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

March 31, 1987

Docket No. 50-219

Mr. P. B. Fiedler
Vice President and Director
Oyster Creek Nuclear Generating Station
Post Office Box 388
Forked River, New Jersey 08731

Dear Mr. Fiedler:

SUBJECT: CONTAINMENT HIGH RANGE RADIATION MONITORS (TSCR 152, TAC 64381 AND 64015)

Re: Oyster Creek Nuclear Generating Station

The Commission has issued the enclosed Amendment No. 116 to Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station. This amendment is in response to your application dated November 26, 1986.

The amendment adds requirements to Section 3.1 and 4.1, Protective Instrumentation; Section 3.5, Containment; and Sections 3.13 and 4.13, Accident Monitoring Instrumentation, of the Appendix A Technical Specifications (TS). These new requirements concern limiting conditions for operation and surveillance on the containment high range radiation monitors and the isolation capability upon high radiation of the large containment vent and purge isolation valves. The amendment authorizes the following: (1) adding the containment high radiation trip system to isolate the large containment vent and purge valves to Table 3.1.1 and adding text on this to the Bases for Table 3.1.1; (2) identifying these large valves in Table 3.5.2; (3) adding the limiting conditions for plant operation for the containment high range radiation monitors in Section 3.13 and Table 3.13.1; (4) adding the surveillance on the containment high range radiation monitor trip instrumentation for its containment isolation function to Tables 4.1.1 and 4.1.2; and (5) adding surveillance requirements for the containment high range monitors, to monitor high radiation, to Section 4.13 and Table 4.13.1.

As discussed in the enclosed Safety Evaluation (SE) on this amendment action, you are requested to propose additional TS to initiate the preplanned alternate method capable of monitoring the containment radiation if both containment high range radiation monitors are inoperable for more than 7 days. The date to submit this TS should be negotiated with the NRC Project Manager. The Notice of Issuance will be included in the Commission's biweekly <u>Federal Register</u> notices.

Sincerely,

Original signed by

Marshall Grotenhuis, Acting Director
BWR Project Directorate #1
Division of BWR Licensing

Enclosures:
1. Amendment No. 116 to
   License No. DPR-16
2. Safety Evaluation

cc w/enclosures:
See next page

DISTRIBUTION

Docket File	JZwolinski	BGrimes	EButcher	NThompson
NRC PDR	OGC-BETH	ACRS (10)	ELJordan	GLainas
Local PDR	LJHarmon	JDonohew	PDiggs	OC File
BWD1 Reading	TBarnhart (4)	DVassallo	WJones	
CJamerson (2)	OPA	JPartlow	RBernero	

DBL:BWD1	DBL:BWD1	OGC-BETH	DBL:BWD1
CJamerson	JDonohew:	MYoung	MGrotenhuis
03/13/87	03/13/87	03/20/87	03/31/87

Mr. P. B. Fiedler
Oyster Creek Nuclear Generating Station

Oyster Creek Nuclear
Generating Station

cc:
Mr. Ernest L. Blake, Jr.
Shaw, Pittman, Potts and Trowbridge
2300 N Street, NW
Washington, D.C. 20037

J.B. Liberman, Esquire
Bishop, Liberman, Cook, et al.
1155 Avenue of the Americas
New York, New York 10036

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, Pennsylvania 19406

BWR Licensing Manager
GPU Nuclear Corporation
1 Upper Pond Road
Parsippany, New Jersey 07054

Deputy Attorney General
State of New Jersey
Department of Law and Public Safety
36 West State Street - CN 112
Trenton, New Jersey 08625

Mayor
Lacey Township
818 West Lacey Road
Forked River, New Jersey 08731

Licensing Manager
Oyster Creek Nuclear Generating Station
Mail Stop: Site Emergency Bldg.
P. O. Box 388
Forked River, New Jersey 08731

Resident Inspector
c/o U.S. NRC
Post Office Box 445
Forked River, New Jersey 08731

