December 4, 1992

Docket No. 50-219

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Mr. John J. Barton
Vice President and Director
GPU Nuclear Corporation
Oyster Creek Nuclear Generating Station
Post Office Box 388
Forked River, New Jersey 08731

Dear Mr. Barton:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. M82847)

The Commission has issued the enclosed Amendment No. 160 to Facility Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station, in response to your application dated February 19, 1992.

The amendment deletes the auto-start logic of the Containment Spray System (CSS) by plant modifications to be performed in the 14R refueling outage. In order to achieve this, revisions are necessary to Technical Specifications 3.1 and 3.4 Bases sections; deletion of the instrumentation requirements of Table 3.1-1 Section E; deletion of Containment Spray System from Table 4.1.2 (which lists surveillance test frequencies for Automatic Trip Systems); and deletion of the surveillance requirement for Technical Specification 4.4.C.2 for auto-start actuation test.

These changes are necessary in order to remove the auto-initiation logic from CSS control circuits and associated interfaces with Emergency Water System supply and actuation circuitry, and diesel generator block loading sequence.

A copy of the related Safety Evaluation is enclosed. Also enclosed is the notice of issuance which has been forwarded to the Office of the Federal Register for publication.

Sincerely,

Original signed
by

Alexander W. Dromerick, Senior Project Manager
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. to DPR-16
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

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Mr. John J. Barton
GPU Nuclear Corporation

Oyster Creek Nuclear
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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 FACILITY OPERATING LICENSE

Amendment No. 160
License No. DPR-16

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

   A. The application for amendment by GPU Nuclear Corporation, et al., (the licensee), dated February 19, 1992, 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 No. 160, 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 issuance, to be implemented within 30 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
Project Directorate I-4
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 4, 1992

ATTACHMENT TO LICENSE AMENDMENT NO. 160

FACILITY OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

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

Remove	Insert
Pages 1.0-3	Pages 1.0-3
3.1-4	3.1-4
----	3.1-4a
3.1-11 (Table 3.1.1)	3.1-11 (Table 3.1.1)
3.1-14	3.1-14
3.1-15	3.1-15
3.4-8	3.4-8
4.1-9 (Table 4.1.2)	4.1-9 (Table 4.1.2)
4.4-2	4.4-2

1.14 SECONDARY CONTAINMENT INTEGRITY

Secondary containment integrity means that the reactor building is closed and the following conditions are met:

A. At least one door at each access opening is closed.

B. The standby treatment system is operable.

C. All reactor building ventilation system automatic isolation valves are operable or are secured in the closed position.

1.15 (DELETED)

1.16 RATED FLUX

Rated flux is the neutron flux that corresponds to a steady state power level of 1930 NW(t). Use of the term 100 percent also refers to the 1930 thermal megawatt power level.

1.17 REACTOR THERMAL POWER-TO-WATER

Reactor thermal power-to-water is the sum of (1) the instantaneous integral over the entire fuel clad outer surface of the product of heat transfer area increment and position dependent heat flux and (2) the instantaneous rate of energy deposition by neutron and gamma reactions in all the water and core components except fuel rods in the cylindrical volume defined by the active core height and the inner surface of the core shroud.

1.18 PROTECTIVE INSTRUMENTATION LOGIC DEFINITIONS

A. Instrument Channel

An instrument channel means an arrangement of a sensor and auxiliary equipment required to generate and transmit to a trip system a single trip signal related to the plant parameter monitored by that instrument channel.

B. Trip System

A trip system means an arrangement of instrument channel trip signals and auxiliary equipment required to initiate action to accomplish a protective trip function. A trip system may require one or more instrument channel trip signals related to one or more plant parameters in order to initiate trip system action. Initiation of protective action may require the tripping of a single trip system (e.g., initiation of a core spray loop, automatic depressurization, isolation of an isolation condenser, offgas system isolation, reactor building isolation, standby gas treatment and rod block) or the coincident tripping of two trip systems (e.g., initiation of scram, isolation condenser, reactor isolation, and primary containment isolation).

particular protection instrument is not required; or the plant is placed in the protection or safe condition that the instrument initiates. This is accomplished in a normal manner without subjecting the plant to abnormal operations conditions. The action and out-of-service requirements apply to all instrumentation within a particular function, e.g., if the requirements on any one of the ten scram functions cannot be met then control rods shall be inserted.

The trip level settings not specified in Specification 2.3 have been included in this specification. The bases for these settings are discussed below.

