least SHUTDOWN CONDITION within 24 hours.

kk. One channel may be placed in an inoperable status for up to two hours for required surveillance without placing the trip system in the tripped condition.

ll. This function not required to be OPERABLE with the reactor vessel head removed or unbolted.

mm. "Instrument Channel" in this case refers to the bellows which sense vacuum in each of the three condensers (A, B, and C), and "Trip System" refers to vacuum trip systems 1 and 2.

nn. With one required channel inoperable in one Trip System, within 12 hours, restore the inoperable channel or place the inoperable channel and/or that Trip System in the tripped▲ condition.

With two or more required channels inoperable:

1. Within one hour, verify sufficient channels remain OPERABLE or tripped▲ to maintain trip capability, and

2. Within 6 hours, place the inoperable channel(s) in one Trip System and/or that Trip System▲▲ in the tripped condition▲, and

3. Within 12 hours, restore the inoperable channels in the other Trip System to an OPERABLE status or tripped▲.

Otherwise, take the Action Required.

▲ An inoperable channel or Trip System need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, if the inoperable channel is not restored to OPERABLE status within the required time, the Action Required shall be taken.

▲▲ This action applies to that Trip System with the most inoperable channels; if both Trip Systems have the same number of inoperable channels, the action can be applied to either Trip System.

oo. With one required channel inoperable in <u>one</u> Trip System, either

1. Place the inoperable channel in the tripped condition within

   a. 12 hours for parameters common to Scram Instrumentation, and
   b. 24 hours for parameters not common to Scram Instrumentation.

or

2. Take the Action Required.

With one required channel inoperable in both Trip Systems,

1. Place the inoperable channel in one Trip System in the tripped condition within one hour, and

2. a. Place the inoperable channel in the remaining Trip System in the tripped condition within

   (1) 12 hours for parameters common to Scram Instrumentation, and
   (2) 24 hours for parameters not common to Scram Instrumentation.

or

   b. Take the Action Required.

pp. With one or more required channels inoperable per Trip System:

1. For one channel inoperable, within 24 hours place the inoperable channel in the tripped condition or take the Action Required.

2. With more than one channel inoperable, take the Action Required.

SECTION 4

SUREVEILLANCE REQUIREMENTS

4.1 PROTECTIVE INSTRUMENTATION

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

Objective: To specify the minimum frequency and type of surveillance to be applied to the safety instrumentation.

Specification: Instrumentation shall be checked, tested, and calibrated as indicated in Tables 4.1.1 and 4.1.2 using the definitions given in Section 1.

4.1 PROTECTIVE INSTRUMENTATION

Bases:

Surveillance intervals are based on reliability analyses and have been determined in accordance with General Electric Licensing Topical Reports given in References 1 through 5.

The functions listed in Table 4.1.1 logically divide into three groups:

a. On-off sensors that provide a scram function or some other equally important function.

b. Analog devices coupled with a bi-stable trip that provides a scram function or some other vitally important function.

c. Devices which only serve a useful function during some restricted mode of operation, such as startup or shutdown, or for which the only practical test is one that can be performed only at shutdown.

Group (b) devices utilize an analog sensor followed by an amplifier and bi-stable trip circuit. The sensor and amplifier are active components and a failure would generally result in an upscale signal, a downscale signal, or no signal. These conditions are alarmed so a failure would not go undetected. The bi-stable portion does need to be tested in order to prove that it will assume its tripped state when required.

Group (c) devices are active only during a given portion of the operational cycle. For example, the IRM is inactive during full-power operation and active during startup. Thus, the only test that is significant is the one performed just prior to shutdown and startup. The condenser Low Vacuum trip can only be tested during shutdown, and although it is connected into the reactor protection system, it is not required to protect the reactor. Testing at each REFUELING OUTAGE is adequate. The switches for the high temperature main steamline tunnel are not accessible during normal operation because of their location above the main steam lines. Therefore, after initial calibration and in-place OPERABILITY checks, they will not be tested between refueling shutdowns. Considering the physical arrangement of the piping which would allow a steam leak at any of the four sensing locations to affect the other locations, it is considered that the function is not jeopardized by limiting calibration and testing to refueling outages.

The logic of the instrument safety systems in Table 4.1.1 is such that testing the instrument channels also trips the trip system, verifying that it is OPERABLE. However, certain systems require coincident instrument channel trips to completely test their trip systems. Therefore, Table 4.1.2 specifies the minimum trip system test frequency for these tripped systems. This assures that all trip systems for protective instrumentation are adequately tested, from sensors through the trip system.