Commissioner
New Jersey Department of Energy
101 Commerce Street
Newark, New Jersey 07102

Mr. David M. Scott, Acting Chief
Bureau of Nuclear Engineering
Department of Environmental Protection
CN 411
Trenton, New Jersey 08625

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

GPU NUCLEAR CORPORATION

AND

JERSEY CENTRAL POWER & LIGHT COMPANY

DOCKET NO. 50-219

OYSTER CREEK NUCLEAR GENERATING STATION

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 116
License No. DPR-16

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

   A. The application for amendment by GPU Nuclear Corporation and Jersey Central Power and Light Company (the licensees) dated November 26, 1986, 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 Provisional 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 No. 116 , are hereby incorporated in the license. GPU Nuclear Corporation shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Marshall Grotenhuis

Marshall Grotenhuis, Acting Director
BWR Project Directorate #1
Division of BWR Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: March 31, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 116

PROVISIONAL OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain vertical lines indicating the area of change.

REMOVE	INSERT
3.1-7	3.1-7
3.1-16	3.1-16
3.1-17	3.1-17
3.1-18	3.1-18
--	3.1-19
3.5-13	3.5-13
3.13-3	3.13-3
3.13-4	3.13-4
--	3.13-5
4.1-8	4.1-8
4.1-9	4.1-9
4.13-1	4.13-1
4.13-2	4.13-2

Under assumed loss-of-coolant accident conditions and under certain loss of offsite power conditions with no assumed loss-of-coolant accident, it is inadvisable to allow the simultaneous starting of emergency core cooling and heavy load auxiliary systems in order to minimize the voltage drop across the emergency buses and to protect against a potential diesel generator overload. The diesel generator load sequence time delay relays provide this protective function and are set accordingly. The repetitive accuracy rating of the timer mechanism as well as parametric analyses to evaluate the maximum acceptable tolerances for the diesel loading sequence timers were considered in the establishment of the appropriate load sequencing.

Manual actuation can be accomplished by the operator and is considered appropriate only when the automatic load sequencing has been completed. This will prevent simultaneous starting of heavy load auxiliary systems and protect against the potential for diesel generator overload.

Also, the Closed Cooling Water and Service Water pump circuit breakers will trip whenever a loss-of-coolant accident condition exists. This is justified by Amendment 42 of the Licensing Application which determined that these pumps were not required during this accident condition.

The drywell high radiation setpoint will ensure a timely closure of the large vent and purge isolation valves to prevent releases from exceeding ten percent of the dose guideline values allowed by 10 CFR 100. The containment vent and purge isolation function is provided in response to NUREG 0737 Item II E.4.2.7.

Reference:

(1) NEDO-10189 "An Analysis of Functional Common Mode Failures in GE BWR Protection and Control Instrumentation," L. G. Frederick, et al., July 1970.

TABLE 3.1.1 PROTECTIVE INSTRUMENTATION REQUIREMENTS (CONTD)

Function	Trip Setting	Reactor Modes in which Function Must Be Operable: Shutdown	Refuel	Startup	Run	Min. No. of Operable or Operating [tripped] Trip Systems	Min. No. of Instrument Channels Per Operable Trip Systems	Action Required*
O. Containment Vent and Purge Isolation								
1. Drywell High Radiation	$\leq$74.6 R/hr	X(u)	X(u)	X(u)	X	1	1	Isolate vent and purge path-ways or place in cold shut-down condition

TABLE 3.1.1 (CONTD)

* Action required when minimum conditions for operation are not satisfied. Also permissible to trip inoperable trip system. When necessary to conduct tests and calibrations, one channel may be made inoperable for up to two hours per Technical Specification required surveillance without tripping its trip system.

