DISTRIBUTION
W. Dooly, DR
R. Engelken, CO (2)
~~Shapar, OGC~~
. Buchanan, ORNL
H. McAlduff, OROO
W. Gilbert
W. Kenna, NMS
T. Laughlin, DTIE
PDR
Docket File
DR Reading
DRL Reading
Branch, Reading, R. Boyd
ACRS (3), R. DeYoung
F. Schroeder, R. Schemel
D. Skovholt, T. Wambach
R. Vollmer, S. Teets

NOV 5 - 1971

Docket No. 50-219

Jersey Central Power & Light Company
ATTN: Mr. R. H. Sims, Vice President
Madison Avenue at Punch Bowl Road
Morristown, New Jersey 07960

Gentlemen:

Amendment No. 3 to Provisional Operating License No. DPR-16 is enclosed. The amendment authorizes the operation of the Oyster Creek Nuclear Power Plant Unit No. 1 at steady state power levels up to 1930 megawatts (thermal).

It is our understanding that you plan to transmit to us for our use and distribution a reissued set of Technical Specifications that will incorporate all the changes made to date to the Oyster Creek Technical Specifications.

A copy of the report of the ACRS, dated June 18, 1971, is attached as Appendix A to the Safety Evaluation. There are some matters identified in this report that require further action on your part and review and follow-up by us. Accordingly, we refer you to Section 7.0 of our Safety Evaluation for a listing of the reports that we require to be submitted on a timely basis so that we may evaluate your resolution of these matters. Please provide a schedule indicating the dates by which you estimate these reports will be submitted.

We understand that you are presently revising your Environmental Monitoring Program and will submit this revised program for our review in the near future.

A copy of our Safety Evaluation and a copy of the Federal Register Notice are also enclosed.

Sincerely,

/S/

Peter A. Morris, Director
Division of Reactor Licensing

Enclosures:
1. Amendment No. 3
2. Safety Evaluation
   Appendix A - ACRS Report
3. Federal Register Notice

Dispatched 11/12/71

cc: George F. Trowbridge, Esquire
~~Shaw, Pittman, Potts, Trowbridge & Madden~~

SEE ATTACHED YELLOW FOR OTHER CONCURRENCES

OFFICE ▶	DRL JVW	DRL	OGC	DRL	DRL	DRL
SURNAME ▶	TWambach:pdl STeets	RJSchemel		DJSkovholt	FSchroeder	PAMorris LB
DATE ▶	10/1/71	10/1/71	10/ /71	10/19/71	11/2/71	11/2/71

Form AEC-318 (Rev. 9-53) U.S. GOVERNMENT PRINTING OFFICE : 1969—O-364-598 of these matters. Please provide a schedule indicating the dates by which you estimate these reports will be submitted.

We understand that you are presently revising your Environmental Monitoring Program and will submit this revised program for our review in the near future.

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Peter A. Morris, Director
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cc: George F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge
& Madden
910 - 17th Street, N. W.
Washington, D. C. 20006

OFFICE ▶	DRL	DRL	OGC	DRL	DRL	DRL
SURNAME ▶	TWambach:ew SATeets	RJSchemel		DJSkovholt	FSchroeder	PAMorris
DATE ▶	7/15/71	7/16/71	7/16/71	7/ /71	7/ /71	7/ /71

]

Distribution
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R. Boyd
R. DeYoung

Docket No. 50-219

Jersey Central Power & Light Company
ATTN: Mr. R. H. Sims, Vice President
Madison Avenue at Punch Bowl Road
Morristown, New Jersey 07960

Gentlemen:

Amendment No. 3 to Provisional Operating License No. DPR-16 is enclosed. The amendment authorizes the operation of the Oyster Creek Nuclear Power Plant Unit No. 1 at steady state power levels up to 1930 megawatts (thermal). ~~Initial operation will be limited to 1865 MWt until installation of the fifth relief valve. Prior to increasing power above 1865 MWt, an Atomic Energy Commission representative will inspect the relief valve installation and review the testing results.~~ OUT

OUT ~~We understand that, in addition to performing the Power Test Program as described in Amendment No. 65 Section D, you will also perform all of the Phase 3 tests except, No. 14 Recirculation Pumps and No. 17 Turbine Trip, upon increasing power initially to 1865 MWt.~~

It is ~~also~~ OUT our understanding that you plan to transmit to us for our use and distribution a reissued set of Technical Specifications that will incorporate all the changes made to date to the Oyster Creek Technical Specifications.

A copy of the report of the ACRS, dated June 18, 1971, is attached as Appendix A to the Safety Evaluation. There are some matters identified in this report that require further action on your part and review and follow-up by us. Accordingly, we refer you to Section 7.0 of our Safety Evaluation for a listing of the reports that we require to be submitted on a timely basis so that we may evaluate your resolution

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

November 5, 1971

Docket No. 50-219

Jersey Central Power & Light Company
ATTN: Mr. R. H. Sims, Vice President
Madison Avenue at Punch Bowl Road
Morristown, New Jersey 07960

Gentlemen:

Amendment No. 3 to Provisional Operating License No. DPR-16 is enclosed. The amendment authorizes the operation of the Oyster Creek Nuclear Power Plant Unit No. 1 at steady state power levels up to 1930 megawatts (thermal).

It is our understanding that you plan to transmit to us for our use and distribution a reissued set of Technical Specifications that will incorporate all the changes made to date to the Oyster Creek Technical Specifications.

A copy of the report of the ACRS, dated June 18, 1971, is attached as Appendix A to the Safety Evaluation. There are some matters identified in this report that require further action on your part and review and follow-up by us. Accordingly, we refer you to Section 7.0 of our Safety Evaluation for a listing of the reports that we require to be submitted on a timely basis so that we may evaluate your resolution of these matters. Please provide a schedule indicating the dates by which you estimate these reports will be submitted.

We understand that you are presently revising your Environmental Monitoring Program and will submit this revised program for our review in the near future.

A copy of our Safety Evaluation and a copy of the Federal Register Notice are also enclosed.

Sincerely,

Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Enclosures:
1. Amendment No. 3
2. Safety Evaluation
   Appendix A - ACRS Report
3. Federal Register Notice

cc: George F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge & Madden

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

JERSEY CENTRAL POWER & LIGHT COMPANY

DOCKET NO. 50-219

AMENDED FACILITY LICENSE

Amendment No. 3
License NO. DPR-16

The Atomic Energy Commission ("the Commission") having found that:

a. The application, as amended, complies with the requirements of the Atomic Energy Act of 1954, as amended ("the Act"), and the rules and regulations of the Commission set forth in 10 CFR, Chapter I;

b. Construction of the facility has been substantially completed in conformity with Provisional Construction Permit No. CPPR-15, the application, the provisions of the Act and the rules and regulations of the Commission;

c. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

d. There is reasonable assurance (i) that the facility can be operated at power levels not in excess of 1930 megawatts (thermal) in accordance with this license without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission;

e. The applicant is technically and financially qualified to engage in the activities authorized by this operating license, in accordance with the rules and regulations of the Commission;

f. The applicable provisions of 10 CFR Part 140 have been satisfied; and

g. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the public.

Provisional Operating License No. DPR-16, as amended, is hereby amended in its entirety to read as follows:

1. This license applies to the Oyster Creek Nuclear Power Plant Unit No. 1, a single cycle, forced circulation, boiling light-water reactor, and electric generating equipment (the facility). The facility is located on Jersey Central's Oyster Creek site

in Lacey Township, Ocean County, New Jersey, approximately sixty miles south of Newark and forty-five miles east of Philadelphia, Pennsylvania, and is described in license application Amendment No. 3, "Facility Description and Safety Analysis Report", as supplemented and amended (Amendments No. 4 through 65 and 67).

2. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses Jersey Central:

   A. Pursuant to Section 104b of the Atomic Energy Act of 1954, as amended ("the Act"), and 10 CFR Part 50, "Licensing of Production and Utilization Facilities", to possess, use and operate the facility as a utilization facility at the designated location on Jersey Central's Oyster Creek site;

   B. Pursuant to the Act and 10 CFR Part 70, "Special Nuclear Material", to receive, possess and use at any one time up to 3600 kilograms of contained uranium 235 in connection with operation of the facility;

   C. Pursuant to the Act and 10 CFR Part 30, "Rules of General Applicability to Licensing of Byproduct Materials", to receive, possess and use 12,000 curies of antimony 124 and 15 curies of americium 241 as Sb-Be and Am-Be neutron sources, 1 curie of Co-60 and 5 curies of Cs-137 as sealed sources; and

   D. Pursuant to the Act and Parts 30 and 70, to possess, but not to separate, such byproduct and special nuclear material as may be produced by operation of the facility.

3. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations: 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Sections 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

   A. <u>Maximum Power Level</u>

   Jersey Central is authorized to operate the facility at steady state power levels up to a maximum of 1930 megawatts (thermal).

B. Technical Specifications

The Technical Specifications contained in Appendix A, issued with Provisional Operating License No. DPR-16 dated April 9, 1969, including previously issued changes to the Technical Specifications, and as modified by Attachment A appended hereto (designated as Change No. 7), are hereby incorporated into this license. Jersey Central may operate the facility at power levels not in excess of 1930 megawatts (thermal) in accordance with the Technical Specifications and may make changes therein only when authorized by the Commission in accordance with the provisions of Section 50.59 of 10 CFR Part 50, "Licensing of Production and Utilization Facilities".

C. Reports

Jersey Central shall make certain reports in accordance with the requirements of the Technical Specifications.

D. Records

Jersey Central shall keep facility operating records in accordance with the requirements of the Technical Specifications.

4. This license is effective as of the date of issuance and shall expire at midnight April 9, 1972.

FOR THE ATOMIC ENERGY COMMISSION

Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
Attachment A (Change No. 7
to the Technical Specifications)

Date of Issuance: November 5, 1971

ATTACHMENT A TO AMENDMENT NO. 3

JERSEY CENTRAL POWER & LIGHT COMPANY

LICENSE NO. DPR-16

DOCKET NO. 50-219

CHANGE NO. 7 TO APPENDIX A

Page 1.0-2

Paragraph 1.12	Add the following: "Following the first refueling outage, the time between successive tests or surveillance shall not exceed 20 months."

