Docket file

September 13, 1993

Docket No. 50-219

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

No commitment JEQ

Dear Mr. Barton:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. M81093)

The Commission has issued the enclosed Amendment No.165 to Facility Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station, in response to your application dated July 22, 1991, as supplemented February 14, 1992 August 19, 1992, and July 12, 1993.

The amendment revises Technical Specification 5.2.A and Bases to change the current containment drywell design pressure of 62 psig and design temperature of 175°F to the new design pressure of 44 psig and new design temperature of 292°F. Related changes to Technical Specification Bases are also revised. Unrelated changes to the Bases of Technical Specifications 3.4 and 3.5 were also made.

A copy of the related Safety Evaluation is also 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. 165 to DPR-16
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

OGC
8/24/93

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

September 13, 1993

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Vice President and Director
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Mr. John J. Barton
GPU Nuclear Corporation

Oyster Creek Nuclear
Generating Station

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

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. 165
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 July 22, 1991, as supplemented February 14, 1992, August 19, 1992, and July 12, 1993, 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.

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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) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 165 , 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: September 13, 1993

ATTACHMENT TO LICENSE AMENDMENT NO. 165

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
3.4-8	3.4-8
3.5-8	3.5-8
4.5-12	4.5-12
4.5-16	4.5-16
5.2-1	5.2-1

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 required would not result in pressure build-up in the drywell or torus, the containment spray system may be made inoperable under these conditions.

<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.

rod worth such that a rod drop would not result in any fuel damage. In addition, in the unlikely event that an excursion did occur, the reactor building and standby gas treatment system, which shall be operational during this time, offer a sufficient barrier to keep off-site doses well below 10 CFR 100 limits.

The absorption chamber water volume provides the heat sink for the reactor coolant system energy released following the loss-of-coolant accident. The core spray pumps and containment spray pumps are located in the corner rooms and due to their proximity to the torus, the ambient temperature in those rooms could rise during the design basis accident. Calculations(7) made, assuming an initial torus water temperature of 100°F and a minimum water volume of 82,000 ft^3, indicate that the corner room ambient temperature would not exceed the core spray and containment spray pump motor operating temperature limits and, therefore, would not adversely affect the long-term core cooling capability. The maximum water volume limit allows for an operating range without significantly affecting accident analyses with respect to free air volume in the absorption chamber. For example, the containment capability(8) with a maximum water volume of 92,000 ft^3 is reduced by not more than 5.5% metal-water reaction below the capability with 82,000 ft^3.

Experimental data indicate that excessive steam condensing loads can be avoided if the peak temperature of the suppression pool is maintained below 160°F during any period of relief valve operation with sonic conditions at the discharge exit. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

The technical specifications allow for torus repair work or inspections that might require draining of the suppression pool when all irradiated fuel is removed or when the potential for draining the reactor vessel has been minimized. This specification also provides assurance that the irradiated fuel has an adequate cooling water supply for normal and emergency conditions with the reactor mode switch in shutdown or refuel whenever the suppression pool is drained for inspection or repair.

The purpose of the vacuum relief valves is to equalize the pressure between the drywell and suppression chamber, and suppression chamber and reactor building so that the containment external design pressure limits are not exceeded.

The vacuum relief system from the reactor building to the pressure suppression chamber consists of two 100% vacuum relief breaker subsystems (2 parallel sets of 2 valves in series). Operation of either subsystem will maintain the containment external pressure less than the 2 psi external design pressure of the drywell; the external design pressure of the suppression chamber is 1 psi (FDSAR Amendment 15, Section 11).

The capacity of the 14 suppression chamber to drywell vacuum relief valves is sized to limit the external pressure of the drywell during post-accident drywell cooling operations to the design limit of

Basis: In the event of a loss-of-coolant accident, the peak drywell pressure would be 38 psig which would rapidly reduce to 20 psig within 100 seconds following the pipe break. The total time the drywell pressure would be above 35 psig is calculated to be about 7 seconds. Following the pipe break, absorption chamber pressure rises to 20 psig within 8 seconds, equalizes with drywell pressure at 25 psig within 60 seconds and thereafter rapidly decays with the drywell pressure decay.(1)

The design pressures of the drywell and absorption chamber are 62 psig and 35 psig, respectively.(2) The original calculated 38 psig peak drywell pressure was subsequently reconfirmed.(3) A 15% margin was applied to revise the drywell design pressure to 44 psig. The design leak rate is 0.5%/day at a pressure of 35 psig. As pointed out above, the pressure response of the drywell and absorption chamber following an accident would be the same after about 60 seconds. Based on the calculated primary containment pressure response discussed above and the absorption chamber design pressure, primary containment pre-operational test pressures were chosen. Also, based on the primary containment pressure response and the fact that the drywell and absorption chamber function as a unit, the primary containment will be tested as a unit rather than testing the individual components separately.