The high drywell pressure trip setting is $\leq$ 3.5 psig. This trip will scram the reactor, initiate core spray, initiate primary containment isolation, initiate automatic depressurization in conjunction with low-low-low-reactor water level, initiate the standby gas treatment system and isolate the reactor building. The scram function shuts the core down during the loss-of-coolant accidents. A steam leak of about 15 gpm and a liquid leak of about 35 gpm from the primary system will cause drywell pressure to reach the scram point; and, therefore, the scram provides protection for breaks greater than the above.

High drywell pressure provides a second means of initiating the core spray to mitigate the consequences of loss-of-coolant accident. Its trip setting of $\leq$3.5 psig initiates the core spray in time to provide adequate core cooling. The break size coverage of high drywell pressure was discussed above. Low-low water level and high drywell pressure in addition to initiating core spray also causes isolation valve closure. These settings are adequate to cause isolation to minimize the offsite dose within required limits.

It is permissible to make the drywell pressure instrument channels inoperable during performance of the integrated primary containment leakage rate test provided the reactor is in the cold shutdown condition. The reason for this is that the Engineered Safety Features, which are effective in case of a LOCA under these conditions, will still be effective because they will be activated (when the Engineered Safety Features system is required as identified in the technical specification of the system) by low-low reactor water level.*

The scram discharge volume has two separate instrument volumes utilized to detect water accumulation. The high water level is based on the design that the water in the SDIV's, as detected by either set of level instruments, shall not be allowed to exceed 29.0 gallons; thereby, permitting 137 control rods to scram. To provide further margin, an accumulation of not more than 14.0 gallons of water, as detected by either instrument volume, will result in a rod block and an alarm. The accumulation of not more than 7.0 gallons of water, as detected in either instrument volume will result in an alarm.

Detailed analyses of transients have shown that sufficient protection is provided by other scrams below 45% power to permit bypassing of the turbine trip and generator load rejection scrams. However, for operational convenience, 40% of rated power has been chosen as the setpoint below which these trips are bypassed. This setpoint is coincident with bypass valve capacity.

A low condenser vacuum scram trip of 20 inches Hg has been provided to protect the main condenser in the event that vacuum is lost. A loss of condenser vacuum would cause the turbine stop valves to close, resulting in a turbine trip transient.

TABLE 3.1.1 PROTECTIVE INSTRUMENTATION REQUIREMENTS (CONT'D)

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*
D. Core Spray								
1. Low-Low Reactor Water Level	**	X(t)	X(t)	X(t)	X	2	2	Consider the respective core spray loop inoperable, and comply with spec. 3.4
2. High Drywell Pressure	≤ 3.5 psig	X(t)	X(t)	X(t)	X	2(k)	2(k)	
3. Low Reactor Pressure (valve permissive)	≥ 285 psig	X(t)	X(t)	X(t)	X	2	2	
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)	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	

TABLE 3.1.1 PROTECTIVE INSTRUMENTATION REQUIREMENTS (CONT'D)

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*
6. IRM Upscale	$\leq$ 108/125 fullscale		X	X		2	3	
7. a) water level high scram discharge volume North	$\leq$ 14 gallons		X(z)	X(z)	X(z)	1	1 per instrum. volume	
b) water level high scram discharge volume South	$\leq$ 14 gallons		X(z)	X(z)	X(z)	1	1 per instrum. volume	
L. Condenser Vacuum Pump Isolation								
1. High Radiation in Main Steam Tunnel	$\leq$ 10 x Normal background		During Startup and Run when vacuum pump operating			2	2	Insert Control Rods
M. Diesel Generator load Sequence Timers	Time delay after energization of relay							
1. CRD pump	60 sec $\pm$ 15%	X	X	X	X	2(m)	1(n)	Consider the pump inoperable and comply with Spec. 3.4.D (see Note q)

TABLE 3.1.1 PROTECTIVE INSTRUMENTATION REQUIREMENTS (CONT'D)

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*
2. Service Water Pump (aa)	120 sec.± 15% (SK1A) 10 sec. ± 15% (SK2A) (SK7A) (SK8A)	X	X	X	X	2(o)	2(p)	Consider the pump inoperable and comply within 7 days (See Note q)
3. Closed Cooling Water Pump (bb)	166 Sec.± 15%	X	X	X	X	2(m)	1(n)	Consider the pump inoperable and comply within 7 days (See Note q)
N. Loss of Power								
a. 4.16KV Emergency ** Bus Undervoltage (Loss of Voltage)		X(ff)	X(ff)	X(ff)	X(ff)	2	1	
b. 4.16 KV Emergency ** Bus undervoltage (Degraded Voltage)		X(ff)	X(ff)	X(ff)	X(ff)	2	3	See note ee