IRM calibration is to be performed during reactor startup. The calibration of the IRMs during startup will be significant since the IRMs will be relied on for neutron monitoring and reactor protection up to 38.4% of rated power during a reactor startup.

To ensure that the APRMs are accurately indicating the true core average power, the APRMs are calibrated to the reactor power calculated from a heat balance. Limiting Safety System Settings (LSSS) 2.3.A.1 allows the APRMs to be reading greater than actual thermal power to compensate for localized power peaking. When this adjustment is made, the requirement for the absolute difference between the APRM channels and the calculated power to indicate within 2% RTP is modified to include any gain adjustments required by LSSS 2.3.A.1.

LPRM gain settings are determined from the local flux profiles measured by the Traversing Incore Probe (TIP) System. This establishes the relative local flux profile for appropriate representative input to the APRM System. The 1000 MWD/T Frequency is based on operating experience with LPRM sensitivity changes.

General Electric Licensing Topical Report NEDC-30851P-A (Reference 1), Section 5.7 indicates that the major contributor to reactor protection system unavailability is common cause failure of the automatic scram contactors. Analysis showed a weekly test interval to be optimum for scram contactors. The test of the automatic scram contactors can be performed as part of the Channel Calibration or Test of Scram Functions or by use of the subchannel test switches.

References:

(1) NEDC-30851P-A, "Technical Specification Improvement Analyses for BWR Reactor Protection System."

(2) NEDC-30936P-A, "BWR Owners' Group Technical Specification Improvement Methodology (With Demonstration for BWR ECCS Actuation Instrumentation)," Parts 1 and 2.

(3) NEDC-30851P-A, Supplement 1, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation."

(4) NEDC-30851P-A, Supplement 2, "Technical Specification Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation."

(5) NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation."

MINIMUM CHECK, CALIBRATION AND TEST FREQUENCY FOR PROTECTIVE INSTRUMENTATION

Instrument Channel	Check	Calibrate	Test	Remarks (Applies to Test & Calibration)
1. High Reactor Pressure	1/d	Note 3	1/3 mo.	
2. High Drywell Pressure (Scram)	N/A	1/3 mo.	1/3 mo.	By application of test pressure
3. Low Reactor Water Level	1/d	Note 3	1/3 mo.	
4. Low-Low Water Level	1/d	Note 3	1/3 mo.	
5. High Water Level in Scram Discharge Volume				
a. Digital	N/A	1/3 mo.	1/3 mo.	By varying level in sensor columns
b. Analog	N/A	Note 3	1/3 mo.	
6. Low-Low-Low Water Level	N/A	1/3 mo.	1/3 mo.	By application of test pressure
7. High Flow in Main Steamline	1/d	1/3 mo.	1/3 mo.	By application of test pressure
8. Low Pressure in Main Steamline	N/A	1/3 mo.	1/3 mo.	By application of test pressure
9. High Drywell Pressure (Core Cooling)	1/d	1/3 mo.	1/3 mo.	By application of test pressure
10. Main Steam Isolation Valve (Scram)	N/A	N/A	1/3 mo.	By exercising valve

MINIMUM CHECK, CALIBRATION AND TEST FREQUENCY FOR PROTECTIVE INSTRUMENTATION

Instrument Channel		Check	Calibrate	Test	Remarks (Applies to Test & Calibration)
11.	APRM Level	N/A	1/3d	N/A	Verify the absolute difference between the APRM channels and the calculated power is ≤ 2% rated thermal power [plus any gains required by LSSS 2.3.A.1].
	APRM Scram Trips • Flow biased neutron flux - high • Fixed neutron flux - high or inop • Downscale	Note 2	1/3 mo.	1/3 mo.	Using built-in calibration equipment during POWER OPERATION
12.	APRM Rod Blocks	Note 2	1/3 mo.	1/3 mo.	Upscale and downscale
13.	DELETED				
14.	High Radiation in Reactor Building				
	Operating Floor	1/s	1/3 mo.	1/3 mo.	Using gamma source for calibration
	Ventilation Exhaust	1/s	1/3 mo.	1/3 mo.	
15.	High Radiation on Air Ejector Off-Gas		1/3 mo.	1/3 mo.	Using built-in calibration equipment
		1/s			Channel Check
		1/mo.			Source check
			1/24 mo.		Calibration according to established station calibration procedures
				1/24 mo.	Note a