The design basis loss-of-coolant accident was evaluated at the primary containment maximum allowable accident leak rate of 1.0%/day at 35 psig. The analysis showed that with this leak rate and a standby gas treatment system filter efficiency of 90 percent for halogens, 95% for particulates, and assuming the fission product release fractions stated in TID-14844, the maximum total whole body passing cloud dose is about 10 rem and the maximum total thyroid dose is about 139 rem at the site boundary considering fumigation conditions over an exposure duration of two hours. The resultant doses that would occur for the duration of the accident at the low population distance of 2 miles are lower than those stated due to the variability of meteorological conditions that would be expected to occur over a 30-day period. Thus, the doses reported are the maximum that would be expected in the unlikely event of a design basis loss-of-coolant accident. These doses are also based on the assumption of no holdup in the secondary containment resulting in a direct release of fission product from the primary containment through the filters and stack to the environs. Therefore, the specified primary containment leak rate and filter efficiency are conservative and provide margin between expected offsite doses and 10 CFR 100 guideline limits.

Although the dose calculations suggest that the allowable test leak rate could be allowed to increase to about 2.0%/day before the guideline thyroid dose limit given in 10 CFR 100 would be exceeded, establishing the limit of 1.0%/day provides an adequate margin of safety to assure the health and safety of the general public. It is further considered that the allowable leak rate should not deviate significantly from the containment design value to take advantage of the design leak-tightness capability of the structure over its service lifetime. Additional margin to maintain the containment in the "as

The drywell exterior was coated with Firebar D prior to concrete pouring during construction. The Firebar D separated the drywell steel plate from the concrete. After installation, the drywell liner was heated and expanded to compress the Firebar D to supply a gap between the steel drywell and the concrete. The gap prevents contact of the drywell wall with the concrete which might cause excessive local stresses during drywell expansion in a loss-of-coolant accident. The surveillance program is being conducted to demonstrate that the Firebar D will maintain its integrity and not deteriorate throughout plant life. The surveillance frequency is adequate to detect any deterioration tendency of the material.[8]

The operability of the instrument line flow check valves are demonstrated to assure isolation capability for excess flow and to assure the operability of the instrument sensor when required.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be continually monitored and also observed during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

References

(1) Licensing Application, Amendment 32, Question 3
(2) FDSAR, Volume I, Section V-1.1
(3) GE-NE 770-07-1090, "Oyster Creek LOCA Drywell Pressure Response," February 1991
(4) Technical Safety Guide, "Reactor Containment Leakage Testing and Surveillance Requirements," USAEC Division of Safety Standards, Revised Draft, December 15, 1966.
(5) FDSAR, Volume I, Sections V-1.5 and V-1.6
(6) FDSAR, Volume I, Sections V-1.6 and XIII-3.4
(7) FDSAR, Volume I, Section XIII-2
(8) Licensing Application, Amendment 11, Question III-18

5.2 CONTAINMENT

A. The primary containment shall be of the pressure suppression type having a drywell and an absorption chamber constructed of steel. The drywell shall have a volume of approximately 180,000 ft^3 and conforms to the ASME Boiler and Pressure Vessel Code, Section VIII, for an internal pressure of 44 psig at 292°F and an external pressure of 2 psig at 150°F to 205°F. The absorption chamber shall have a total volume of approximately 210,000 ft^3. It is designed to conform to ASME Boiler and Pressure Vessel Code, Section VIII, for an internal pressure of 35 psig at 150°F and an external pressure of 1 psig at 150°F.

B. Penetrations added to the primary containment shall be designed in accordance with standards set forth in Section V-1.5 of the Facility Description and Safety Analysis Report. Piping passing through such penetrations shall have isolation valves in accordance with standards set forth in Section V-1.6 of the Facility Description and Safety Analysis Report.