** 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 800 psia 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 one hour 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 IRM shall be inserted and operable until the APRM's are operable and reading at least 2/150 full scale.
e. Offgas system isolation trip set at $\leq 2.1/\bar{E}$ Ci/sec where $\bar{E}$ = average gamma energy from noble gas in offgas after holdup line (Mev). 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. One instrument channel in each trip system may be inoperable provided the circuit which it operates in the trip system is placed in a simulated tripped condition. If repairs cannot be completed within 72 hours the reactor shall be placed in the cold shudown condition. If more than one instrument channel in any trip system becomes inoperable, the reactor shall be placed in the cold shutdown condition. Relief valve controllers shall not be bypassed for 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. The interlock is not required during the start-up test program and demonstration of plant electrical output but shall be provided following these actions.
j. Not required below 40% of turbine rated steam flow.

TABLE 3.1.1 (CONTD)

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.
t. These functions may be operable 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. The number of operable channels may be reduced to 2 per Specification 3.9-E and F.
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 Min. No. of Operable Instrument Channels per Operable Trip Systems, 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.

TABLE 3.1.1 (CONTD)

*gg. These functions are not required to be operable when secondary containment is not required to be maintained or when the conditions of Sections 3.5.b.1.a, b, c, and d are met, and reactor water level is closely monitored and logged hourly. The Standby Gas Treatment System will be manually initiated if reactor water level drops to the low level trip set point.

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 to be 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

* This note is applicable only during the Cycle 10M outage.

TABLE 3.5.2

CONTAINMENT ISOLATION VALVES

VALVE FUNCTION/VALVE DESIGNATON	ISOLATION SIGNALS
Main Steam Isolation Valves (NS03A, NS03B, NS04A, NS04B)	1
Main Steam Condensate Drain Valves (V-1-106, V-1-107, V-1-110, V-1-111)	1
Reactor Building Closed Cooling Valves (V-5-147, V-5-166, V-5-167)	2
Instrument Air Valve (V-6-395)	1
Emergency Condenser Vent Valves (V-14-1, V-14-5, V-14-19, V-14-20)	1
Reactor Cleanup Valves (V-16-1, V-16-2, V-16-14, V-16-61)	3
Shutdown Cooling Valves (V-17-19, V-17-54)	3
Drywell Equipment Drain Tank Valves (V-22-1, V-22-2)	3
Drywell Sump Valves (V-22-28, V-22-29)	3
Drywell and Torus Atmosphere Control Valves (V-27-1**, V-27-2**, V-27-3**, V-27-4**, V-28-17**, V-28-18**, V-23-21, V-23-22, V-28-47, V-23-13, V-23-14, V-23-15, V-23-16, V-23-17, V-23-18, V-23-19, V-23-20)	3
Reactor Recirculation Loop Sample Valves (V-24-29, V-24-30)	1
Torus to Reactor Building Vacuum Relief Valves (V-26-16, V-26-18)	3*
Traversing In-Core Probe System (Tip machine ball valve No. 1, No. 2, No. 3, No. 4)	3

1)Reactor Isolation Signals as shown in Table 3.1.1
2)Low-Low Reactor Water Level and High Drywell Pressure; or Low-Low-Low Reactor Water Level.
3)Primary Containment Isolation Signals as shown in Table 3.1.1

*Valves automatically reset to provide vacuum relief
**Valves Isolate on Drywell High Radiation Signals.

2. With the number of operable channels less than the total number of channels shown in Table 3.13.1, restore the inoperable channel to operable status within 30 days or place the reactor in the shutdown condition within the next 24 hours.

3. With the number of operable channels less than the minimum channels operable requirements of Table 3.13.1, restore at least one channel to operable status within 7 days or place the reactor in the shutdown condition within the next 24 hours.

G. Containment High-Range Radiation Monitor

1. Two in-containment high range radiation monitors shall be operable at all times except for cold shutdown and other times when primary containment is not required.

2. In case of failure of one or more monitors, appropriate actions shall be taken to restore its operable capability as soon as possible. Also, refer to Table 3.1.1 for any additional action which may be required.