MINIMUM CHECK, CALIBRATION AND TEST FREQUENCY FOR PROTECTIVE INSTRUMENTATION

Instrument Channel		Check	Calibrate	Test	Remarks (Applies to Test & Calibration)
16.	IRM Level	N/A	Each startup	N/A	
	IRM Scram	*	*	*	Using built-in calibration equipment
17.	IRM Blocks	N/A	Prior to startup and shutdown	Prior to startup and shutdown	Upscale and downscale
18.	Condenser Low Vacuum	N/A	1/24 mo.	1/24 mo.	
19.	Manual Scram Buttons	N/A	N/A	1/3 mo.	
20.	High Temperature Main Steamline Tunnel	N/A	1/24 mo.	Each refueling outage	Using heat source box
21.	SRM	*	*	*	Using built-in calibration equipment
22.	Isolation Condenser High Flow ΔP (Steam & Water)	N/A	1/3 mo.	1/3 mo.	By application of test pressure
23.	Turbine Trip Scram	N/A	N/A	1/3 mo.	
24.	Generator Load Rejection Scram	N/A	1/3 mo.	1/3 mo.	
25.	Recirculation Loop Flow	N/A	1/24 mo.	N/A	By application of test pressure
26.	Low Reactor Pressure Core Spray Valve Permissive	N/A	1/3 mo.	1/3 mo.	By application of test pressure

MINIMUM CHECK, CALIBRATION AND TEST FREQUENCY FOR PROTECTIVE INSTRUMENTATION

Instrument Channel	Check	Calibrate	Test	Remarks (Applies to Test & Calibration)
27. Scram Discharge Volume (Rod Block)				
a) Water level high	N/A	Each refueling outage	1/3 mo.	Calibrate by varying level in sensor column
b) Scram Trip bypass	N/A	N/A	Each refueling outage	
28. Loss of Power				
a) 4.16 KV Emergency Bus Undervoltage (Loss of Voltage)	1/d	1/24 mo.	1/mo.	
b) 4.16 KV Emergency Bus Undervoltage (Degraded Voltage)	1/d	1/24 mo.	1/mo.	
29. Drywell High Radiation	N/A	Each refueling outage	Each refueling outage	
30. Automatic Scram Contactors	N/A	N/A	1/wk	Note 1
31. Core Spray Booster Pump Differential Pressure	N/A	1/3 mo.	1/3 mo.	By application of a test pressure

MINIMUM CHECK, CALIBRATION AND TEST FREQUENCY FOR PROTECTIVE INSTRUMENTATION

Instrument Channel			Check	Calibrate	Test	Remarks (Applies to Test & Calibration)
32.	LPRM Level					
	a)	Electronics	N/A	1/12 mo.	1/12 mo.	
	b)	Detectors	N/A	Note 4	N/A	

* Calibrate prior to startup and normal shutdown and thereafter check 1/s and test 1/wk until no longer required.

Legend:
N/A = Not Applicable
1/s = Once per shift
1/d = Once per day
1/3d = Once per 3 days;
1/wk = Once per week
1/mo. = Once per month
1/3 mo. = Once every 3 months;
1/12 mo. = Once every 12 months
1/24 mo. = Once every 24 months

MINIMUM CHECK, CALIBRATION AND TEST FREQUENCY FOR PROTECTIVE INSTRUMENTATION

NOTE 1: Each automatic scram contactor is required to be tested at least once per week. When not tested by other means, the weekly test can be performed by using the subchannel test switches.

NOTE 2: At least daily during reactor POWER OPERATION, the reactor neutron flux peaking factor shall be estimated and flow-referenced APRM scram and rod block settings shall be adjusted, if necessary, as specified in Section 2.3 Specifications A.1 and A.2.

NOTE 3: Calibrate electronic bistable trips by injection of an external test current once per 3 months. Calibrate transmitters by application of test pressure once per 12 months.

NOTE 4: Perform LPRM detectors calibration every 1000 MWD/MT Average Core Exposure

The following notes are only for Item 15 of Table 4.1.1:

A channel may be taken out of service for the purpose of a check, calibration, test or maintenance without declaring the channel to be inoperable.

a. The Channel Test shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:

1) Instrument indicates measured levels above the alarm setpoint.
2) Instrument indicates a downscale failure.
3) Instrument controls not set in operate mode.
4) Instrument electrical power loss.