BASIS

The drywell pressure of 44 psig is based upon a conservatively calculated peak drywell pressure of 38.1 psig plus an added 15% allowance. The calculated peak pressure results from a design basis loss of coolant accident (DBLOCA). The corresponding coincident drywell temperature of 292° F is the saturated steam temperature of the containment atmosphere for the 44 psig pressure. The specified coincident pressure and temperature condition represent the bounding case for the structural pressure/temperature design of the drywell.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 165

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 July 22, 1991 (Reference 1), as supplemented February 14, 1992 (Reference 2), August 19, 1992 (Reference 3) and July 12, 1993 (Reference 8), GPU Nuclear Corporation (GPUN/the licensee), proposed to lower the containment drywell design pressure from 62 psig to 44 psig and change the containment drywell design temperature from 175°F to 292°F. The Technical Specifications (TS) 5.2.A and related Bases changes to TS 4.5 and T.S. 5.2.A were revised to reflect this change. These changes are necessary as part of the drywell mitigation program at Oyster Creek. The licensee also proposed unrelated editorial changes to TS 3.4 and 3.5. We find these editorial changes acceptable.

2.0 EVALUATION

The existing containment drywell pressure of 62 psig was based on simulation tests conducted to confirm the design adequacy of the Bodega Bay Plant (Reference 4). The actual value was established by adding 10 psig to the estimated 52 psig for Bodega Bay. The 10 psig was added for margin and conservatism. However, differences in the Bodega Bay and Oyster Creek plants suggest that the peak drywell pressure at Oyster Creek would be less than the Bodega Bay.

2.1 Reevaluation of the Drywell Pressure

For an updated estimate of the drywell design pressure a double-ended guillotine break of a recirculation loop pipe was assumed for a Mark I pressure suppression containment. The TRACG and RELAP5 MOD 3 codes were used for the estimation of the reactor vessel blowdown (References 5 and 6). Both codes yield best estimate values.

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2.1.1 Primary System Blowdown (Mass and Energy Release)

2.1.1.1 Code Validation

RELAP5 has not been designed for Boiling Water Reactors (BWRs) but in this application, i.e., vessel blowdown, it can be used regardless of the type of reactor. The nodalization was designed to account for the Oyster Creek geometry as well as anticipated phenomena. The RELAP5 blowdown model was compared to actual test data from Marviken (Reference 7). This comparison was used to determine a multiplier in the flow rate to address uncertainties. This multiplier was set at 1.3; as a result, the calculated integrated mass and energy entering into the containment is in excess of what can actually occur.

The TRACG code results were compared to simple blowdown tests, scaled integral BWR tests like the Two Loop Test Apparatus (TLTA) and full size reactor vessel tests like Marviken.

For both codes each blowdown was divided into two regimes: when subcooled conditions prevail in the break and for two phase conditions. For each flow regime an average break mass flow was obtained and a model multiplier was developed. The maximum multiplier for the TRACG was 1.25 which was then applied to estimate the break flow.

2.1.1.2 Results

Comparison to the Marviken test results shows that the estimated mass flow rates, when the above multipliers are used, were conservative with respect to the containment estimate. Comparison of the RELAP5 and TRACG results are very close with respect to flow rate as well as time for the peak flow.

The NRC staff reviewed the licensee's blowdown analysis. On the basis of the review, the staff finds the licensee's estimate acceptable. This finding is based on the fact that GPUN used two independent computer codes which were validated with actual blowdown tests and use of conservative flow multipliers.

2.1.2 Drywell Response (Drywell Pressure and Temperature)

A series of analyses using blowdown mass and energy releases to containment discussed in Section 2.1.1 were evaluated by the licensee to support the proposed changes. The cases were a large break loss-of-coolant accident (LOCA), small break LOCA, main steam line break (MSLB), and a stuck open relief valve. Each analysis used the CONTEMPT EI-28C (Energy Incorporated version of CONTEMPT-LT 28) and M3CPT (General Electric Co. model for containment pressure and temperature analysis) computer codes.