3. If the monitor or monitors are not restored to operable condition within 7 days after the failure, a special report shall be submitted to the NRC within 14 days following the event, outlining the cause of inoperability, actions taken and the planned schedule for restoring the equipment to operable status.

BASES

The purpose of the safety/relief valve accident monitoring instrumentation is to alert the operator to a stuck open safety/relief valve which could result in an inventory threatening event.

As the safety valves present distinctly different concerns than those related to relief valves, the technical specifications are separated as to the actions taken upon inoperability. Clearly, the actuation of a safety valve will be immediately detectable by observed increase in drywell pressure. Further confirmation can be gained by observing reactor pressure and water level. Operator action in response to these symptoms would be taken regardless of the acoustic monitoring system status. Acoustic monitors act only to confirm the reseating of the safety valve. In actuality, the operator actions in response to the lifting of a safety valve will not change whether or not the safety valve reseats. Therefore, the actions taken for inoperable acoustic monitors on safety valves are significantly less stringent than that taken for those monitors associated with relief valves.

Should an acoustic monitor on a safety valve become inoperable, setpoints on adjacent monitors will be reduced to assure alarm actuation should the safety valve lift, since it is of no importance to the operator as to which valves lift but only that one has lifted. Analyses, using very conservative blowdown forces and attenuation factors, show that reducing the alarm setpoint on adjacent monitors to <1.4g will assure alarm actuation should the adjacent safety valve lift. Minimum blowdown force considered was 30g with a maximum attenuation of 27dB. In actuality, a safety valve lift would result in considerably larger blowdown force. The maximum attenuation of 27dB was determined based on actual testing of a similar monitoring system installed in a similar configuration.

The operability of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. This capability is consistent with NUREGs 0578 and 0737.

TABLE 3.13.1

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Relief Valve Position Indicator (Primary Detector*)	1/valve	1/valve
Relief Valve Position Indicator (Backup Indications**)	1/valve	
2. Wide Range Drywell Pressure Monitor (PT/PR-53 & 54)	2	1
3. Wide Range Torus Water Level Monitor (LT/LR-37 7 38)	2	1
4. Drywell H2 Monitor	2	1
5. Containment High Range Radiation Monitor	2	1

* Acoustic Monitor
** Thermocouple
Thermocouple TE 65A can be substituted for thermocouple TE210-43V, W, or X
Thermocouple TE 65B can be substituted for thermocouple TE210-43Y, or Z

Table 4.1.1 (cont'd)

	Instrument Channel	Check	Calibrate	Test	Remarks (Applies to Test and Calibration)
27.	Scram Discharge Volume (Rod Block)				
	a) Water level high	N A	Each refueling outage	Every 3 months	By varying level in switch column
	b) Scram trip bypass	NA	NA	Each refueling outage	
28.	Loss of Power				
	a) 4.16 KV Emergency Bus Undervoltage (Loss of voltage)	Daily	1/18 mos.	1/mo	
	b) 4.16 KV Emergency Bus Undervoltage (Degraded Voltage)	Daily	1/18 mos.	1/mo	
29.	Drywell High Radiation	NA	Each Re-fueling outage	Each Refuel-ing outage	

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

Legend:
NA = Not applicable; 1/s = Once per shift; 1/d = Once per day; 1/3d = Once per three days; 1/wk = Once per week; 1/3 mo = Once every 3 months; 1/18 mos. = Once every 18 months.

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 functional 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) Containment Spray, each trip system, one at a time	1/3 mo. and each refueling outage
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	Each refueling outage

4.13 ACCIDENT MONITORING INSTRUMENTATION

Applicability: Applies to surveillance requirements for the accident monitoring instrumentation

Objective: To verify the operability of the accident monitoring instrumentation.

Specification:

A. Safety & Relief Valve Position Indicators

Each accident monitoring instrumentation channel shall be demonstrated operable by performance of the Channel Check and Channel Calibration operations at the frequencies shown in Table 4.13.1.