Page 1.0-3

Paragraph 1.15A.	Replace "Exceeds a Limiting Safety System Setting as" with "Results in a Limiting Safety System Setting less conservative than that"
Paragraph 1.15C.	Replace "Causes any unplanned reactor trip, or" with "Causes any uncontrolled or unanticipated change in reactivity, or"
Paragraph 1.16	Change "1690 MWt" to "1930 MWt". Add a second sentence: "The use of the term 100 percent also refers to the 1930 thermal megawatt power level."

Page 2.1-1

Paragraph 2.1.B.	Change "5 percent" to "10 percent". Change "320 MWt" to "354 MWt".
Paragraph 2.1.C.	Change "2.5" to "1.75".

Page 2.1-2

(1)	In the second sentence of the second paragraph, change "APED 3892(1)" to "APED 5286(1)".
(2)	In the second and last sentences of the fourth paragraph, replace "1015 psia" with "1035 psia".

Page 2.1-3

(1) Replace the first paragraph with the following:

"The power shape assumed in the calculation of these curves results in a total peaking factor of 3.03. The axial distribution of power is such that the power peak occurs above the core midplane. This distribution results in a smaller CHFR and thus a more conservative calculation of the safety limit curve than that which would be obtained with a more realistic axial power shape in which the power peak occurs at or below the core midplane. The actual power distribution in the core is established by control rod sequencing and is monitored continuously by the in-core LPRM system. The total power peaking factor is to be less than 3.03 at rated power. It is possible that during temporary control rod manipulation or near the end of core life when it might be desirable to delay a refueling outage, a peaking factor greater than 3.03 could result at power levels less than rated. However, to maintain applicability of the safety limit curve, the safety limit will be lowered according to the equation given in Figure 2.1.1 for those short periods during which the total peaking factor might exceed 3.03."

(2) In the first sentence of the second paragraph, replace "322.8°F at 1015 psia" with "334°F at 1035 psia".

(3) In the second sentence of the fifth paragraph, replace "5%" with "10%".

(4) In the sixth sentence of the fifth paragraph, replace "320 MWt (19% of rated)" with "354 MWt (18.3% of rated)".

Page 2.1-4

(1) At the end of the first paragraph, add reference 9 to the other references.

(2) In the first and last sentences of the second paragraph, replace "2.5 seconds" with "1.75 seconds".

Page 2.1-5

(1) Replace reference (1) with the following:

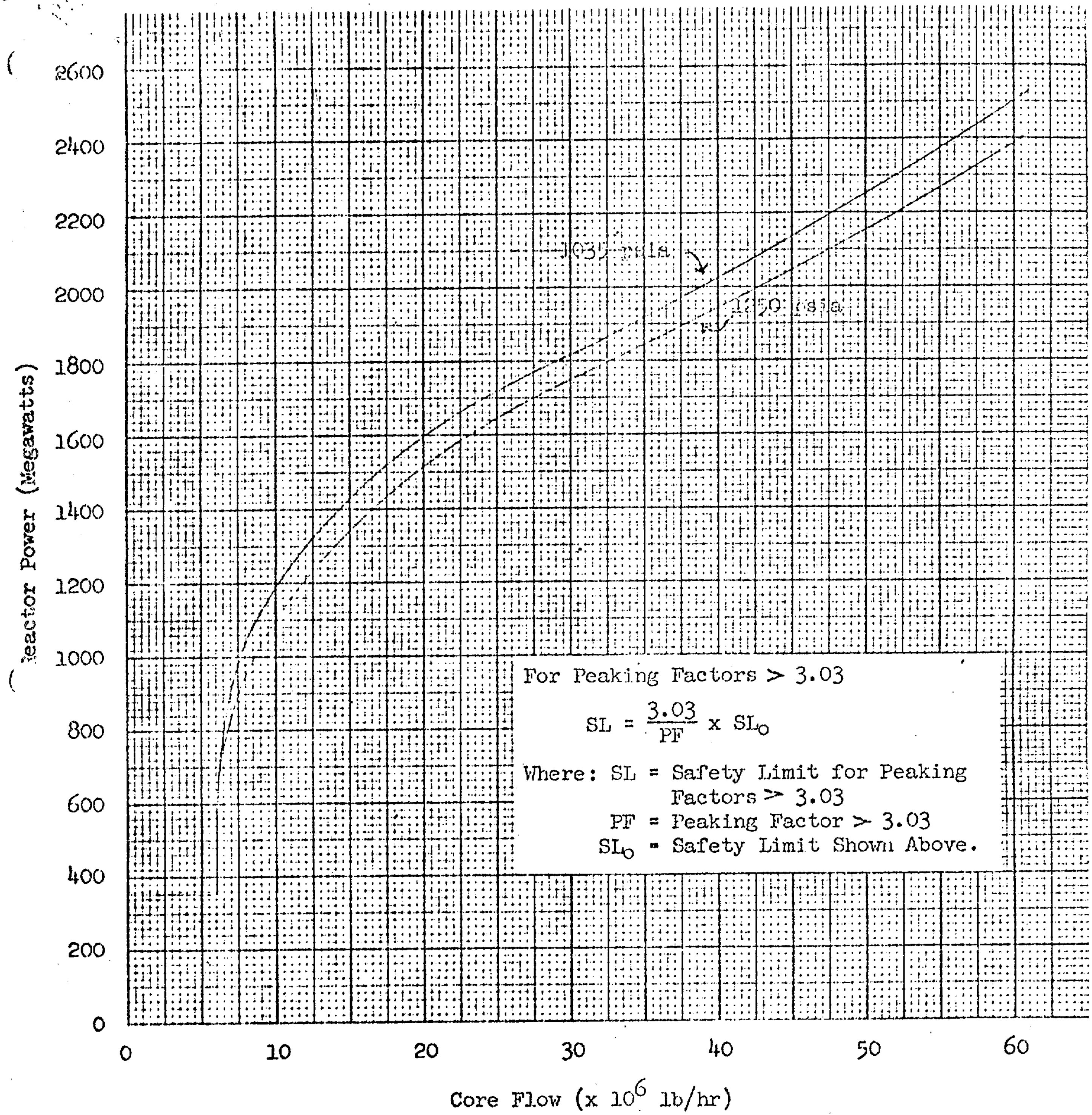
"(1) J. M. Healzer, J. R. Hench, E. Janssen, and S. Levy, 'Design Basis for Critical Heat Flux Conditions in Boiling Water Reactors', APED 5286, September 1966."

(2) Add the following reference to the end of the list:

"(9) Licensing Application Amendment 65, Sections B.IV, B.VIII, and B.XI."

Page 2.1-6

(1) Replace Figure 2.1.1, Fuel Cladding Integrity Safety Limit (Revised December 2, 1970) with Figure 2.1.1, Fuel Cladding Integrity Safety Limit (Revised 11/5/71).



NOTES:

1. Rated Power is 1930 MWt.
2. Rated Flow is 61.0×10^6 lb/hr.
3. Total Peaking Factor is 3.03.
4. Core Pressure $\geq$ 600 psia.
5. Reactor Water Level is > 10 ft. 7 in. above the top of the active fuel.

FIGURE 2.1.1

FUEL CLADDING INTEGRITY SAFETY LIMIT
(Revised 11/5/71)

Pages 2.2-1 & 2.2-2

(1) Replace the fifth paragraph of the Bases (last paragraph on page 2.2-1 continued on page 2.2-2) with the following:

"The normal operating pressure of the reactor coolant system is 1020 psig. For the turbine trip and loss of electrical load transients, the turbine trip scram or the generator load rejection scram in combination with the turbine bypass system limit reactor pressure to less than 1120 psig.(2) In addition, pressure relief valves have been provided to reduce the probability of the safety valves operating in the event that the turbine bypass should fail. These valves and reactor scram limit reactor pressure to less than 1183 psig.(2) Finally, the safety valves are sized to keep the reactor coolant system pressure below 1375 psig with no credit taken for scram or any pressure relieving devices other than the safety valves.(3)"

Page 2.2-2

(1) Replace reference (2) with the following:

"(2) License Application Amendment 65, Sections B.IV and B.XI."

(2) Replace reference (3) with the following:

"(3) License Application Amendment 65, Section B.IV."

Page 2.3-1

(1) In Specification 2.3, replace the Limiting Safety System Setting for (1) Neutron Flux, Scram (a) APRM with the following:

"For recirculation flow, $W \leq 61 \times 10^6$ lb/hr:

$\leq [(1.12 \times 10^{-6})W+51.76]$ percent of rated neutron flux for total peaking factors ≤ 3.03.

$\leq [(1.12 \times 10^{-6})W+51.76] \left(\frac{3.03}{PF}\right)$ percent of rated neutron flux for total peaking factors, PF $>$ 3.03.

For recirculation flow, W $>$ 61 x 10^6 lb/hr:

$\leq$ 120 percent of rated neutron flux for total peaking factors $\leq$ 3.03.

$\leq 120 \left(\frac{3.03}{PF}\right)$ percent of rated neutron flux for total peaking factors, PF $>$ 3.03."

(2) In Specification 2.3, replace the Limiting Safety System Setting for (2) Neutron Flux, Control Rod Block (a) APRM with the following:

"For recirculation flow, W $\leq$ 61 x 10^6 lb/hr:

$\leq [(1.12 \times 10^{-6})W+37.72]$ percent of rated neutron flux for total peaking factors $\leq$ 3.03.

$\leq [(1.12 \times 10^{-6})W+37.72] \left(\frac{3.03}{PF}\right)$ percent of rated neutron flux for total peaking factors, PF $>$ 3.03.

For recirculation flow, W $>$ 61 x 10^6 lb/hr:

$\leq$ 106 percent of rated neutron flux for total peaking factors $\leq$ 3.03.

$\leq 106 \left(\frac{3.03}{PF}\right)$ percent of rated neutron flux for total peaking factors, PF $>$ 3.03."

(3) In Specification 2.3, change the Limiting Safety System Setting for (3) Reactor High Pressure, Scram from " $\leq$ 1060 psig" to " $\leq$ 1070 psig".

(4) In Specification 2.3, Limiting Safety System Setting for (5) Reactor High Pressure, Isolation Condenser Initiation, replace "1060" with "1070".