GPUN provided a discussion on how they were in compliance with Section 6.2.1.1.C, "Pressure Suppression Type BWR Containments," of the Standard Review Plan (SRP) by their analysis of the containment pressure temperature response. The staff has provided guidance to licensees within the SRP which described an analytical approach which has been found acceptable to the staff. This discussion of compliance with SRP Section 6.2.1.1.C and the slightly different approach by GPUN was to identify the differences in the analytical methods and to compare their analysis methods, in this case CONTEMPT-EI, with that of the staff's method, CONTEMPT-LT 28. The licensee provided this information in GPUN letter dated August 19, 1992, Subject: Oyster Creek Containment Peak Pressure Analysis.

CONTEMPT-EI 28C was used by GPUN to generate containment pressure and temperature profiles to evaluate the limiting containment response. In addition, investigations were performed to evaluate the impact of eliminating the automatic spray initiation feature. Comparisons were made between containment pressure and temperature profiles utilizing automatic spray actuation and those which had the sprays manually actuated 10 minutes after the break. It was further assumed that one loop of the containment spray system (CSS) was placed in the dynamic test mode to provide heat removal from the torus pool after drywell pressure has been reduced by manual spray actuation.

A steam line break was chosen as an analysis case since this type of break defines the equipment qualification temperature profile for electrical equipment inside the drywell. The licensee submitted their Environmental Qualification Analysis (EQ) on October 28, 1980. The EQ temperature analysis does not change as a result of this amendment because a delay of 10 minutes for manual spray starting was assumed in the October 28, 1980 analysis. For calculation of peak drywell temperature the main steam line break is the most limiting transient. Peak drywell temperature and pressure was calculated by the licensee to be 311°F at 19 psig resulting from a 0.75 ft^2 main steam line break. In spite of the automatic initiation feature, it was necessary to manually initiate the sprays for the most limiting steam line break considered in the analysis because the low-low level setpoint was not reached if feedwater was available. As in the original analysis, manual actuation was assumed to occur in 10 minutes. In the case of the stuck open relief valve, the energy released is deposited directly into the torus pool. This release mode fails to cause a large enough increase in drywell pressure to automatically actuate the CSS. Since the automatic option to initiate the CSS is not triggered due to a MSLB or a stuck open relief valve, these cases will be unaffected by the deletion of the auto-start logic for the CSS. The original analysis remains valid which includes manual actuation of the containment sprays in 10 minutes.

The large break LOCA considered was a double-ended guillotine break of a recirculation loop. Two different models were used to evaluate the containment response. The first used assumptions intended to maximize the containment pressure profile, and was used to calculate drywell pressure, torus pressure, and drywell atmospheric temperature. The second model maximized the heatup of the torus water pool to evaluate the effects on core spray and containment spray pumps net positive suction head (NPSH) limits. The cases that used automatic spray initiation assumed that the sprays came on at the start of the LOCA. In reality, the CSS would not actuate for about 85 seconds because of the loss of offsite power assumption for this kind of accident. This delay has no effect on the peak parameters which are given below. Consistent with the original analyses, manual spray actuation was assumed to occur in 10 minutes.

PARAMETER	CASE 1 AUTOMATIC SPRAYS	CASE 2 MANUAL SPRAYS AT 10 MIN
Drywell Pressure, psig	38.1 at 5 sec	38.1 at 5 sec
Torus Pressure, psig	26.6 at 99 sec	27.0 at 612 sec
Drywell Vapor Temperature, °F	285.0 at 5 sec	285.0 at 5 sec
Torus Liquid Temperature, °F	158.8 at 10,890 sec	159.4 at 10,530 sec

For the two cases the peak values of drywell vapor temperature are virtually identical with or without the automatic spray actuation. The torus pool temperature profile also shows little difference between the two cases. In addition, neither case resulted in a condition which would violate the NPSH limits for the core spray or containment spray pumps for the maximum flow as required by the Appendix K LOCA analysis.