B. Wide Range Drywell Pressure Monitor

Each accident monitoring instrumentation channel shall be demonstrated operable by performance of the Channel Check and Channel Calibration operations at the frequencies shown in Table 4.13.1.

C. Wide Range Torus Water Level Monitor

Each accident monitoring instrumentation channel shall be demonstrated operable by performance of the Channel Check and Channel Calibration operations at the frequencies shown in Table 4.13.1.

D. Drywell H_2 Monitor

Each accident monitoring instrumentation channel shall be demonstrated operable by performance of the Channel Check and Channel Calibration operations at the frequencies shown in Table 4.13.1.

E. Containment High-Range Radiation Monitor

Each accident monitoring instrumentation channel shall be demonstrated operable by performance of the Channel Check and Channel Calibration operations at the frequencies shown in Table 4.13-1.

Bases:

The operability of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. This capability is consistent with NUREGs 0578 and 0737.

TABLE 4.13-1

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHECK	CALIBRATION
1. Primary and Safety Valve Positon Indicator (Primary Detector*)	A	B
Relief and Safety Valve Position Indicator (Backup Indications**)	A	B
Relief Valve Position Indicator (Common Header Temperature Element**)	C	B***
2. Wide Range Drywell Pressure Monitor (PT/PR 53 & 54)	A	D
3. Wide Range Torus Water Level Monitor (LT/LR 37 & 38)	A	D
4. Drywell H_2 Monitor	A^1	E
5. Containment High Range Radiation Monitor	A	F****

Legend:

A = at least once per 31 days; B = at least once per 18 months (550 days).
C = at least once per 15 days until channel calibration is performed and thence at least once per 31 days.
D = at least once per 6 months; E = at least once per 12 months; 1 = Span and Zero using calibration gases.
F = each refueling outage; 1 = Span and Zero using calibration gases.

* Acoustic Monitor

** Thermocouple

*** This surveillance will commence at the first cold shutdown after July 1, 1985.

**** Channel calibration shall consist of electronic signal substitution of the channel, not including the detector, for all decades above 10R/hr and a one point calibration check of the detector at or below 10R/hr by means of a calibrated portable radiation source traceable to NBS.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 116 TO PROVISIONAL OPERATING LICENSE NO. DPR-16

GPU NUCLEAR CORPORATION AND

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

1.0 INTRODUCTION

By letter dated November 26, 1986, GPU Nuclear (the licensee) requested an amendment to Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station (Oyster Creek). This amendment would authorize added requirements to Section 3.1 and 4.1, Protective Instrumentation; Section 3.5, Containment; and Sections 3.13 and 4.13, Accident Monitoring Instrumentation, of the Appendix A Technical Specifications (TS). These new requirements concern limiting conditions for operation (LCO) and surveillance on the containment, or drywell, high range radiation monitors and the isolation capability upon high radiation of the large containment vent and purge isolation valves.

The licensee has proposed to (1) add the containment high radiation trip system to isolate the large containment vent and purge valves to Table 3.1.1 and text to the Bases for Table 3.1.1; (2) identify these large valves in Table 3.5.2; (3) add the limiting conditions for plant operation for the containment high range radiation monitors in Section 3.13 and Table 3.13.1; (4) add the surveillance on the containment high range radiation monitor trip instrumentation for its containment isolation function to Tables 4.1.1 and 4.1.2; and (5) add surveillance requirements for the containment high range monitors, to monitor high radiation, to Section 4.13 and Table 4.13.1.

2.0 DISCUSSION AND EVALUATION

The licensee has proposed Technical Specification Change Request (TSCR) No. 152 to add requirements concerning LCO and surveillance for the containment high range radiation monitors. TMI Action Plan, NUREG-0737, Item II.F.1.3, required the licensee to have two containment high range monitors to measure radiation inside containment during and following an accident. The licensee installed the two monitors in the containment or drywell during the current Cycle 11 Refueling (Cycle 11R) outage. The monitors are used to measure the radiation inside containment or drywell and provide a trip signal to close the large containment isolation valves. This isolation capability was also installed in the Cycle 11R outage to meet TMI Item II.E.4.2.7.