Page 2.3-2

(1) Delete "*W = recirculation flow in lb/hr

$W \leq 66.2 \times 10^6$ lb/hr".

(2) In the first sentence of the last paragraph on this page, replace "(66.2×10^6 lb/hr)" with "(61.0×10^6 lb/hr) or greater".

Page 2.3-3

(1) In the fourth paragraph, delete the second sentence and replace it with the following:

"At conditions of rated flow or greater, the rod block is initiated at 106% power where MCHFR is greater than 1.4."

(2) In the last sentence of the fourth paragraph, replace "70% of rated" with "61% of rated".

(3) After the fourth paragraph, add the following paragraph:

"The safety curves of Figure 2.1.1 are based on a total peaking factor of 3.03, and these curves are to be adjusted downward (by the equation shown on Figure 2.1.1) in the unusual event of higher peaking factors. Also, to insure MCHFR's greater than 1.0 during expected transients, neutron flux, scram and control rod block settings must be correspondingly reduced. The equations describing these setpoints make allowance for peaking factors greater than 3.03 by reducing the setpoints at rated neutron flux by the ratio of 3.03/PF.

(4) In the last partial sentence at the bottom of the page, replace "27% thermal margin" with "22% thermal margin".

Page 2.3-4

(1) In the first line, replace "19% of rated" with "18.3% of rated".

(2) Replace the second paragraph ("The settings on the reactor high pressure scram, . . .") with the following:

"The settings on the reactor high pressure scram, anticipatory scrams, reactor coolant system relief valves and isolation condenser have been established to assure never reaching the reactor coolant system pressure safety limit as well as assuring the system pressure does not exceed the range of the fuel cladding integrity safety limit. In addition, the APRM neutron flux scram and the turbine bypass system also provide protection for these safety limits, e.g., turbine trip and loss of electrical load transients[(8)]. In addition to preventing power operation above 1070 psig, the pressure scram backs up the other scrams for these transients and other steam line isolation type transients. With the addition of the anticipatory scrams, the transient analysis for operation at 1930 MWt shows that the turbine trip with failure of the bypass system transient is the worst case transient with respect to peak pressure. Analysis of this transient shows that the relief valves limit the peak pressure to 1183 psig[(9)], well below the 1250 psig range of applicability of the fuel cladding integrity safety limit and the 1375 psig reactor coolant system pressure safety limit. Actuation of the isolation condenser during these transients removes the reactor decay heat without further loss of reactor coolant thus protecting the reactor water level safety limit."

Page 2.3-5

(1) In the first complete sentence of the page ("The safety valves are sized . . ."), replace "1860 MWt" with "1930 MWt".

(2) In the fourth complete sentence of the page ("With the safety valves set . . ."), replace "1315 psig" with "1301 psig", replace "1340 psig" with "1315 psig", and replace "35 psi" with "60 psi".

(3) Delete the last two sentences of the second paragraph and replace them with the following sentence:

"With the scrams set at 10% valve closure, there is no increase in neutron flux and the peak pressure is limited to 1136 psig[10]."

Page 2.3-6

(1) In the second sentence of the second paragraph, replace "1690 MWt" with "1930 MWt".

(2) At the bottom of the page, replace references (7) through (12) with the following:

"(7) License Application Amendment 65, Section B.VII.4.

(8) License Application Amendment 65, Section B.XI.

(9) License Application Amendment 65, Section B.IV.

(10) License Application Amendment 65, Section B.IV.

(11) License Application Amendment 65, Section B.XI.

(12) License Application Amendment 65, Section B.IV."

Page 3.1-1

Section 3.1.B.2 In the last sentence of this specification, replace "70% of rated power" with "61% of rated power".

Page 3.1-4

Delete the paragraph that begins "Bypass of the turbine . . ." and replace it with the following paragraph:

"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."

Page 3.1-5

After the paragraph ending "the normal flow rate of 3.2 x 10^5 lb/hr", insert the following new paragraph:

"The setting of ten times the stack release limit for isolation of the air-ejector offgas line is to permit the operator to perform normal, immediate remedial action if the stack limit is exceeded. The time necessary for this action would be extremely short when considering the annual averaging which is allowed under 10 CFR 20.106, and, therefore, would produce insignificant effects on doses to the public."

Page 3.1-6

In the fifth sentence of the paragraph beginning "Specification 3.1.B.1 . . .", replace "70%" in two places with "61%".

Page 3.1-10

In Table 3.1.1., Function I "Offgas System Isolation", under Trip Setting, change to read as follows:

"$\leq$10 x Stack Release Limit (see 3.6-A.1.)"

Page 3.1-12a

In note j. to Table 3.1.1, change "45%" to "40%".

Page 3.2-1

Section 3.2.B.2. After the sentence beginning "The rod worth minimizer . . .", add the following:

"Except for low power physics tests, control rod patterns shall be established so that the maximum worth of any operable control rod shall be less than 2.5 percent Δ k."

Page 3.2-2

Replace Specification 3.2.D. in its entirety with the following:

"D. Reactivity Anomalies

The difference between an observed and predicted control rod inventory shall not exceed the equivalent of one percent in reactivity. If this limit is exceeded and the discrepancy cannot be explained, the reactor shall be brought to the cold, shutdown condition by normal orderly shutdown procedure. Operation shall not be permitted until the cause has been evaluated and appropriate corrective action has been completed. The AEC shall be notified within 24 hours of this situation in accordance with Specification 6.6.B."

Page 3.2-4

(1) After the paragraph ending "Thus requiring operation of the RWM below 10% rated power is conservative.", insert the following paragraph:

"The analysis of a control rod drop accident assuming a maximum rod worth of 2.5 percent Δ k has been performed and the results found to be acceptable."

Page 3.2-6

In the last sentence of the paragraph beginning "The standby liquid control system . . .", replace "50-100 minutes" with "60-120 minutes".

Page 3.2-8

In the eighth and ninth lines of the last paragraph, delete ". . . at any base, equilibrium core state to predicted rod inventory at that state", and replace it with ". . . with expected inventory based on appropriately corrected past data".

Pages 3.2-8 & 3.2-9

Delete the sentence which begins "During an initial period, ..." starting on the bottom of page 3.2-8 and ending at the top of page 3.2-9.

Page 3.2-9

Change the first full sentence on this page to read as follows:

"Experience at Oyster Creek and other operating BWR's indicates that the control rod inventory should be predictable to the equivalent of one percent in reactivity."

Page 3.4-1

In Specification 3.4.B.1, change "Four electromatic relief valves" to "Five electromatic relief valves".

Page 3.4-2

Replace the present Specification 3.4.B.2 with the following:

"2. At any time when there are only four operable electromatic relief valves, the reactor may remain in operation provided that its maximum steady state power level is limited to 1865 MWt. If there are only three operable electromatic relief valves, the reactor may remain in operation at a maximum steady-state power level of 1690 MWt for a time not to exceed eight hours."

Page 3.4.3A

Replace the second sentence of "Bases" with the following:

"Based on the loss of coolant analysis for the worst line break, a core spray of at least 3400 gpm is required within 35 seconds to assure effective core cooling[(1)]."

Page 3.4-4

In the second sentence of the fourth paragraph, change "Three of the four relief valves" to "Three of the five relief valves".

Page 3.4-5

Replace reference (1) with the following:

"(1) License Application Amendment 65, Section B.VI.6."

Pages 3.6-1 & 3.6-2

Replace Section 3.6, Applicability, Objective, and Specification 3.6.A. and B., inclusive with the following:

"3.6 Radioactive Effluents

Applicability: Applies to the radioactive effluents of the facility.

Objective: To assure that radioactive material is not released to the environment in an uncontrolled manner and to assure that the radioactive concentrations of any material released is kept to a practical minimum and in any event, within the limits of 10 CFR 20.

Specification: A. Plant Stack Effluents

(1) The maximum release rate of gross activity, except iodines and particulates with half lives longer than eight days, shall be limited in accordance with the following equation:

$$Q = \frac{0.21}{\bar{E}} \text{ Ci/sec.}$$

where Q is the stack release rate (Ci/sec) of gross activity and $\bar{E}$ is the average gamma energy per disintegration (MeV/dis).

(2) The maximum release rate of iodines and particulates with half lives longer than eight days shall not exceed 4 μCi/sec.

(3) Radiogases released from the stack shall be continuously monitored except for the short time during monitor filter changes. If this specification cannot be met, the reactor shall be placed in the isolated condition.

B. Discharge Canal Effluents

(1) The release of radioactive liquid effluents shall be limited such that the concentration of radionuclides in the discharge canal at the site boundary shall not at any time exceed the concentrations given in Appendix B, Table II, Column 2, of 10 CFR 20 and notes 1 through 5 thereto.

(2) Radioactive liquid effluent being released into the discharge canal shall be continuously monitored, or, if the monitor is inoperative, two independent samples of any tank to be discharged shall be taken, one prior to discharge and one near the completion of discharge, and two station personnel shall independently check valving prior to discharge of radioactive liquid effluents.

Page 3.6-2

(1) In Specification 3.6.D., Reactor Coolant Radioactivity, change "20 µCi/cc" to "8.0 µCi/gm".

(2) Add Specification 3.6.E. as follows:

"E. Liquid Radioactive Waste Control

Equipment installed for the treatment of liquid wastes shall be used if release of an untreated batch would result in concentrations in excess of 20 percent of the limits given in Section 3.6.B.(1)."

Page 3.6-3

(1) Add the following paragraph after the paragraph ending ". . . and buoyancy of a continuously emitted plume.":

"Independent dose calculations for several locations offsite have been made by the AEC staff. The method utilized onsite meteorological data developed by the licensee and utilized diffusion assumptions appropriate to the site. The method is described in Section 7-5.2.5 of 'Meteorology and Atomic Energy - 1968', equation 7.63 being used. The results of these calculations were equivalent to those generated by the licensee provided the average gamma energy per disintegration for the assumed noble gas mixture with a 30-minute holdup is 0.7 MeV per disintegration. Based on these calculations, a maximum release rate limit of gross activity, except for iodines and particulates with half lives longer than eight days, in the amount of $0.21/\bar{E}$ curies per second will not result in offsite annual doses in excess of the limits specified in 10 CFR 20. The $\bar{E}$ determination need consider only the average gamma energy per disintegration since the controlling whole body dose is due to the cloud passage over the receptor and not cloud submersion in which the beta dose could be additive."