TABLE 4.1.2

MINIMUM TEST FREQUENCIES FOR TRIP SYSTEMS

Trip System		Minimum Test Frequency
1)	Dual Channel (Scram)	Same as for respective instrumentation in Table 4.1.1
2)	Rod Block	Same as for respective instrumentation in Table 4.1.1
3)	DELETED	DELETED
4)	Automatic Depressurization each trip system, one at a time	Each refueling outage
5)	MSIV Closure each closure logic circuit independently (1 valve at a time)	Each refueling outage
6)	Core Spray each trip system, one at a time	1/3 mo and each refueling outage
7)	Primary Containment Isolation, each closure circuit independently (1 valve at a time)	Each refueling outage
8)	Refueling Interlocks	Prior to each refueling operation
9)	Isolation Condenser Actuation and Isolation each trip circuit independently (1 valve at a time)	Each refueling outage
10)	Reactor Building Isolation and SGTS Initiation	Same as for respective instrumentation in Table 4.1.1
11)	Condenser Vacuum Pump Isolation	Prior to each startup
12)	Air Ejector Offgas Line Isolation	Each refueling outage
13)	Containment Vent and Purge Isolation	1/24 mo.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 208

TO FACILITY OPERATING LICENSE NO. DPR-16

GPU NUCLEAR, INC., AND

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

1.0 INTRODUCTION

By letter dated November 5, 1998, as supplemented February 18, 1999, GPU Nuclear, Inc. (the licensee) submitted a request for changes to the Oyster Creek Nuclear Generating Station Technical Specifications (TSs). The requested changes would modify the safety limits and surveillances of the local-power range monitor (LPRM) and average-power range monitor (APRM) systems and related Bases pages to ensure the APRM channels respond within the necessary range and accuracy and to verify channel operability. In addition, an unrelated change to the Bases of Specification 2.3 is included to clarify some ambiguous language. The February 18, 1999, letter provided clarifying information within the scope of the original application and did not change the staff's initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

The neutron monitoring system provides the capability to monitor neutron flux in the reactor core. For power range monitoring, selected groups of LPRMs provide input signals to the APRMs for bulk power level monitoring and automatic core protection. Four APRM channels are connected in each of the two reactor protection system (RPS) channels.

On December 28, 1997, the NRC completed an integrated inspection at the Oyster Creek Nuclear Generating Station. The findings of this inspection were forwarded to the licensee by letter dated February 8, 1998, (Reference 2), "NRC Integrated Inspection Report No. 50-219/97-10 and Notice of Violation." During the inspection, the bypass of an APRM channel to perform LPRM calibrations on February 21, 1995 was reviewed. In accordance with TS 3.1.B.3, "Protective Instrumentation," an APRM channel cannot be bypassed in a quadrant diagonally opposite a quadrant containing more than one bypassed or failed LPRM detector on the same axial level unless one of the following conditions exist:

1. The APRM channel bypass is in support of a TS required LPRM/APRM surveillance.

2. Power is reduced below the 80% rod line.

3. The corresponding reactor protection system (RPS) trip system is placed in the tripped condition.

During the February 21, 1995, LPRM calibration, APRM channel 5 was bypassed in support of an LPRM calibration at the same time that two LPRMs at the same axial level and in the diagonally opposite quadrant were inoperable. It was the licensee's interpretation at that time, that it considered the LPRM calibration to be a TS surveillance required to ensure that the APRMs were operable. According to this interpretation, the licensee concluded that TS 3.1.B.3 was met and that there was no need to decrease power below the 80% rod line or place the RPS trip system in the tripped condition. The NRC concluded during its inspection, however, that since the calibration is not listed in the Oyster Creek TS, it is not a TS-required surveillance, and that TS 3.1.B.3 had been violated. It is the licensee's position that this type of calibration is appropriate and ensures the accuracy and operability of the APRM system. The licensee conducted a review of the Oyster Creek TS to identify any necessary changes to the TS in order to eliminate ambiguity in this area. As a result, the licensee has proposed to modify their TSs as discussed below.

3.0 EVALUATION

The Safety Limits contained in TSs 2.1 and 2.3 have been established to protect the integrity of the fuel cladding and reactor coolant system barriers. Automatic protective devices have been provided in the plant design for corrective actions to prevent safety limits from being exceeded during normal operation or operational transients. TS 2.3, "Limiting Safety Systems Settings," establishes the trip settings for these automatic protective devices. The licensee has proposed to change TS 2.3 and the associated Bases section to add limiting safety systems setting 2.3.A.3, "APRM downscale $\geq$ 2% Rated Thermal Power coincident with IRM Upscale (high-high) or Inoperative." The APRM downscale signal ensures that there is adequate neutron monitoring system protection if the reactor mode switch is placed in the run position prior to APRMs coming on scale. With the reactor mode switch in run, an APRM downscale signal, coincident with an associated IRM upscale (high-high) or inoperative, generates a reactor trip signal.