The containment spray system is provided to remove heat energy from the containment in the event of a loss-of-coolant accident. Actuation of the containment spray system in accordance with plant emergency operating procedures ensures that containment and torus pressure and temperature conditions are within the design basis for containment integrity, EQ, and core spray NPSH requirements. The flow from one pump in either loop is more than ample to provide the required heat removal capability(2). The emergency service water system provides cooling to the containment spray heat exchangers and, therefore, is required to provide the ultimate heat sink for the energy release in the event of a loss-of-coolant accident. The emergency service water pumping requirements are those which correspond to containment cooling heat exchanger performance implicit in the containment cooling description. Since the loss-of-coolant accident while in the cold shutdown condition would not require containment spray, the system may be deactivated to permit integrated leak rate testing of the primary containment while the reactor is in the cold shutdown condition.

The control rod drive hydraulic system can provide high pressure coolant injection capability. For break sizes up to 0.002 ft^2, a single control rod drive pump with a flow of 110 gpm is adequate for maintaining the water level nearly five feet above the core, thus alleviating the necessity for auto-relief actuation(3).

The core spray main pump compartments and containment spray pump compartments were provided with water-tight doors(4). Specification 3.4.E ensures that the doors are in place to perform their intended function.

Similarly, since a loss-of-coolant accident when primary containment integrity is not being maintained would not result in pressure build-up in the drywell or torus, the system may be made inoperable under these conditions. This prevents possible personnel injury associated with contact with chromated torus water.

<u>References</u>

1. NEDC-31462P, "Oyster Creek Nuclear Generating Station SAFER/CORECOOL/GESTR-LOCA Loss-of-Coolant Accident Analysis," August 1987.

2. Licensing Application, Amendment 32, Question 3

3. Licensing Application, Amendment 18, Question 1

4. Licensing Application, Amendment 18, Question 4

5. GPUN Topical Report 053, "Thermal Limits with One Core Spray Sparger" December 1988.

6. NEDE-30010A, "Performance Evaluation of the Oyster Creek Core Spray Sparger", January 1984.

7. Letter and enclosed Safety Evaluation, Walter A. Paulson (NRC) to P. B. Fiedler (GPUN), July 20, 1984.

8. APED-5736, "Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards", April 1969.

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	1/20 mo.

C. Containment Cooling System

Item	Frequency
2. Motor-operated valve operability	Every 3 months
3. Pump compartment water-tight doors closed	Once/week and after each entry

D. Emergency Service Water System

1. Pump Operability	Once/month. Also after major maintenance and prior to startup following a refueling outage.

E. Control Rod Drive Hydraulic System

1. Pump Operability	Once/month. Also after major maintenance and prior to startup following a refueling outage.

F. Fire Protection System

1. Pump and Isolation valve operability	Once/month. Also after major maintenance and prior to startup following a refueling outage.

Bases:

It is during major maintenance or repair that a system's design intent may be violated accidentally. Therefore, a functional test is required after every major maintenance operation. During an extended outage, such as a refueling outage, major repair and maintenance may be performed on many systems. To be sure that these repairs on other systems do not encroach unintentionally on critical standby cooling systems, they should be given a functional test prior to startup.

Motor operated pumps, valves and other active devices that are normally on standby should be exercised periodically to make sure that they are free to operate. Motors on pumps should operate long enough to approach equilibrium temperature to ensure there is no overheat problem. Whenever practical, valves should be stroked full length to ensure that nothing impedes their motion. Engineering judgment based on experience and availability analyses of the type presented in Appendix L of the FDSAR indicates that testing these components more often than once a month over a long period of time does not significantly improve the system reliability. Also, at this frequency of testing wearout should not be a problem through the life of the plant.

During tests of the electromatic relief valves, steam from the reactor vessel will be discharged directly to the absorption chamber pool. Scheduling the tests in conjunction with the refueling outage permits the tests to be run at low power, prior to 5 percent power, enhancing the safety of the plant by assuring EMRV operability before higher power levels are reached.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 160

TO FACILITY 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 February 19, 1992, GPU Nuclear Corporation, (GPUN, the licensee), proposed to delete the auto-start logic from the containment spray system (CSS). The licensee's reason for deleting the auto-start logic was based on a commitment made to the NRC in Licensee Event Report (LER) #86-023 as well as an overall concern of inadvertent spray actuation. In that LER, the licensee described the discovery of a single failure mode of the CSS as a result of a power supply failure. Their corrective action for the deficiency included deletion of the auto-start signal for CSS.