(2) Delete the paragraphs on this page that begin as follows:

"The maximum dose rate . . ."

"Based on a dose rate . . ."

"The instantaneous release rate . . ."

"A short-term release rate . . ."

Page 3.6-4

(1) Delete the first two lines on the page.

(2) Delete the first complete paragraph on the page that begins "Therefore, a limit of 13 μc/sec . . .".

(3) In the second sentence of the paragraph beginning "It is recognized . . .", replace "(5)" with "(Section 4.6)".

(4) In the last sentence of the paragraph beginning "It is recognized . . .", insert the word "not" between the words "that" and "averaging".

(5) In the paragraph beginning "The radioactive liquid effluents . . .", delete the last word in the ninth line "gross". Also delete the last sentence of this paragraph.

(6) Delete the last partial paragraph on this page that begins "Specification 3.6.B.1.a. requires . . .".

Page 3.6-5

Delete all paragraphs on this page and replace with the following:

"The radioactivity concentration limits for the liquid effluents set forth in Specification 3.6.B.(1) are based on the limits contained in 10 CFR 20, Appendix B, Table II, Column 2. By excluding averaging for any time period, a margin is maintained between releases made in conformance with this limit and the limit specified in 10 CFR 20.106.

When discharging on the basis of the limit for a mixture of unidentified isotopes ($1 \times 10^{-7} \mu Ci/cc$), an estimate of radionuclide concentrations in aquatic biota has been made that correlates the resultant activity levels in the biota with the water limits for each isotope given in 10 CFR 20, Appendix B, Table II, Column 2. Based on conditions of minimum bay flushing and with a circulating water flow rate of 450,000 gpm, the predicted concentration adjacent to the outlet of the discharge canal has a value of $1.5 \times 10^{-12} \mu Ci/cc$ per μCi/day discharged.(7,8) This represents the concentration in the discharge canal undiluted by dispersion in the bay and based on this value, the average μCi/day release rate that will yield a discharge canal concentration not exceeding $1 \times 10^{-7} \mu Ci/cc$ is approximately $6.7 \times 10^{4} \mu Ci$/day or about 25 curies/year. Assuming such releases, which is equivalent to releasing continuously at the limit given in this specification, estimates are presented for clams, crabs, and finfish in reference 9. The estimated concentration is less in each case than that permitted in drinking water for that radioisotope. There are several factors which tend to make the estimates higher than would be expected. First, the estimates of bay concentrations are based on dispersion experiments conducted during a period of minimal dilution. Average dilution should be greater. Second, the recirculation effects assumed are greater than those calculated by the mathematical model that was used to estimate the effects of recirculation.

When discharging on the basis of the limits for identified isotopes, consideration must be given to the reconcentration factors cited in reference 9. A major consideration is that with all batch releases being less than the limit given in 10 CFR 20, Appendix B, Table II, Column 2 for each radioisotope, all periods of time when batch releases are not being made will apply in offsetting the effect of reconcentration. Verification of the adequacy of these limits will be obtained by performance of the environmental monitoring program (Section 4.6). If the releases ever reach a level such that the biota sampling shows an increase in the background levels, such measurements will provide a basis for adjusting the isotopic limits long before the effect in the environment is of any concern for permissible dose."

Page 3.6-6

(1) Delete the first paragraph on this page.

(2) Replace the third paragraph on this page with the following:

"The primary coolant radioactivity concentration limit of 8.0 µCi total iodine per gram of water was calculated based on a steamline-break accident which is isolated in 10.5 seconds. For this accident analysis, all the iodine in the mass of coolant released in this time period is assumed to be released to the atmosphere at the top of the turbine building (30 meters). By limiting the thyroid dose at the site boundary to a maximum of 30 Rem, the iodine concentration in the primary coolant is back-calculated assuming fumigation meteorology, Pasquill Type F at 1 m/sec. The iodine concentration in the primary coolant resulting from this analysis is 8.4 µCi/gm."

The required use of the equipment installed for the treatment of liquid waste is specified for the purpose of limiting the liquid effluent radioactivity levels to a practical minimum. Twenty percent of the Technical Specification limit for release of unidentified isotopes is equivalent to the guide value for design objectives given in the Proposed Appendix I to 10 CFR 50.

Page 3.8-2

In the first complete sentence on this page, change "1690 MWt" to "1930 MWt".

Page 4.1-5

(1) In "<u>Check</u>" column of Table 4.1.1, for items 11 and 12, "APRM Scram Trips" and "APRM Rod Blocks", replace "NA" with "Note 2".

(2) At the bottom of the page, add the following:

"Note 2: At least daily during reactor power operation, the reactor neutron flux peaking factor shall be estimated and the flow-referenced APRM scram and rod block settings shall be adjusted, if necessary, as specified in Section 2.3, Specifications (1)(a) and (2)(a)."

Page 4.1-6A

Add the following items to Table 4.1.1:

Instrument Channel	Check	Calibrate	Test	Remarks
"25. Recirc. Loop Flow	NA	Each refueling outage	NA	By application of test pressure
26. Low Reactor Pressure, Core Spray Valve Permissive	NA	Every 3 months	Every 3 mos.	By application of test pressure."

Page 4.2-1

(1) Replace Specification C in its entirety with the following:

"C. (1) After each major refueling outage and prior to resuming power operation, all operable control rods shall be scram time tested from the fully withdrawn position with reactor pressure above 800 psig.

(2) Following each reactor scram from rated pressure, the mean 90% insertion time shall be determined for eight selected rods. If the mean 90% insertion time of the selected control rod drives does not fall within the range of 2.4 to 3.1 seconds or the measured scram time of any one drive for 90% insertion does not fall within the range of 1.9 to 3.6 seconds, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is maintained.

(3) Following any outage not initiated by a reactor scram, eight rods shall be scram tested with reactor pressure above 800 psig provided these have not been measured in six months. The same criteria of 4.2.C.(2) shall apply."

(2) Add Specification F as follows:

"F. At specific power operating conditions, the actual control rod configuration will be compared with the expected configuration based upon appropriately corrected past data. This comparison shall be made every equivalent full power month. The initial rod inventory measurement performed when equilibrium conditions are established after a refueling or major core alteration will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle.

Page 4.2-3

(1) Replace the second paragraph on this page with the following two paragraphs:

"The scram insertion times for all control rods[3] will be determined at the time of each refueling outage. The scram times generated at each refueling outage when compared to scram times previously recorded gives a measurement of the functional effects of deterioration for each control rod drive. The more frequent scram insertion time measurements of eight selected rods are performed on a representative sample basis to monitor performance and give an early indication of possible deterioration and required maintenance. The times given for the eight-rod tests are based on the testing experience of control rod drives which were known to be in good condition.

The weekly control rod exercise test serves as a periodic check against deterioration of the control rod system. Experience with this control rod system has indicated that weekly tests are adequate, and that rods which move by drive pressure will scram when required as the pressure applied is much higher. The frequency of exercising the control rods has been increased under the conditions of two or more control rods which are valved out of service in order to provide even further assurance of the reliability of the remaining control rods."

(2) Add a paragraph at the bottom of the page as follows:

"The control rod inventory check provides detection of reactivity anomalies and additional verification of control rod position at a frequency which is compatible with the time and power varying parameters being checked."

Page 4.5-2

Change Specification 4.5.B.2. to read as follows:

The allowable test leak rate L_t (20) shall not exceed the lesser value established as follows:

$$L_t\ (20) = 1.0\ L_m\ (20)/L_m\ (35)$$

or

$$L_t\ (20) = 1.0 \left[\frac{P_t\ (20)}{P_t\ (35)}\right]^{1/2}$$

where P_t (20) and P_t (35) are measured values in absolute pressure

Page 4.5-4

In Specification I.2, change the time for required closure of the Main Steamline Isolation Valve from " $\leq$ 10 seconds" to " $\geq$ 3 seconds and $\leq$ 10 seconds".

Page 4.5-6

Add a new surveillance specification as follows:

"O. <u>Instrument Line Flow Check Valves Surveillance</u>

The capability of each instrument line flow check valve to isolate shall be tested at least once in every period between refueling outages. Each time an instrument line is returned to service after any condition which could have produced a pressure or flow disturbance in that line, the open position of the flow check valve in that line shall be verified. Such conditions include:

Leakage at instrument fittings and valves

Venting an instrument or instrument line

Isolating an instrument

Flushing or draining an instrument"

Page 4.5-6A

(1) In the third line of the paragraph beginning "The design basis . . .", change "1.25%/day" to "1.0%/day".

(2) In the last line on this page, change "6.0 Rem" to "10 Rem".

Page 4.5-7

(1) In the first line on the page, change "170 Rem" to "139 Rem".

(2) Delete the paragraph beginning "The maximum allowable test leak rate . . .".

(3) In the second line of the paragraph beginning "Although the dose calculations suggest . . .", change "1.75%" to "2.0%".

Page 4.5-9

At the top of the page after "the required closing time", add the following:

"The minimum time of 3 seconds is based on the transient analysis of the isolation valve closure that shows the pressure peak 76 psig below the lowest safety valve setting. The maximum time of 10 seconds provides a 0.5 second margin to the 10.5 seconds that is assumed for the main steamline break dose calculations."

Page 4.5-10

Add a paragraph as follows:

"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."

Page 4.5-11

Reference (3), delete "TID-20583, Leakage Characteristics of Steel Containment Vessels and the Analysis of Leakage Rate Determinations" and replace with "Deleted".

Page 4.6-1

(1) Replace the Objective for Section 4.6 with the following:

"Objective: To verify that discharge of radioactive effluents to the environment is kept to a practical minimum and, in any event, within the limits of 10 CFR 20."

(2) Replace Specification B with the following:

"B. (1) Stack Release

(a) Station records of gross stack release rate of gaseous activity and meteorological conditions shall be maintained on an hourly basis to assure that the specified rates are not exceeded, to provide data for calculating offsite dose and to yield information concerning general integrity of the fuel cladding.