In TSCR No. 152, the licensee is proposing new LCO and surveillance requirements to cover (1) the newly installed monitor and (2) its containment isolation function. These two issues will be discussed separately below:

2.1 Containment High Range Radiation Monitor

The containment high range radiation monitors were discussed in the staff's evaluation of the licensee's response to Generic Letter (GL) 83-36, NUREG-0737 Technical Specifications, dated November 1, 1983. The staff's evaluation was issued on November 22, 1985. The staff stated in its evaluation that appropriate TS for these monitors were the model TS in Enclosure 3 to GL 83-36.

The licensee has proposed the following additional TS for these monitors: (1) the two monitors shall be operable at all times except for cold shutdown and when the primary containment is not required, (2) if a monitor is inoperable take action to restore the monitor to its operable capability, (3) if the monitor or monitors is not returned to the operable condition within 7 days, a special report is submitted to NRC within 14 days following the event, (4) each monitor channel shall be demonstrated operable by a channel check at least once per 31 days and by a channel calibration each refueling outage and (5) each channel calibration shall consist of an electronic signal substitution of the channel, not including the detector, for all decades above 10R/hr and a one point calibration check of the detector at or below 10R/hr by means of a calibrated portable radiation source traceable to NBS. These proposed additional TS are for the appropriate Sections 3.13 and 4.13, Accident Monitoring Instrumentation, and Tables 3.13-1 and 4.13-1 of the TS.

The licensee's proposed TS for these accident monitors are consistent with the model TS in GL 83-36 except for one difference. The model TS require the licensee to initiate a preplanned alternate method of monitoring containment radiation within 72 hours if one monitor is inoperable. This requirement has been revised in the BWR Standard Technical Specifications (STS), NUREG-0123, Revision 4, to have the alternate method of monitoring initiated after 72 hours when both monitors are inoperable. The second monitor is redundant to the first monitor. The STS on accident monitoring instrumentation applies to Oyster Creek.

By letter fowarding this safety evaluation, the licensee will be requested to propose the additional TS from NUREG-0123 Revision 4 on the preplanned alternate method of monitoring or provide its justification for not needing this TS for Oyster Creek.

Based on the above, the staff concludes that the licensee's proposed TS are acceptable.

2.2 Radiation Signal to Containment Purge/Vent Isolation Valves

The radiation signal for the containment purge and vent isolation valves was discussed in the staff's evaluation of the licensee's response to GL 83-02, NUREG-0737 Technical Specifications, dated January 10, 1983. This is TMI Action Plan Item II.E.4.2.7. The staff's evaluation was issued on May 30, 1985. The staff stated in its evaluation that the licensee should propose appropriate TS limiting conditions for operation and surveillance requirements for this item before plant restart form the Cycle 11R outage. The licensee did this in its application dated November 26, 1986.

The licensee has proposed the following additional TS for this item: (1) the radiation trip setting on high drywell radiation for the isolation signal is less than or equal to 74.6 R/hr, (2) the trip instrumentation is operable in all reactor modes except when primary containment is not required, (3) the minimum number of operable trip systems and of operable instrument channels per operable trip system is 1, (4) the action required if the minimum number is not met is to isolate the containment vent and purge lines or place the plant in cold shutdown, (5) indicate the specific containment isolation valves in TS Table 3.5.2 which are isolated on drywell high radiation, (6) the surveillance is a channel calibration and a channel test each refueling outage, and (7) the minimum test frequency is each refueling outage. These proposed TS are for Sections 3.1 and 4.1, Protective Instrumentation, and Tables 3.1.1, 4.1.1 and 4.1.2 of the TS.