The licensee also proposed to delete those technical specifications (TS) and bases which would become obsolete due to the deletion of the auto-start logic. The TS changes include: revisions to TS bases sections 3.1, "Protective Instrumentation," and 3.4, "Emergency Cooling," deletion of the Instrumentation Requirements of Table 3.1.1-Section E, "Containment Spray," deletion of the Containment Spray System from Table 4.1.2, "Minimum Test Frequencies for Trip Systems," and, deletion of the surveillance requirement of TS 4.4.C.2 for auto-start actuation test of the CSS.

2.0 EVALUATION

Presently, the logic for the CSS at Oyster Creek automatically actuates on low-low reactor water level coincident with high drywell pressure in a "one out of two twice" logic. The inputs which complete the logic assure that the automatic logic will function for an event which causes a loss-of-coolant accident (LOCA) large enough to decrease reactor water level below the low-low setpoint and sufficient blowdown to raise drywell pressure to the high setpoint value.

However, it was recognized at the time of the Systematic Evaluation Program (SEP) review that some classes of small diameter pipe breaks (small break LOCA) do not automatically actuate the CSS. Small break LOCAs would not reduce the reactor vessel level sufficiently to reach the low-low level and

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actuate the CSS if operation of the feedwater system continued post LOCA. This zone of manual actuation was described in a GPUN submittal (ref. 1). That submittal was reviewed by the staff and reported on in a safety evaluation report (ref. 2).

The February 19, 1992, GPUN submittal was reviewed by the staff and documented in the subsequent staff safety evaluation. The staff concluded that the analyses submitted by the licensee, have satisfactorily demonstrated the adequacy of the containment functional design. As part of this evaluation, the staff concluded that the two fundamental aspects of this submittal which will change the design are acceptable.

The first obvious change was that there was a portion of the pipe break classes where manual actuation was necessary, the small breaks in the main steam line. The staff had approved manual actuation for main steam line breaks up to and including 2.0 sq.ft., with 2.0 sq.ft. being the largest break size that can blowdown without initiating the automatic spray system. The 0.75 sq.ft. main steam line break produces the highest drywell temperature. For larger pipe breaks (greater than 2.0 sq.ft.), the CSS would receive an auto-start signal. Secondly, the acceptance of manual starting of the CSS, the staff had previously accepted a 10 minute manual initiation of the containment sprays. For purposes of this evaluation, the staff has not reevaluated the bases for these two critical assumptions which have been previously accepted by the staff in reference 2.

The licensee has requested that the acceptable zone for manual activation be expanded to include the entire spectrum of possible pipe breaks. In other words, the automatic actuation signal for the sprays would be removed. Therefore, the sprays would be manually actuated for all accident events requiring containment spray. GPUN had submitted results from their analysis which indicated that the elimination of the automatic spray start signal will not have an effect on the Oyster Creek Environmental Qualification (EQ) temperature profile or the containment design basis. The containment design basis and EQ temperature profile were not affected by this change due to the small main steam line breaks (0.01 and 0.75 sq.ft.) never being sufficiently large enough to blowdown reactor water level below the low-low setpoint and actuate the containment spray (ref. 1). Therefore, automatic CSS was never given credit for in past safety analysis and the assumption made for CSS was that the sprays were manually started in 10 minutes.

The licensee stated that a 0.75 sq.ft. main steam line break produces the highest calculated temperature within the drywell. The staff had performed confirmatory analysis for the main steam line break, including 0.01 and 0.75 sq. ft. main steam line pipe breaks and concluded that GPUN's calculations were acceptable (ref. 2). Therefore, the analysis submitted by GPUN in reference 1 and approved by the staff in reference 2 bounds the maximum calculated temperature profile used for the EQ temperature profile since the small pipe break classes are not affected by this change to the CSS.

To support the elimination of the automatic signal, a series of analyses were conducted by the licensee. These analyses focused on the large break LOCA since that was the region that would be most affected by the elimination of the automatic signal to CSS. The large break considered was the bounding double ended guillotine break area of 6.2 sq.ft. of a recirculation loop. Comparative analyses were performed considering the presence of the automatic signal and a parallel case considering manual actuation. A comparison of the two cases for the double ended break case is provided below.

PARAMETER	CASE 1 AUTOMATIC SPRAYS	CASE 2 MANUAL SPRAYS AT 10 MIN
Drywell Pressure, psig	38.4 at 5 sec	38.4 at 5 sec
Torus Pressure, psig	26.6 at 99 sec	27.0 at 612 sec
Drywell Vapor Temperature, °F	282.7 at 5 sec	282.7 at 5 sec
Torus Liquid Temperature, °F	158.8 at 10,890 sec	159.4 at 10,530 sec

The data from GPUN's analysis shows that there is almost no effect on the maximum containment parameter values when the automatic CSS logic is eliminated.