(b) Within one month after issuance of these specifications and within one month following refuelings, an isotopic analysis will be made of a gaseous activity release sample which identifies at least 90 percent of the total activity. From this sample, a ratio of long-lived ($>$ 8 day half life) and short-lived activity will be established.

(c) Samples of off-gas will be taken at least every 96 hours and gross ratio of long-lived ($>$ 8 day half life) and short-lived activity determined.

(d) An isotopic analysis of off-gas will be performed monthly unless the ratio determined in (c) differs from the ratio established by the previous isotopic analysis by more than 20 percent. If this occurs, a new isotopic analysis shall be performed.

(e) Gaseous release of tritium shall be measured at least quarterly.

(f) Station records of stack release of iodines and particulates with half lives greater than eight days shall be maintained on the basis of all filter cartridges counted.

(g) These cartridges shall be analyzed weekly for gross alpha, beta and gamma activity, Ba-140, La-140 and I-131 when the iodine or particulate release rate is less than 4 percent of the maximum release rate given in Specification 3.6.A(2), otherwise the cartridges shall be removed for analysis twice a week.

(h) When the gross gaseous release rate exceeds 1 percent of the maximum release rate given in Specification 3.6.A(1) and the average daily gross activity release rate increased by 50 percent over the previous full operating day, the cartridges shall be analyzed to determine the release rate increase for iodines and particulates.

(i) An isotopic analysis of iodines and particulate radionuclides shall be performed at least quarterly.

(2) Liquid Release

(a) Station records shall be maintained of the radioactive concentration and volume before dilution of each batch of liquid effluent released and of the average dilution flow and length of time over which each discharge occurred.

(b) A weekly proportional composite* of samples of each batch discharged during the week shall be analyzed for gross alpha, beta and gamma activity, Ba-140, La-140, I-131, dissolved gases such as Xe-133 and other shorter lived radionuclides (half lives of 15 days or less) which are associated with routes of potential exposure to man.

*A proportional composite is one in which the quantity of liquid added to the composite is proportioned to the quantity of liquid in the batch that was released.

(c) A monthly proportional composite of samples of each batch discharged during the month shall be analyzed for gross alpha, beta and gamma activity, tritium and the principal gamma emitting fission and activation products in the sample, including longer lived radionuclides associated with routes of potential exposure to man. The analysis should account for at least 90 percent of the total activity, exclusive of tritium and dissolved gases, and should include at least Cs-137, Cs-134, Co-60, Co-58, Cr-51, Mn-54 and Zn-65.

(d) A quarterly proportional composite shall be analyzed for Sr-90.

(e) Each batch of liquid effluent released shall be analyzed for gross alpha, beta and gamma activity and the results recorded. Should there be any unexplained significant change in gross alpha, beta or gamma activity from previous isotopic analyses, a new isotopic analysis shall be performed.

(f) If a batch is to be released on an identified radionuclide basis, the analysis shall also include a gamma scan. If gamma peaks different from those determined by previous isotopic analyses are found or if the mixture concentration is greater than 10 percent of the mixture MPC, a new isotopic analysis shall be performed and recorded.

(3) <u>Environmental Program</u>

The environmental program described in Section B.II.6 of Amendment 65 to the Application for Reactor Operating License shall be conducted. The sampling frequencies specified in Table B-II-1 of Amendment 65 shall be adhered to as closely as conditions permit."

(3) Add Specification 4.6.E. as follows:

"E. The operability of all equipment installed for the treatment of liquid wastes shall be verified at least once per quarter."

(4) Replace the second paragraph under Basis that begins "Sampling and analysis . . ." with the following:

"Continuous monitoring of the gaseous and collection of the particulate stack effluents provides the means for determining that the limits of Specification 3.6.A are not exceeded and for recording the actual levels of radioactivity that are being released from the stack. The frequencies of filter and cartridge analyses and isotopic analyses are specified to assure proper identification of the isotopes being released. The sampling and analysis of each batch of the radioactive liquid effluent provide the means for determining the release rate to the discharge canal to assure the limits of Specification 3.6.B are not exceeded. The isotopic analyses of the weekly and monthly proportional composites of liquid waste samples provide the data for recording and reporting the average concentrations of radioactivity and total radioactivity released from the discharge canal. These isotopic analyses shall also provide the normal means for calibrating gross alpha, beta and gamma analyses that are used to determine the concentration of batch for discharge on an unidentified basis. More frequent isotopic analyses shall be required in conformance with 4.6.B.2.(e) & (f) to assure that the calibration of gross counts has not been altered by a change in the mixture of radioisotopes.

The release of effluents on an identified radionuclide basis shall be based on the isotopic analysis of a typical waste batch provided that the gross counting analysis and the gamma scan indicate no significant change in the mixture constituents or the resultant mixture after dilution does not exceed 10 percent of the mixture MPC. If either of these two conditions occur, an isotopic analysis of the batch to be discharged shall be performed.

A minimum dilution factor for the isotopic mixture shall be determined using the following formula:

$$\text{Minimum D.F.} = \frac{C_1}{MPC_1} + \frac{C_2}{MPC_2} + \ldots + \frac{C_n}{MPC_n}$$

Where: C_1 = concentration of isotope 1, etc.

MPC_1 = MPC of isotope 1 from Appendix B, Table II, Column 2, 10 CFR 20, etc.

C_n will normally be the concentration of unidentified activity remaining after identification of isotopes.

This dilution factor can be expressed as a MPC for the isotopic mixture thus:

$$\text{Mixture MPC} = \frac{\text{gross concentration}}{\text{Minimum D.F.}}$$

This mixture MPC shall be used to determine the appropriate discharge rates and dilution for waste batches but can only be used for the particular mixture as determined above."

Page 6.3-1

In the last line, replace "when appropriate to the AEC" with "in accordance with the requirements of Section 6.6.B".

Page 6.4-1

To the last sentence, add "in accordance with the requirements of Section 6.6.B".

Page 6.6-1

(1) Delete the first four lines of Section 6.6 and replace with the following:

"In addition to reports required by applicable regulations, the following information shall also be provided:

A. A routine operating report shall be prepared for each six-month period to January 1 and July 1 of each year. Such reports are to be submitted within 60 days after the end of each reporting period. The following information shall be provided:"

(2) Replace Reporting Requirement 6.6.A.4. with the following:

"4. Maintenance (Having safety significance on systems or components designed to prevent or mitigate the consequences of nuclear accidents)

a. Nature of the maintenance, e.g., routine, emergency, preventive, or corrective.

b. The effect, if any, on the safe operation of the reactor.

c. The cause of any malfunction for which corrective maintenance was required.

d. Corrective and preventive action taken to preclude recurrence of malfunctions.

e. Time required for completion."

(3) Replace Reporting Requirement 6.6.A.5. with the following:

"5. Radioactive Liquid Waste*

a. total radioactivity (in curies) released, other than tritium and dissolved gases, and average concentration (in µCi/cc) at point of discharge;

b. total tritium and alpha radioactivity (in curies) released, and average concentration (in µCi/cc) at point of discharge;

c. total dissolved gas radioactivity (in curies) and average concentration (in µCi/cc) at point of discharge;

d. total volume (in gallons before dilution) of liquid waste discharged;

e. total volume (in gallons) of dilution water used;

f. the maximum concentration of total radioactivity other than tritium and dissolved gases released in any single batch;

g. the estimated total radioactivity (in curies) released, by nuclide, based on the results of required isotopic analyses; and

h. percent of MPC for total activity released, calculated in accordance with the instructions of Appendix B of 10 CFR Part 20 and MPC value used."

Page 6.6-2

(1) Replace Reporting Requirement 6.6.A.6. with the following:

"6. Gaseous Waste*

a. total radioactivity released excluding natural radioactivity (in curies) of:

1. noble gases
2. tritium
3. iodines
4. particulates
5. particulate alpha emitters

b. maximum hourly average release rate (for any one-hour period);

c. estimated total radioactivity (in curies) released, by nuclide, based on the results of the required isotopic analysis;

d. percent of MPC, calculated in accordance with the instructions of Appendix B of 10 CFR Part 20, and the MPC value used for:

1. noble gases

2. tritium

3. iodines

4. particulates

e. average meteorological conditions during release period including windspeeds and relative frequencies with which wind was blowing from the 16 cardinal directions, if rates of release of radioactive materials in the stack effluent averaged over any calendar quarter are such that the estimated annual average release rate will be greater than 2 percent of the stack gaseous limit specified in Section 3.6.A(1)."

(2) Replace Reporting Requirement 6.6.A.8. with the following:

"8. Environmental Monitoring

a. For each medium sampled during a six-month-period, the report of the results should include:

1. the number of sampling locations,

2. the total number of samples,

3. the quarterly average concentrations of the gross activity and specific radionuclides and total levels of external radiation for each location as well as a description of that location,

4. information should be provided on the minimum level of sensitivity of the analysis; where this has been previously reported a reference may be provided.

b. If levels of radioactive material in environmental media indicate the likelihood of public intakes in excess of 3 percent of those that could result from continuous exposure to the concentration values listed in Appendix B, Table II, Part 20, estimates of the likely resultant exposure to individuals and to population groups and assumptions upon which estimates are based shall be provided.

c. If offsite environmental concentrations are observed which are greater than normal background fluctuations, correlation of these results with effluent releases shall be provided.

(3) Delete Reporting Requirement 6.6.A.9. and replace with the following:

"9. <u>Facility Changes</u>

A summary description of safety related changes in the facility or in the procedures."

(4) Delete note at the bottom of the page and replace with the following:

"*Summarized on a monthly basis."

(5) Add a new Reporting Requirement 6.6.A.10. as follows:

"10. <u>Tests and Experiments</u>

A summary description of safety related tests and experiments performed during the reporting period including surveillance tests and their results."

Add the following Reporting Requirements to Section 6.6.

"B. The events listed below require reports within 24 hours by telephone or telegraph to Region I Compliance Office followed by a written report within 10 days to the Director, Division of Reactor Licensing, USAEC, Washington, D. C. 20545, with a copy to Region I Compliance Office. The written report, and to the extent possible the preliminary telephone or telegraph report, shall describe, analyze and evaluate safety implications, and outline the corrective actions and measures taken or planned to prevent recurrence of 1., 2., and 3. below:

1. Any significant variation of measured values of thermal, nuclear or hydraulic characteristics from a corresponding predicted value.