A 0.1 ft^2 recirculation line break was evaluated as a small break LOCA by the licensee. The combination of the break and the automatic depressurization system (ADS) actuation decreases the reactor pressure which reduces the blowdown to almost zero within 500 seconds. No additional steam is added after this point due to the subcooling effect of the core spray. Again, consistent with the original design analysis, manual sprays were assumed to actuate in 10 minutes. A summary of the data is as follows:

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 above, the peak values are only slightly higher due to the elimination of the CSS automatic start logic. Case 4 drywell and torus pressure peak values are greater than Case 3. This is because the containment spray system will have an effect more quickly due to the smaller mass and energy addition rates to the containment for the size break. The maximum calculated drywell pressure was determined to be 38.1 psig in Case 2. This calculated pressure is less than the new drywell design pressure of 44 psig which the licensee proposes to use for their structural evaluation of the drywell and the resultant corrosion. The drywell vapor temperature is within the drywell design temperature of 292°F as defined in the July 12, 1993 letter. The peak torus pool temperature is below that of the design basis accident LOCA, therefore, adequate NPSH is assured. The drywell design pressure and temperature is based on the double-ended guillotine break of the recirculation loop maximum calculated pressure and temperature plus a 15% margin, which is 44 psig at 292°F. Although the design temperature of the drywell is below the 311°F at 19 psig condition resulting from a main steam line break, the maximum structural load on the drywell is generated by the higher pressure (38.1 psig plus 15% margin) resulting from a recirculation loop break. Therefore, the higher pressure caused by a recirculation line break becomes the controlling structural load on the drywell.

Based on the above comparisons and the finding that the licensee's analyses have been performed in accordance with the Standard Review Plan, the staff finds that the licensee's proposed changes for the drywell maximum calculated pressure and temperature are acceptable. The acceptance of the 10 minute spray actuation time was not reevaluated since this assumption was part of the original analyses which had been found acceptable. In addition, the staff finds the revision to the TS and Bases acceptable.

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 September 10, 1993 (58 FR 47768). 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 to USNRC from J. J. Barton, GPU Nuclear Corporation, "Oyster Creek Nuclear Generating Station Docket No. 50-219, Technical Specification Change Request No. 198," dated July 22, 1991.

2. Letter to USNRC from J. J. Barton, GPU Nuclear Corporation, "Oyster Creek Nuclear Generating Station, Technical Specification Change Request No. 198, Reactor Vessel Blowdown Multiplier," February 14, 1992.

3. Letter to USNRC from J. C. Devine, Jr., GPU Nuclear Corporation, "Oyster Creek Nuclear Generating Station Docket No. 50-219, Facility Operating License No. DPR-16, Oyster Creek Containment Peak Pressure Analysis," August 19, 1992.

4. Preliminary Hazards Summary Report, Bodega Bay Atomic Park Unit No. 1, Docket No. 50-205, December 28, 1962.

5. NUREG/CR-4127-1, 2, & 3, EPRI NP-3987-1, 2, & 3, GEAP-30875-1, 2, & 3, "BWR Full Integral Simulation Test (FIST) Program: TRAC-BWR Model Development," 1985.

6. NUREG/CR-4312, EGG-2396, "Appendix A RELAP5 Input Data Requirements Prepared for Release of RELAP5 MOD 3," EG&G Idaho Falls, ID, January 1990.

7. Letter from D. Spaughterbeck, Intermountain Technologies, to N. Trikouros GPU Nuclear, "Transmittal of REALP-MARVIKEN Simulation Tests (FIST) Program," DCS-636-90.

8. Letter to USNRC from J. J. Barton, GPU Nuclear Corporation, Oyster Creek Nuclear Generating Station (OCNGS) Docket No. 50-219, Technical Specification Change Request No. 198, Rev. 1 - Drywell Pressure Temperature Change.

Principal Contributors: A. D'Angelo
L. Lois

Date: September 13, 1993

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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. 165 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 revises Technical Specification 5.2.A to change the current containment drywell design pressure of 62 psig to the new design pressure of 44 psig. Related changes to Technical Specification Bases are also revised. Unrelated changes to the Bases of Technical Specifications 3.4 and 3.5 were also made.

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 on August 8, 1991 (56 FR 37732). No request for a hearing or petition for leave to intervene was filed following this notice.

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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 (58 FR 47768).

For further details with respect to the action see (1) the application for amendment dated July 22, 1991, as supplemented February 14, 1992, August 19, 1992, and July 12, 1993 (2) Amendment No. 165 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.

Dated at Rockville, Maryland this 13th day of September 1993.

FOR THE NUCLEAR REGULATORY COMMISSION

Alexander W. Dromerick

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