The GL 83-02 did not provide acceptable TS for the radiation signal on purge valves. The licensee's basis for the proposed drywell high radiation setpoint is that it will ensure a timely closure of the large purge and vent valves to prevent releases from exceeding 10% of the exposure guidelines in 10 CFR Part 100, Reactor Site Criteria. This setpoint is acceptable because the staff has concluded that the monitor setpoint should, as a minimum, prevent the radiological consequences of accidental releases through the purge and vent lines from exceeding a small fraction (10%) of 10 CFR Part 100. This is in Enclosure 2 to the letter attached to the staff's Safety Evaluation dated August 4, 1986.

The licensee has proposed text to the Bases for Section 3.1 to explain the basis for the trip setpoint. This text is correct, therefore, adding the basis for the setpoint to the Bases is acceptable.

The licensee has proposed for TS Table 3.1.1 that the trip system and instrument channels per operable trip system to isolate the containment vent and purge isolation valves shall be operable in the shutdown, refuel, startup and run modes except (footnote u) if containment integrity is not required to be maintained. Therefore, the trip system and its associated instrumentation channel must be operable whenever containment integrity is required. The footnote u was not proposed for the run mode because this mode requires containment integrity. This is consistent with the purpose of the TMI Action Plan Item II.E.4.2 on containment isolation dependability and Item II.E.4.2.7 on a high radiation signal to close the large containment purge and vent isolation valves. The proposed applicable reactor modes are also the same as the existing applicable reactor modes for operable trip systems for containment isolation in Item F of Table 3.1.1. Therefore, based on this, the proposed TS on the applicable reactor modes for an operable trip system and associated instrument channel for containment purge and vent isolation on drywell high radiation is acceptable.

The licensee has proposed for Table 3.1.1 that the minimum number of operable or operating trip systems is 1 and the minimum number of operable instrument channels per operable trip system is 1. This is based on the design of the trip system. In its telephone discussion on the design of the trip systems on January 28, 1986, the licensee explained that the isolation signal to containment purge and bent isolation valves is provided by the two containment high range radiation monitors. The staff accepted these monitors to provide this signal in its evaluation dated August 31, 1983, on TMI Item II.E.4.2.7. There are two instrument channels and two trip systems. Each trip system has one instrument channel. A signal from either trip system will cause the containment isolation of the purge and vent valves. Therefore, the minimum number of operable trip systems and minimum number of operable instrument channels per operable trip system should be 1 and the proposed TS are acceptable.

In NUREG-0737, Clarification of TMI Action Plan Requirements, dated November 1980, the review of the trip system design for II.E.4.2.7 was a post implementation review. The licensee provided design data on this system in the meeting at the site on February 2, 1987. This data was the following: Class 1E and important to safety, seismic, fire protection, environmental qualification per 10 CFR 50.49, electrical separation and single failure criteria. The staff concludes that this design is acceptable. It does not conflict with NUREG-0737.

The licensee proposed to isolate the purge and vent pathways or place the plant in cold shutdown if there is an insufficient number of operable trip systems or of operable instrument channels per operable trip systems. This action does not state the number of hours or days that the licensee can take before the action required must be completed. This is the same for all the existing Actions Required in Table 3.1.1. The licensee would follow TS 3.0.A which requires the plant to be in cold shutdown within 30 hours if a limiting condition for operation cannot be satisfied. The proposed action required would (1) place the purge and vent pathways in the same condition as if the trip system had acted on high radiation or (2) place the reactor in cold shutdown where containment is not required and the trip system is not required to be operable. Therefore, the staff concludes that the proposed action required is acceptable.

The licensee proposed a change to Table 3.5.2, Containment Isolation Valves, to identify the containment isolation valves that will be closed by the trip system on containment high radiation. These are valves V-27-1, V-27-2, V-27-3, V-27-4, V-28-17, and V-28-18. These valves are the large containment purge and vent isolation valves identified in a figure in the staff's evaluation on containment vent and purge valve isolation on high radiation, dated August 4, 1986. Therefore, the staff concludes that this proposed change is acceptable.

The licensee did not identify any small containment purge and vent isolation valves which will close upon receiving the high radiation trip signal. In its letter dated December 2, 1986, the licensee submitted its justification for not providing this signal to close these small valves. The licensee stated that it has confirmed it meets the staff's criteria for not closing these small valves on a containment high radiation signal. These criteria are accident doses to the public which are a small fraction (10%) of the guidelines in 10 CFR Part 100. These criteria are given in Attachment I to the staff's evaluation dated August 4, 1986. Therefore, the fact that the licensee did not identify any of the small containment purge and vent isolation valves (i.e., V-23-1, V-23-2, V-28-47) in TS Table 3.5.2 as receiving a containment high radiation closure signal is acceptable. The question of the acceptability of not isolating these small valves on containment high radiation will be the subject of a separate letter addressing the licensee's December 2, 1986 letter.

In its application, the licensee proposed for Table 4.1.1 that the surveillance and frequency of surveillance for the containment or drywell high radiation instrument channel be the following: (1) the channel check is not applicable, and (2) the channel calibration and the channel test is every refueling outage. The licensee also proposed for Table 4.1.2 to have the minimum test frequency for the containment vent and purge isolation (i.e., large valves) trip system be each refueling outage. The channel check being not applicable is based on the two monitors reading out in the control room do not indicate low enough for these readouts to be used for a channel check. Nothing else exists to serve as a channel check.

The frequency for channel calibration and channel test is based on when the large containment vent and purge isolation valves are operated during an operating cycle. As explained in Section 3.0 of the staff's evaluation dated August 4, 1986, these large valves are open only for the following times: (1) assist in achieving an inerted containment atmosphere within 24 hours of startup, (2) assist in de-inerting the containment atmosphere within 24 hours of shutdown and (3) briefly monthly to assist in performing required tests on the drywell-to-torus vacuum breakers. This latter use may have been replaced by only using the small containment vent and purge isolation valves. Based on the above, the staff concludes that the proposed surveillance and frequency of surveillance is acceptable.

2.3 Conclusion

The staff has reviewed and evaluated the licensee's application for an amendment to the TS, dated November 26, 1986. Based on its evaluation discussed above, the staff approves this proposed amendment. The staff also, as explained in Section 2.1 above, requests that the licensee propose the following additional TS: initiate the preplanned alternate method capable of monitoring the containment radiation if both containment high range radiation monitors are inoperable for more than 7 days.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes to the surveillance requirements. The 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement nor environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 CONCLUSION

The staff 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, and (2) 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 nor to the health and safety of the public.

5.0 REFERENCES

1. Staff Generic Letter 83-02, NUREG-0737 Technical Specifications, January 10, 1983.

2. Staff Generic Letter 83-36, NUREG-0737 Technical Specifications, November 1, 1983.

3. Letter from J. A. Zwolinski (NRC) to P. B. Fiedler (GPUN), Generic Letter 83-02, May 30, 1985.

4. Letter from J. A. Zwolinski (NRC) to P. B. Fiedler (GPUN), Generic Letter 83-36, November 22, 1985.

5. BWR Standard Technical Specifications, NUREG-0123, Revision 4.

6. Letter from J. N. Donohew, Jr. (NRC) to P. B. Fiedler (GPUN), Containment Vent And Purge Isolation On High Radiation, August 4, 1986.

7. Letter from P. B. Fiedler (GPUN) to J. A. Zwolinski (NRC), TSCR No. 152, November 26, 1986.

8. Telephone discussion between J. N. Donohew (NRC) and M. Laggart (GPUN), January 28, 1987.

9. Progress Review Meeting at the site on licensing issues, J. N. Donohew (NRC) and M. Laggart (GPUN), February 2, 1987.

Principal Contributor: J. Donohew, Jr.

Dated: March 31, 1987