2. Any abnormal occurrences as specified in Section 1.15 of these specifications.

3. Incidents or conditions which resulted in a safety limit established in these Specifications being exceeded.

C. The events listed below require reports within 30 days in writing to the Director, Division of Reactor Licensing, USAEC, Washington, D. C. 20545, with a copy to Region I Compliance Office:

1. Any change in transient or accident analyses, as described in the Final Safety Analysis Report and its amendments, which involves an unreviewed safety question as defined in Section 50.59(c) of 10 CFR 50.

2. Any changes in plant operating organization which involve positions for which minimum qualifications are specified in the Technical Specifications, or in personnel assigned to these positions."

UNITED STATES ATOMIC ENERGY COMMISSION

SAFETY EVALUATION BY THE DIVISION OF REACTOR LICENSING

IN THE MATTER OF

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR POWER PLANT UNIT NO. 1

DOCKET NO. 50-219

TABLE OF CONTENTS

1.0 INTRODUCTION

On August 1, 1969, the Atomic Energy Commission issued Amendment No. 1 to Provisional Operating License (POL) No. DPR-16 to Jersey Central Power & Light Company (JCPL) which authorized operation of the Oyster Creek Nuclear Power Plant at steady state power levels up to 1600 MWt. On December 2, 1970, Amendment No. 2 was issued which authorized operation at steady state power levels up to 1690 MWt. By application dated December 30, 1970 (Amendment No. 65 to the Application for Provisional Operating License), JCPL applied for an amendment to authorize operation at steady state power levels up to 1930 MWt. Additional material was submitted in Supplement No. 1 to Amendment 65 on January 26, 1971, and in Supplement No. 2 to Amendment 65 on June 4, 1971. The Start-up Tests and the Power Tests at 1600 MWt and 1690 MWt have produced results in good agreement with the predictions and analysis presented in the applications for the operating authorizations. Plant performance during the two years since initial criticality was achieved has been satisfactory. Shakedown problems, such as the plugging of strainers on the control rod drives, seal damage in the control rod drives, and turbine control valve oscillations have occurred and have been remedied.

We have evaluated the Oyster Creek Nuclear Power Plant for operation at power levels up to 1930 MWt with the present core loading. This evaluation is based on: review of "Application for an Increase in Power Level", Amendment 65, dated December 30, 1970, Supplement No. 1 to Amendment 65 dated January 26, 1971, and Supplement No. 2 to Amendment 65 dated June 4, 1971; review of the 1-1/2 years of power operation of this plant; review of the start-up test program results; review of the results of the evaluation of the Hench-Levy correlation for setting the safety limits, and review of the results of the evaluation of the Emergency Core Cooling System (ECCS), Amendment 67, dated September 3, 1971.

We have examined the reanalyses provided by the licensee of all anticipated operational transients that might be expected to result from any single operator error or equipment malfunction. The results show that the design and performance objectives will be satisfied for the proposed operation at 1930 MWt. In addition, the design basis accidents have been reanalyzed for the higher power level using current calculational models. The calculated doses are maintained acceptably low by reducing the allowable containment leakage and by reducing the allowable iodine activity in the reactor coolant.

The AEC has adopted interim acceptance criteria for the performance of Emergency Core Cooling System (ECCS) in light-water nuclear power plants. An Appendix A to the interim policy statement describes an evaluation model acceptable to the Commission for plants incorporating a nuclear steam supply designated by the General Electric Company. On the basis of results of calculations using this model, we conclude that the Oyster Creek ECCS is adequate to cope with postulated loss-of-coolant accidents at steady state power levels up to 1930 MWt.

Several changes to the Technical Specifications are necessary to reflect the increased power rating. In addition to these changes, the Technical Specifications are being updated, notably in the area of effluent release limits.

The only physical change to the facility required by the increase in power is the addition of a fifth relief valve. This valve is not required for power levels up to 1865 MWt.

Based upon our evaluation of the facility as presented in subsequent sections, we conclude that there is reasonable assurance that the health and safety of the public will not be endangered by the operation of Oyster Creek Nuclear Power Plant at steady state power levels up to a maximum of 1930 MWt.

2.0 SITE AND ENVIRONMENT

Jersey Central Power & Light submitted, in Amendment 65, a description of the operational phase of the Environmental Monitoring Program as a revision to the FD&SAR. We reviewed the submittal and have incorporated this program in the Surveillance Section of the Technical Specifications. We have discussed with the applicant improvements to the Environmental Monitoring Program and JCPL is preparing a revised program. This revised program will involve moving some of the monitor stations and requires installation of power supplies for the continuous air monitors and collecting overlapping data from the two programs. JCPL will submit the revised program for our review and approval.

Current requirements for the monitoring and reporting of effluents are incorporated in Change No. 7 to the Technical Specifications being issued with this amendment.

3.0 FACILITY DESIGN

The following sections describe the safety-related aspects of the facility design that are affected by the power increase.

3.1 Reactor Design

The licensee's verification of the core thermal and hydraulic performance for the 1600 MWt operation was based on the in-core chamber data. In a representative case reported in the Oyster Creek Start-up Test Results, NEDE-13109, the calculated MCHFR was 2.94 at a reactor power of 1585 MWt and a core flow of 61×10^6 lb/hr. Evaluated at 120% overpower, the MCHFR for these conditions was 2.33. These results showed that the reactor could be operated well within the criterion of a MCHFR of 1.5 at 120% power. The critical heat fluxes used for the evaluation of these data were based on the older Janssen-Levy correlation (APED-3892).

For the proposed operation at 1930 MWt, the licensee calculates the critical heat flux based on the more recent Hench-Levy correlation (APED-5286). We found the use of this correlation to be acceptable during our operating license review of Dresden Unit No. 2 and the correlation has been the basis for safety limit determinations on subsequent BWR's. Table I compares the thermal and hydraulic data for Oyster Creek at the proposed power rating of 1930 MWt with the data for Nine Mile Point and Dresden Unit No. 2 at their licensed ratings.

TABLE I

Comparison of OC-1 with NMP and Dresden-2 Thermal and Hydraulic Design Parameters

	Oyster Creek	Nine Mile Point	Dresden-2
Power Level, MWt	1930	1850	2527
Number of Fuel Bundles	560	532	724
Av. Power per Bundle, MWt	3.45	3.5	3.5
MCHFR	$\geq$1.9 at above power	$\gtrsim$1.9 at above power	$\gtrsim$1.9 at above power
Av. Power Density, kW/liter	41	41	41
Max. linear heat generation rate, kW/ft	17.2	17.5	17.5
Peak Heat Flux, Btu/hr-ft^2	3.93×10^5	4.00×10^5	4.05×10^5
Av. Heat Flux, Btu/hr-ft^2	1.30×10^5	1.31×10^5	1.32 10^5
Max. Center Fuel Temp., ^oF	4175	4250	4530
Peaking Factors			
Local	1.30	1.30	1.30
Axial	1.57	1.57	1.57
Radial	1.49	1.50	1.50
Gross (Product)	3.03	3.06	3.06
Steam Dome Pressure, psig	1020	1030	1000
Core Flow Rate, 10^6 lbs/hr	61.0	67.5	98
Steam Flow Rate, 10^6 lbs/hr	7.25	7.29	9.95
Core Inlet Enthalpy, Btu/lb	517.3	526	522
Core Av. Void Fraction, %	29.2	31.0	29.9
Hot Channel Coolant Flow 10^6 lbs/hr	0.0998	0.112	0.117

This table shows that all of the thermal and hydraulic design parameters except fuel bundle flow rate are more conservative for Oyster Creek than for the other two reactors. The reduced flow rate is compensated by a substantial increase in the core inlet subcooling and by slight reductions in the average heat flux and the radial peaking factor. The core thermal and hydraulic Safety Limits are based on a feedwater temperature of 334°F which is the maximum feedwater temperature attainable with the feedwater heater system design. For any lower feedwater temperature, subcooling is increased and an increased margin of safety results. The expected feedwater temperature, obtained by the calculated plant heat balance at 1930 MWt, is 315°F. Steam to the shell side of the final feedwater heater is turbine extraction steam at 90 psig with a saturation temperature of 330°F. The pressure on the shell side is limited by a relief valve set at 100 psig, limiting the temperature available for heating to 338°F. We have concluded that this satisfactorily limits the feedwater temperature, and thereby assures that the subcooling will be at least as great as that assumed for the safety limit curves.

We have compared the transient analyses revised for 1930 MWt with the analyses presented for the 1600 MWt operation and will comparable analyses for Nine Mile Point. All limiting design and performance criteria remain unchanged from the 1600 MWt analyses and are met for all transients analyzed. In order to meet the criterion of not lifting safety valves for the turbine trip concurrent with failure of by-pass valves, a fifth relief valve is required. A turbine trip scram and load rejection scram were incorporated for the 1690 MWt operation and, therefore, are included in the analyses for 1930 MWt.

The reactivity control characteristics and core nuclear characteristics were verified for the 1600 MWt operation and are not affected significantly by the proposed power increase. The shutdown margin and maximum control rod notch worth were demonstrated to comply with the Technical Specifications requirements during the start-up testing program.

We have concluded that the Oyster Creek reactor core can be operated safely at steady state power levels up to 1930 MWt on the basis of: the satisfactory core performance results from start-up testing and operation for more than a year; previously accepted application of the Hench-Levy correlation for establishing Safety Limits for boiling water reactors; and the acceptable consequences of the transients and accidents, all of which have been reanalyzed for this application.

3.2 Reactor Coolant System

The effects of the power increase on the reactor vessel are a slight increase in operating pressure (from 1000 psig to 1020 psig) and a 20% increase in neutron exposure of the vessel wall. The pressure increase, considering steady state aspects only, results in a very small decrease in the margin to the design pressure rating of the vessel. The 1250 psig vessel rating continues to supply a satisfactory margin for operation. In considering the effects of transients with the increased initial value of pressure and steam flow, some modifications were incorporated to maintain the same margin of safety. The operating margin was reduced from 60 psig to 50 psig since the pressure scram set point was increased by only 10 psig to 1070 psig. The allowance for transients was reduced by 10 psig but was compensated for by the addition of a fifth relief valve. This results in maintaining approximately the same peak pressure for the worst case pressure transient, turbine trip with failure of by-pass.