The licensee has proposed to modify the surveillance requirements of TS table 4.1.1, "Minimum Check, Calibration and Test Frequency for Protective Instrumentation." The current surveillance requirements specify the required check, calibration, and test of APRM scram trips. The proposed change would specify the three APRM scram trips for which these requirements apply: flow biased neutron flux - high; fixed neutron flux - high or inop; and downscale. TS table 4.1.1 also specifies that the APRMs be calibrated to the reactor power calculated from a heat balance equation. This ensures that the APRMs are accurately indicating the true core average power. The licensee has proposed to modify the requirement for the absolute difference between the APRM channel reading and the calculated power to indicate within 2 percent reactor power, to include any gain adjustments required by limiting safety systems setting 2.3.A, "Neutron flux scram."

The addition of the APRM downscale limiting safety system setting and additional surveillance requirements for APRM scram trips is more restrictive than the current TS and provides additional assurance of the capability of the neutron monitoring system to perform its intended safety functions. Therefore, the proposed changes are acceptable.

The licensee has also proposed to modify TS Table 3.1.1, "Protective Instrumentation Requirements," to include the function "APRM Downscale/IRM Upscale," and to modify TS table 4.1.1, "Minimum Check, Calibration and Test Frequency for Protective Instrumentation," to add a calibration for LPRM level. The proposed modification will add instrumentation requirements for LPRMs similar to the requirements specified for APRMs.

The current TS 3.1.B.3 permits bypassing an APRM channel for surveillance tests when one or more LPRM detectors in the opposite quadrant on the same axial level are bypassed. On February 21, 1995, an APRM channel was bypassed to perform a LPRM calibration while two LPRMs at the same axial level in a diagonally opposite quadrant were inoperable. In reviewing this event, an NRC inspection team concluded that because the LPRM calibration was not listed in the Oyster Creek TS, that event violated the requirements of TS Section 3.1.B.3. To resolve this limitation the licensee proposed the addition of LPRM instrumentation to TS Table 4.1.1.

The licensee also reformatted Tables 3.1.1 and 4.1.1 to eliminate style and font inconsistencies and change table orientation to landscape format.

TS Table 3.1.1, Function 13, APRM Downscale/IRM Upscale

This change is in conformance with APRM and other comparable channel functions specified in TS Table 3.1.1 and is, therefore, acceptable.

TS Table 4.1.1, Instrument Channel 32, LPRM Level

The licensee proposed to add a requirement to TS Table 4.1.1 to calibrate and test LPRM level electronics and detectors once every 12 months. The current TS Table 4.1.1, Item 11, specifies testing and calibration of APRM Scram Trip channels once every 3 months and the licensee stated that during this APRM test the LPRM power supplies are checked for drifts. If the power supply is found to be out of tolerance, a step in the APRM test procedure directs the technician to perform a front panel check of the associated LPRM modules.

The licensee also stated that the LPRM front panels, including power supplies, meters, alarms, and rod block functions, are tested on a 48-week schedule in conformance with current plant procedures. This frequency was selected based on historical data. The licensee further stated that quarterly testing would require connection and disconnection of a large number of LPRMs to the test equipment that would increase the likelihood of equipment failures.

The proposed changes are improvements to the current TS and are in conformance with the current TS requirements for APRM or other comparable instrumentation, and are based on plant experience and current plant procedures, and are, therefore, acceptable.

The licensee made some changes to the Bases section 2.3. The Bases are not subject to NRC review and approval; therefore, these changes were not reviewed but the changes are included to maintain the NRC Authority File current.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (63 FR 69342). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

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7.0 REFERENCES

1. Letter from Michael B. Roche (GPU Nuclear, Inc.) to NRC, "Technical Specification Change Request No. 266, APRM/LPRM Safety Limits and Surveillance Requirements," dated November 5, 1998.

2. Letter from Hubert J. Miller, NRC, to Michael B. Roche, GPU Nuclear, Inc., "NRC Integrated Inspection Report No. 50/219/97-10 and Notice of Violation," dated February 8, 1998.