A 0.1 sq.ft. recirculation line break was also evaluated in a similar fashion. The results of this comparative analysis is provided below.

PARAMETER	CASE 3 AUTOMATIC SPRAYS	CASE 4 MANUAL SPRAYS AT 10 MIN
Drywell Pressure, psig	20.6 at 351 sec	20.8 at 355 sec
Torus Pressure, psig	19.0 at 413 sec	19.2 at 598 sec
Drywell Vapor Temperature, °F	259.8 at 351 sec	260.1 at 430 sec
Torus Liquid Temperature, °F	153.2 at 18,310 sec	153.2 at 18,270 sec

As can be seen, the peak values are only slightly higher due to the elimination of the CSS automatic start logic. Manual actuation of the sprays shows that the drywell and torus pressure peak values are greater. However, the peak pressure increases are only 0.2 psi. These differences are considered to be minor. The important consideration is that the peak values are within the design pressure limits for both the drywell and torus.

It is important to note that the sprays were not credited for any design-basis accident (DBA) transients in confirmatory analysis made by the staff in reference 2. Also, the Emergency Operating Procedures (EOPs) which are currently in place at Oyster Creek do contain criteria for the start of the CSS. The licensee stated that the EOPs currently contain sufficient criteria for the operator to determine when to start the CSS and no change to that criteria is required for manual activation of CSS as a result of a postulated large pipe break LOCA. The EOPs currently require starting the CSS on high torus pressure (12 psig) or before reaching 281°F in the drywell and within the Drywell Spray Initiation Limit, or high concentrations of hydrogen and oxygen.

Based on the above findings, the staff finds that the licensee's proposed change is acceptable. In addition, the staff finds the revision to the TS and bases acceptable since they are necessary to properly reflect this change in the TS.

3.0 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.

4.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact have been prepared and published in the Federal Register on October 14, 1992 (57 FR 47125). Accordingly, based upon the environmental assessment, the staff has determined that the issuance of the amendment will not have a significant effect on the quality of the human environment.

5.0 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.

6.0 REFERENCES

1. Letter from GPUN to the staff titled "Oyster Creek Containment Temperature Profile for Environmental Qualification of Equipment," dated November 1, 1980.

2. Staff SER titled Evaluation Report on Containment Pressure and Heat Removal Capability, SEP TOPIC VI-3 and Mass and Energy Release for Possible Pipe Break Inside Containment, SEP TOPIC VI-2.D for the Oyster Creek Nuclear Power Plant.

Principal Contributors: A. D'Angelo, K. Bristow

Date: December 4, 1992

UNITED STATES NUCLEAR REGULATORY COMMISSION

GPU NUCLEAR CORPORATION

DOCKET NO. 50-219

NOTICE OF ISSUANCE OF AMENDMENT TO

FACILITY OPERATING LICENSE

The U.S. Nuclear Regulatory Commission (Commission) has issued Amendment No. 160 to Facility Operating License No. DPR-16 issued to GPU Nuclear Corporation (the licensee), which revised the Technical Specifications for operation of the Oyster Creek Nuclear Generating Station located in Ocean County, New Jersey. The amendment is effective as of the date of issuance.

The amendment modified the Technical Specifications to delete the auto-start logic of the Containment Spray System (CSS) by plant modifications to be performed in the 14R refueling outage. In order to achieve this, revisions were necessary to Technical Specifications 3.1 and 3.4 Bases sections; deletion of the instrumentation requirements of Table 3.1-1 Section E; deletion of Containment Spray System from Table 4.1.2 (which lists surveillance test frequencies for Automatic Trip Systems); and deletion of the surveillance requirement for Technical Specification 4.4.C.2 for auto-start actuation test.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

Notice of Consideration of Issuance of Amendment and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER

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on March 12, 1992 (57 FR 8785). No request for a hearing or petition for leave to intervene was filed following this notice.

The Commission has prepared an Environmental Assessment related to the action and has determined not to prepare an environmental impact statement. Based upon the environmental assessment, the Commission has concluded that the issuance of this amendment will not have a significant effect on the quality of the human environment (57 FR 47125).

For further details with respect to the action see (1) the application for amendment dated February 19, 1992, (2) Amendment No. 160 to License No. DPR-16, (3) the Commission's related Safety Evaluation, and (4) the Commission's Environmental Assessment. All of these items are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street NW., Washington, DC 20555 and at the local public document room located at the Ocean County Library, Reference Department, 101 Washington Street, Toms River, New Jersey 08753. A copy of items (2), (3) and (4) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Document Control Desk.

Dated at Rockville, Maryland this 4th day of December 1992

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

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