In Supplement No. 2, JCPL provided an analysis of the turbine trip with failure of by-pass with four relief valves. This showed that at a power level of 1865 MWt, the margin between the peak pressure and the lowest set safety valve was maintained. They also reported that the fifth relief valve would not be available from the supplier until September. The maximum steady state power level will be limited by the Technical Specifications to 1865 MWt when only four relief valves are operable.

The vessel wall exposure to neutrons with energies greater than 1 MeV at the end of the 40-year design life was calculated to be 1×10^{18} nvt. This was based on a reactor full power level of 1950 MWt and results in a "worst case" NDT temperature shift to 100^{o}F. The vessel wall exposure calculation is to be verified or adjusted as necessary, based on the results of the surveillance of the neutron flux monitors which will be removed at the first refueling outage. (Technical Specifications 4.3.A.)

The applicant's reanalyses of transients associated with primary coolant system were reviewed. These transients were as follows:

a. Relief Valve Sizing - Turbine Trip with Failure of Bypass

b. Inadvertent Opening of Relief Valve

c. Safety Valve Sizing - Turbine Trip with Failure of By-pass Failure of Scram, Failure of Relief-Valves, and Failure of Isolation Condenser

d. Recirculation Pump Trips

e. Recirculation Pump Stall

f. Recirculation Pumps Controller Malfunctions

g. Main Steam Isolation Valve Closure

We have concluded that the results meet the performance criteria of the system and that the appropriate Safety Limits will not be violated.

3.3 Containment

The containment design is based on a power level of 1860 MWt and a reactor pressure of 1000 psig. The increase to 1930 MWt and 1020 psig does not affect significantly the containment design or performance requirements. The Bodega Bay Tests, which provided the data for predicting the containment pressure rise, were performed at 1250 psig test vessel pressure and, therefore, still provide sufficient conservatism.

The peak accident pressures were calculated to be 38 psig in the drywell and 25 psig in the absorption chamber. The design pressures are 62 psig for the drywell and 35 psig for the absorption chamber.

The allowable leakage of 1 wt %/day is verified by periodic surveillance testing which limits the measured leakage to 0.75 wt %/day at 35 psig. This margin is specified to allow for degradation between tests. Two containment tests have been performed to-date. The first test in 1969 showed a leakage rate of 0.48 wt %/day at 35 psig and the second test in 1970 showed a leakage rate of 0.21 wt %/day extrapolated to 35 psig from a test pressure of 20 psig.

We conclude that the Oyster Creek containment is satisfactory for reactor operation at 1930 MWt.

3.4 Emergency Core Cooling System (ECCS)

The AEC has adopted interim acceptance criteria for the performance of Emergency Core Cooling System (ECCS) in light-water nuclear power plants. An Appendix A to the interim policy statement describes an evaluation model acceptable to the Commission for plants incorporating a nuclear steam supply designed by the General Electric Company. The Oyster Creek ECCS was evaluated in accordance with the evaluation model. The results of the calculations indicate that the peak clad temperature does not exceed 2237°F. (Amendment 67)

We conclude that the evaluation model provides a conservative basis for assessing the acceptability of ECCS performance for Oyster Creek. On the basis of results of calculations using this model, we conclude that the Oyster Creek ECCS would:

(a) limit the peak clad temperature to less than 2300°F, which is well below the clad melting temperature,

(b) limit the fuel clad water reaction to less than one percent of the total clad mass,

(c) terminate the temperature transient before the core geometry necessary for core cooling is lost and before the cladding is so embrittled as to fail upon quenching, and

(d) reduce the core temperature and remove core decay heat for an extended period of time for the entire spectrum of postulated break sizes including the double-ended break of a recirculation line.

We, therefore, conclude that the Oyster Creek ECCS is adequate to cope with postulated loss-of-coolant accidents at steady state power levels up to a maximum of 1930 MWt.

3.5 Control and Instrumentation

The anticipatory scrams from turbine trip and generator load rejection were incorporated for the power increase to 1690 MWt and were reviewed and found acceptable at that time. The turbine trip and generator load rejection transients were reanalyzed at 1930 MWt with these anticipatory scrams included. The anticipatory scrams limited the pressure peak to below the setting of the relief valves and these results were found to be acceptable.

The nuclear instrumentation continues to supply satisfactory power monitoring over the full range of reactor operation. The recalibration of the LPRM's and IRM's to cover the highest power retains satisfactory overlap in the ranges of transition between instrument channels.

3.6 Electrical Systems

The applicant has reanalyzed the loss of power accident at 1930 MWt. The loss of all auxiliary power causes loss of condenser cooling water, trip of feedwater pumps, trip of recirculation pumps, and turbine trip at time zero. Two changes in the analysis from that performed at 1600 MWt are the incorporation of the turbine trip scram and the assumption of 1.5 seconds of steam flow through the by-pass valves prior to isolation of the main condenser from loss of vacuum. This latter assumption is based on design requirements (6 seconds of full by-pass flow and decreasing steam flow over the next 24 seconds for a total of 12,000 lbs of steam) and on results of the Start-up Test Program. Since the 1.5 second steam by-pass assumed for this analysis corresponds to only 1200 lbs of steam, this assumption is quite conservative.

The results of this transient are acceptable and we conclude that the system meets its performance requirements.

3.7 Auxiliary Systems

The auxiliary coolant systems have sufficient capacity to absorb the added heat loads imposed by the higher power, higher temperature operation. The evaluation was made with the benefit of the 1-1/2 years of operating experience. We have concluded that the Auxiliary Systems are satisfactory for the higher power operation.

3.8 Turbine Generator and Condensate

The design basis, system description, and components are unchanged for the proposed operation at the higher power level. The applicant's engineering review of the design, along with an analysis of the operational experience at the 1600 and 1690 MWt power levels, concludes that sufficient margin is available to support operation at the full design plant rating of 1930 MWt and 670 MWe. The design evaluation included analyses of several transients which are associated with the performance of malfunctioning of components in these systems. They include loss of electrical load, turbine trips, loss of vacuum, inadvertent opening of a turbine by-pass valve, loss of feedwater, and excess feedwater flow. These analyses consider the effects of incorporating the turbine trip scram and the load rejection scram. We have concluded that the results meet the performance criteria of these systems and that reactor safety limits are not violated.

4.0 ACCIDENT ANALYSES

All of the accidents evaluated previously for Oyster Creek Unit No. 1 at 1860 MWt have been reanalyzed at 1930 MWt using current calculational methods, assumptions and criteria. Most of these methods and assumptions have been published as "Safety Guides for Water Cooled Nuclear Power Plants".

To determine compliance with the guidelines established in 10 CFR Part 100 for this facility, we have evaluated the accident involving loss of coolant inside the drywell, the refueling accident, the steamline break accident outside the drywell, and the control-rod-drop accident.

The results of our analyses for these accidents are summarized in the following subsections and the doses that we have calculated are summarized in Table 11.0. We have assumed only 90 percent efficiency for halogen removal by the standby gas treatment system as compared with the 99 percent which the applicant believes will be achieved. For the refueling and loss-of-coolant accident, we assumed the release of activity from the 110 meter stack. For the postulated break of a steamline outside the drywell, we assumed the release of activity from the top of the turbine building (30 meters). Ground level release of the activity was assumed for the control-rod-drop accident.

The meteorology used in our dose calculations for the loss-of-coolant accident and the refueling accident is that for a plant located less than two miles from large bodies of water as given in Safety Guide No. 3.

The meteorology used in our calculation of the consequences for the steamline break accident outside the drywell is that given in Safety Guide No. 5. The meteorology used for the control-rod-drop accident is that given in Safety Guide No. 3 for a ground level release with a building wake effect calculated to be a factor of approximately 3 at the exclusion radius.

As can be seen from the data in the following table, the doses resulting from accidents are well below the 10 CFR Part 100 guideline values.

Table 11.0

Summary of Calculated Doses

	Exclusion Area* - 2 hour dose		LPZ** - Course of Accident	
Accident	Thyroid (Rem)	Whole Body (Rem)	Thyroid (Rem)	Whole Body (Rem)
1. Loss of Coolant	139	10	104	4.4
2. Refueling	109	6	26	1.5
3. Control Rod Drop	9	0.5	4	0.2
4. Steamline Break	29	0.2	2.2	0.01

*Exclusion area distance varies from 400 m for Accidents 3 and 4 to 600 m for Accidents 1 and 2 to obtain peak dose.

**Low population zone distance is 3200 meters (2 miles).

4.1 Loss-of-Coolant Accident

In calculating the consequences of the loss-of-coolant accident, we have used the method and assumptions given in Safety Guide No. 3. A primary containment leak rate of one percent of the containment volume per day was assumed to remain constant for the duration of the accident. The Technical Specifications require a containment testing program which establishes the expected primary containment leakage under loss-of-coolant accident conditions. Although the closest site boundary is 400 meters in the NNE sector, the highest dose is calculated at 600 meters in the NW sector. This is caused by the 4-hour fumigation condition from the bay, whereas the fumigation condition from land, which contributes to the dose in the NNE sector is assumed to exist for only 1/2 hour. These time periods for possible fumigation conditions over water and land are given in Safety Guide No. 3. The whole body doses include gamma and beta doses as discussed in Safety Guide No. 3.

4.2 Control-Rod-Drop Accident

For the control-rod-drop accident, we assumed that the highest worth rod (2.5% Δ k/k) fell out of the core. This reactivity addition would produce an excursion with a minimum reactor period of 8.5 milliseconds and total energy generation of 4000 MW-sec, resulting in a perforation of 330 fuel rods (enthalpies greater than 170 cal/gm). The Technical Specifications will limit rod worth to 2.5% Δ k/k to establish the above accident limit.

We have evaluated the consequences of the control-rod-drop accident assuming that 330 fuel rods would fail, releasing 100 percent of the noble gases and 50 percent of the halogens from the affected rods to the primary system. The fission product inventory was based upon 330 average fuel rods having a peaking factor of 1.5 with a decay period of 30 minutes. The control-rod-drop accident would result in the worst consequences if it occurred under hot standby conditions, therefore, a decay period of 30 minutes from full power conditions was assumed.

Of the halogens released from the affected rods, 90 percent is assumed to be retained in the primary system and 50 percent of the remaining halogens is assumed to be removed by plate-out. All of the noble gases and 2.5 percent of the halogens would be released from the main condenser which has been automatically isolated on signal of high radiation in the main steam lines. This release is assumed at ground level at a rate of 0.5 volume percent per day and includes a building wake effect. The whole body doses include gamma and beta doses as discussed in Safety Guide No. 3. The Technical Specifications will require the above isolation to occur as stated to limit the atmospheric release from the control-rod-drop accident.

4.3 Refueling Accident

The refueling accident is assumed to occur 24 hours after shutdown. During fuel handling operation, a fuel assembly is assumed to drop with sufficient force to perforate 445 fuel rods contained in 29 fuel assemblies. These 29 fuel assemblies are assumed to be in the highest radial power region of the core having a peaking factor of 1.5 to determine fission product inventory. The assumed release of fission products is 20 percent of the noble gases and 10 percent of the halogens from the damaged rods into the reactor building. Of the halogens released from the affected rods, 90 percent is assumed to be retained in the refueling water. All of the noble gases (20 percent) and one percent of the halogens within the building are assumed to be discharged from the stack through the standby gas treatment system, with an iodine removal efficiency of 90 percent, over a 2-hour period.

4.4 Steamline-Break Accident Outside Drywell

The break of a main steamline outside of both the drywell and the reactor building represents a potential escape route from reactor coolant from the vessel to the atmosphere without passage through the primary containment or the reactor building. We have assumed an isolation valve closure time of 10 seconds (design limit) with the valve closure time terminating the accident. The method and assumptions given in Safety Guide No. 5 were used to evaluate the consequences of this accident. The primary coolant activity limit on total iodine to be given in the Technical Specifications will limit the potential thyroid dose to less than 30 Rem at the nearest site boundary. The primary coolant iodine activity limit to meet this criterion is 8 μCi/gm. The Technical Specifications will also require a minimum acceptable closure time of less than 10 seconds for testing the main steamline isolation valves to limit the total release from a steamline break outside the drywell to that assumed in our accident analysis.

5.0 CONDUCT OF OPERATIONS

Oyster Creek operations at 1600 MWt and 1690 MWt have been satisfactory. The initial testing program for 1600 MWt power operation and the testing program for the power increase to 1690 MWt were completed with satisfactory results. The acceptance criteria were met and predictions of operating characteristics were confirmed.

There have been 45 scrams during plant operation from initial criticality until the end of 1970. The protection system has always performed its intended function and there have been no unsafe failures of the system.

5.1 Effluent Releases

The licensee reports that all releases of radioactive effluents were well below the limits of 10 CFR 20. The values of average concentrations reported by the licensee for 1970 compare with the appropriate limit as follows:

Unidentified Liquid Activity	$<$ 20% of limit
Tritium	$<$ 0.01% of limit
Gaseous halogens and particulates	$<$ 0.5% of limit
Activation and noble gases	$<$ 1.5% of limit

Recent reevaluation of counting methods and calibration techniques used at Oyster Creek resulted in the detection of errors in the reported values for the liquid effluent activity. Independent checks of the Oyster Creek records by the Compliance Division as well as independent analyses of samples obtained from Oyster Creek waste tanks indicate that the liquid activity released may be about four times that reported by the licensee. This results in a release of about 80 percent of 10 CFR 20 limits on an unidentified basis for 1970. Identification of the radionuclides being discharged and applying the appropriate limits for those nuclides would have resulted in a lower value. For example, if the isotopic analysis performed on a waste tank sample in December were assumed to be typical for all batches released in 1970, the liquid activity release would be approximately six percent of 10 CFR 20 limits.

The licensee is presently determining a correction factor which will be used to recalculate the liquid activity released to-date. He has also changed his method of measuring the sample activity and is concentrating on making maximum use of his Radwaste System in order to reclaim as much water as possible for recycling into the plant and, thereby, minimize liquid of activity releases.

Change No. 7 to the Technical Specifications, being issued with the license amendment that authorizes operation at 1930 MWt, has incorporated more detailed monitoring and reporting requirements for the plant effluents. These requirements include more frequent checks on the calibration of monitoring equipment and more frequent identification of isotopes in the effluent releases.

5.2 1930 MWt Power Test Program

Following approval of the requested power increase, the applicant will conduct a test program to demonstrate acceptable performance at 1930 MWt. These tests will include all of those tests of the original start-up test program which are power related. Prior to raising the power level, the applicant plans to obtain a set of base point data at 1600 MWt so that any changes since the original start-up tests would not be attributed to the change in power level. The same acceptance criteria apply to these tests as in the original test program. We have concluded that the proposed tests will provide the information necessary to demonstrate the adequacy of the Oyster Creek Plant to operate at the increased power level of 1930 MWt.

6.0 TECHNICAL SPECIFICATIONS

Several changes to the Technical Specifications are necessary because of the power increase. These changes involve references to power level and reactor pressure. They include the changes in the thermal and hydraulic safety limits and corresponding limiting safety system settings. In addition to the changes directly related to the power increase, the Technical Specifications are being updated in some areas by incorporation of the current requirements regarding effluent releases, testing of instrument line flow check valves, Environmental Monitoring Program, reactivity anomalies and reporting requirements.

7.0 REPORT OF ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The ACRS completed its review of the application for an increase in power level for Oyster Creek during its 134th meeting held June 10-12, 1971. A copy of the ACRS letter, dated June 18, 1971, is attached as Appendix A.

In its letter, the ACRS identified certain items that require further action by the applicant. Although a resolution of these items is not necessary prior to operation at the increased power level, the applicant intends to pursue a resolution of these items by timely submittal of the required information for our review.

These items consist of the following:

(1) An analysis of the buildup and methods of control of combustible gases which might follow in the unlikely event of a loss-of-coolant accident shall be submitted. The submittal will describe the calculational models and assumptions for the gas buildup and for the determination of any doses resulting from the method of

control, the proposed approach to control of the combustible gases, summary procedures, and the necessary hardware. We will evaluate this submittal on the basis of Safety Guide No. 7.

(2) The corrective measures and appropriate modifications necessary to maintain the integrity of the fuel pool in the event that the fuel cask is dropped into the pool shall be submitted. The applicant has stated that a fuel cask will not be handled above the fuel pool until the corrective measures and modifications have been implemented.

(3) The applicant has developed improved plans for in-service inspection of piping inside and outside containment and is studying possible methods of conducting reactor pressure vessel integrity surveillance. These plans shall be submitted for our review. We understand that the new in-service inspection plans are to be implemented on a trial basis in order to arrive at an acceptable program to be included in the Technical Specifications after the fourth year of operation in accordance with Note 3 to Table 4.3.1 of the Technical Specifications.

(4) A report on the performance of the atmospheric radioactivity monitoring system for leak detection in the containment drywell shall be submitted. This report shall include an evaluation of the leak detection sensitivity and the proposed method of use of the system as regards leakage detection and the the effect of such leakage detection on operation of the plant. The applicant has been testing and evaluating this system for about six months. Their original plans indicated that approximately one year would be required for proper evaluation.

(5) An analysis of the effect of an instrument line failure on the integrity of the secondary containment building and of the resulting offsite doses shall be submitted.

(6) A report on design features that make tolerable the consequences of failure to scram during anticipated transients shall be submitted.

The ACRS concluded in its letter that if due regard is given to the items mentioned above and in its previous reports, there is reasonable assurance that the Oyster Creek Nuclear Power Plant can be operated at a power level as high as 1930 MWt without undue risk to the health and safety of the public.

8.0 CONCLUSIONS

Based upon our review of the application, and of relevant information regarding facility operation to date as discussed in this evaluation, we have concluded that there is reasonable assurance that the Oyster Creek Nuclear Station can be operated at steady state power levels up to a maximum of 1930 MWt without endangering the health and safety of the public.

Donald J. Skovholt

Donald J. Skovholt
Assistant Director for
Reactor Operations
Division of Reactor Licensing

Date:

UNITED STATES ATOMIC ENERGY COMMISSION

DOCKET NO. 50-219

JERSEY CENTRAL POWER & LIGHT COMPANY

AMENDMENT TO PROVISIONAL OPERATING LICENSE

No request for hearing or petition to intervene having been filed following publication of a notice of proposed action in the Federal Register on June 12, 1971, at 36 F. R. 11473, the Atomic Energy Commission ("the Commission") has issued Amendment No. 3 to Provisional Operating License No. DPR-16. The amendment authorizes Jersey Central Power & Light Company to operate the Oyster Creek Nuclear Power Plant Unit No. 1 ("the facility") at steady state power levels up to a maximum of 1930 megawatts (thermal). The amendment restates the license in its entirety to delete the record and reporting requirements that are now incorporated in the Technical Specifications as Change No. 7 and appended to Amendment No. 3.

The facility has been inspected by a representative of the Commission who has verified that the modification of the facility (involving the installation of a fifth relief valve and its controls) has been satisfactorily accomplished by Jersey Central.

The Commission has found that the application for the amendment complies with the requirements of the Atomic Energy Act of 1954, as amended ("the Act"), and the Commission's regulations published in 10 CFR Chapter I. The Commision has made the findings required by the Act and the Commission's regulations which are set forth in the amendment and has concluded that the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

For further details with respect to this amendment, see (1) the application for license amendment dated December 31, 1970, and supplements thereto dated January 26, 1971 and June 4, 1971, (2) the amendment to Provisional Operating License No. DPR-16, (3) the report by the Advisory Committee on Reactor Safeguards dated June 18, 1971, and (4) the Safety Evaluation prepared by the Division of Reactor Licensing, which are available for public inspection at the Commission's Public Document Room at 1717 H Street, N. W., Washington, D. C. Copies of items (2), (3) and (4) may be obtained upon request sent to the U. S. Atomic Energy Commission, Washington, D. C. 20545, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this 5th day of November.

FOR THE ATOMIC ENERGY COMMISSION

Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing