Docket No. 50-219

SEP 2 7 1974

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Jersey Central Power & Light Company
ATTN: Mr. I. R. Finfrock, Jr.
Vice President - Generation
Madison Avenue at Punch Bowl Road
Morristown, New Jersey 07960

Gentlemen:

The Commission has issued the enclosed Amendment No. 6 to Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station Unit No. 1. This amendment includes Change No. 22 to the Technical Specifications, Appendix A, and is in response to your Technical Specification Change Request No. 24 dated May 28, 1974.

This amendment provides for modifications to the limiting conditions for operation, surveillance requirements, and bases for the suppression chamber to drywell vacuum breaker system.

Copies of the related Safety Evaluation and Federal Register Notice are also enclosed.

Sincerely,

Original Signed

Karl R. Goller, Assistant Director
for Operating Reactors
Directorate of Licensing

Enclosures:
1. Amendment No. 6
2. Safety Evaluation
3. Federal Register Notice

cc: See next page

bcc: HJMcAlduff
JRBuchanan
TBAbernathy
ARosenthal
NHGoodrich

SEE PREVIOUS CONCURRENCE CHAIN

word change in ltr per RPurple 9/27

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SEP 27 1974

Docket No. 50-219

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Vice President
Morristown, New Jersey 07960

Gentlemen:

The Commission has issued the enclosed Amendment No. 6 to Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station Unit No. 1. This amendment includes Change No. 22 to the Technical Specifications, Appendix A, and is in response to your Technical Specification Change Request No. 24 dated May 28, 1974.

This amendment provides for ~~system~~ modifications to the limiting conditions for operation, surveillance requirements, and bases for the suppression chamber to drywell vacuum breaker system.

Copies of the related Safety Evaluation and Federal Register Notice are also enclosed.

Sincerely,

Original Signed

Karl R. Goller, Assistant Director
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bcc: HJMcAlduff
JRBuchanan
TBAbernathy
ARosenthal
NHGoodrich

concur with change on p. 3 of SE
Change per OGC 9/25

SEE PREVIOUS YELLOW FOR CONCURRENCE CHAIN

OFFICE	ORB#3	ORB#3	ORB#3	OGC	L:AD/ORs	
SURNAME	JIRiesland:kmf	SATeets	GLear GL		KRGoller KRG	
DATE	9/26/74	9/26/74	9/26/74	9/27/74	9/27/74	BN

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

cc: w/enclosures

Jersey Central Power & Light Company
ATTN: Mr. Thomas M. Crimmins, Jr.
Safety and Licensing Manager
GPU Service Corporation
260 Cherry Hill Road
Parsippany, New Jersey 07054

George F. Trowbridge, Esquire
Shaw, Pittman, Potts & Trowbridge
910 - 17th Street, N. W.
Washington, D. C. 20016

The Honorable W. M. Mason
Mayor, Lacey Township
P. O. Box 475
Forded River, New Jersey 08731

Ocean County Library
15 Hooper Avenue
Toms River, New Jersey 08753

Honorable William F. Hyland
Attorney General
State of New Jersey
State House Annex
Trenton, New Jersey 08601

Docket No. 50-219

Jersey Central Power & Light Company
ATTN: Mr. I. R. Finfrock, Jr.
Vice President
Morristown, New Jersey 07960

Gentlemen:

The Commission has issued the enclosed Amendment No. 6 to Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station Unit No. 1. This amendment includes Change No. 22 to the Technical Specifications, Appendix A, and is in response to your Technical Specification Change Request Nos. 23 and 24 dated April 22, 1974 and May 28, 1974, respectively.

This amendment provides for system modifications to the limiting conditions for operation, surveillance requirements, and bases for the suppression chamber to drywell vacuum breaker system.

Copies of the related Safety Evaluation and Federal Register Notice are also enclosed.

Action on your Technical Specification Change Request No. 23, dated April 22, 1974, for a proposed change to the Technical Specifications to permit operation with only six operable suppression chamber to drywell vacuum breakers has been suspended because of inadequate justification.

Sincerely,

Karl R. Goller, Assistant Director
for Operating Reactors
Directorate of Licensing

Enclosures:
1. Amendment No. 6
2. Safety Evaluation
3. Federal Register Notice

OFFICE ➤	ORB#3	ORB#3	OGC	L:AD/OR		
SURNAME ➤	SATeets Riesland:km	GLear G.L.		KGoller		
DATE ➤	9/16/74	9/17/74	9/ /74	9/ /74		

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Docket No. 50-219

SEP 27 1974

Amdt. # 6 to DPR-16
& Chg. # 22

Jersey Central Power & Light Company
ATTN: Mr. I. R. Finfrock, Jr.
Vice President - Generation
Madison Avenue at Punch Bowl Road
Morristown, New Jersey 07960

Gentlemen:

The Commission has issued the enclosed Amendment No. 6 to Provisional Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station Unit No. 1. This amendment includes Change No. 22 to the Technical Specifications, Appendix A, and is in response to your Technical Specification Change Request No. 24 dated May 28, 1974.

This amendment provides for modifications to the limiting conditions for operation, surveillance requirements, and bases for the suppression chamber to drywell vacuum breaker system.

Copies of the related Safety Evaluation and Federal Register Notice are also enclosed.

Sincerely,

for Karl R. Goller, Assistant Director
for Operating Reactors
Directorate of Licensing

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Honorable William F. Hyland
Attorney General
State of New Jersey
State House Annex
Trenton, New Jersey 08601

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

JERSEY CENTRAL POWER & LIGHT COMPANY

DOCKET NO. 50-219

OYSTER CREEK NUCLEAR GENERATING STATION UNIT NO. 1

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 6
License No. DPR-16

1. The Atomic Energy Commission (the Commission) has found that:

   A. The application for amendment by Jersey Central Power & Light Company (the licensee) dated May 28, 1974, 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. Prior public notice of this amendment is not required since the amendment does not involve a significant hazards consideration.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 3.B of Provisional Operating License No. DPR-16 is hereby amended to read as follows:

   "(B) Technical Specifications

   The Technical Specifications contained in Appendix A as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 22."

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

FOR THE ATOMIC ENERGY COMMISSION

Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Directorate of Licensing

Attachment:
Change No. 22 to the
Technical Specifications

Date of Issuance: SEP 27 1974

ATTACHMENT TO LICENSE AMENDMENT NO. 6

CHANGE NO. 22 TO THE TECHNICAL SPECIFICATIONS

PROVISIONAL OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

Replace pages 3.5-1 through 3.5-7 of the Technical Specifications, Appendix A, with the attached pages 3.5-1 through 3.5-7

Replace page 4.5-5 of the Technical Specifications, Appendix A with the attached pages 4.5-5 and 4.5-5a; similarly replace page 4.5-9 with the attached pages 4.5-9 and 4.5-9a.

3.5 CONTAINMENT

Applicability: Applies to the operating status of the primary and secondary containment systems.

Objective: To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment

1. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MWt.

2. The absorption chamber water volume and temperature shall be maintained within the following limits:

   a. Maximum water temperature 100°F

   b. Minimum water volume 82,000 ft^3

   c. Maximum water volume 92,000 ft^3

3. Reactor Building to Suppression Chamber Vacuum Breaker System

   a. Except as specified in Specification 3.5.A.3.b below, two reactor building to suppression chamber vacuum breakers in each line shall be operable at all times when primary containment integrity is required. The set point of the differential pressure instrumentation which actuates the air-operated vacuum breakers shall not exceed 0.5 psid. The vacuum breakers shall move from closed to fully open when subjected to a force equivalent to not greater than 0.5 psid acting on the vacuum breaker disc.

   b. From the time that one of the reactor building to suppression chamber vacuum breakers is made or found to be inoperable, the vacuum breaker shall be locked closed and reactor operation is permissible only during the succeeding seven days unless such vacuum breaker is made operable sooner, provided that the procedure does not violate primary containment integrity.

22

Change No. 22
Date SEP 27 1974

c. If the limits of Specification 3.5.A.3.a are exceeded, reactor shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

4. Pressure Suppression Chamber - Drywell Vacuum Breakers

a. When primary containment is required, all suppression chamber - drywell vacuum breakers shall be operable except during testing and as stated in Specification 3.5.A.4.b and c, below. Suppression chamber - drywell vacuum breakers shall be considered operable if:

(1) The valve is demonstrated to open from closed to fully open with the applied force at all valve positions not exceeding that equivalent to 0.5 psi acting on the suppression chamber face of the valve disk.

22

(2) The valve disk will close by gravity to within not greater than 0.10 inch of any point on the seal surface of the disk when released after being opened by remote or manual means.

(3) The position alarm system will annunciate in the control room if the valve is open more than 0.10 inch at any point along the seal surface of the disk.

b. Two of the fourteen suppression chamber - drywell vacuum breakers may be inoperable provided that they are secured in the closed position.

c. One position alarm circuit for each operable vacuum breaker may be inoperable for up to 15 days provided that each operable suppression chamber - drywell vacuum breaker with one defective alarm circuit is physically verified to be closed immediately and daily during this period.

5. After completion of the startup test program and demonstration of plant electrical output, the primary containment atmosphere shall be reduced to less than 5.0% O_2 with nitrogen gas within 24 hours after the reactor mode selector switch is placed in the run mode. Primary containment deinerting may commence 24 hours prior to a scheduled shutdown.

Change No. 22
Date SEP 27 1974

22 6. If the above specifications cannot be met, reactor shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours.

B. Secondary Containment

1. Secondary containment integrity shall be maintained at all times unless all of the following conditions are met.

   a. The reactor is subcritical and Specification 3.2.A is met.

   b. The reactor is in the cold shutdown condition.

   c. The reactor vessel head or the drywell head are in place.

   d. No work is being performed on the reactor or its connected systems in the reactor building.

   e. No operations are being performed in, above, or around the spent fuel storage pool that could cause release of radioactive materials.

2. The standby gas treatment system shall be operable at all times when secondary containment integrity is required except as specified by Specification 3.5.B.3.

3. One standby gas treatment filter circuit may be inoperable for 7 days, when standby gas treatment system operability is required, except during reactor startup, provided the remaining filter circuit is proved operable daily.

22 4. If Specifications 3.5.B.2 and 3.5.B.3 are not met, reactor shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours and the conditions of Specification 3.5.B.1 shall be met.

Bases: Specifications are placed on the operating status of the containment systems to assure their availability to control the release of any radioactive material from irradiated fuel in the event of an accident condition. The primary containment system (1) provides a barrier against uncontrolled release of fission products to the environs in the event of a break in the reactor coolant systems.

Change No. 22
Date SEP 27 1974

Whenever the reactor coolant water temperature is above 212°F, failure of the reactor coolant system would cause rapid expulsion of the coolant from the reactor with an associated pressure rise in the primary containment. Primary containment is required, therefore, to contain the thermal energy of the expelled coolant and fission products which could be released from any fuel failures resulting from the accident. If the reactor coolant is not above 212°F, there would be no pressure rise in the containment. In addition, the coolant cannot be expelled at a rate which could cause fuel failure to occur before the core spray system restores cooling to the core. Primary containment is not needed while performing low power physics tests since the rod worth minimizer would limit the worst case rod drop accident to 1.5%Δk. This amount of reactivity addition is insufficient to cause fuel damage.

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 the accident analyses with respect to free air volume in the (8) absorption chamber. For example, the containment capability with a maximum water volume of 92,000 ft^3 is reduced by not more than 3.5% metal-water reaction below the capability with 82,000 ft^3.

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 external design pressure; the external design pressure of the drywell is 2 psi; the external design pressure of the suppression chamber is 1 psi (FDSAR Amendment 15, Section II).

22

Change No. 22
Date SEP 27 1974

The capacity of the fourteen 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 2 psi. They are sized on the basis of the Bodega Bay pressure suppression tests.[9][10] In Amendment 15 of the Oyster Creek FDSAR, Section II, the area of 2920 sq. in. is stated as the minimum area for flow of non-condensible gases from the suppression chamber to the drywell. To achieve this requirement, at least 12 of the 14 vacuum breaker valves (18" diameter) must be operable.

Each suppression chamber drywell vacuum breaker is fitted with a redundant pair of limit switches to provide fail safe signals to panel mounted indicators in the Reactor Building and alarms in the Control Room when the disks are open more than 0.1" at any point along the seal surface of the disk. These switches are capable of transmitting the disk closed-to-open signal with 0.01" movement of the switch plunger. Continued reactor operation with failed components is justified because of the redundancy of components and circuits and, most importantly, the accessibility of the valve lever arm and position reference external to the valve. The fail-safe feature of the alarm circuits assures operator attention if a line fault occurs. 22

Conservative estimates of the hydrogen produced, consistent with the core cooling system provided, show that the hydrogen air mixture resulting from a loss-of-coolant accident is considerably below the flammability limit and hence it cannot burn, and inerting would not be needed. However, inerting of the primary containment was included in the proposed design and operation. The 5% oxygen limit is the oxygen concentration limit stated by the American Gas Association for hydrogen-oxygen mixtures below which combustion will not occur.[4]

To preclude the possibility of starting up the reactor and operating a long period of time with a significant leak in the primary system, leak checks must be made when the system is at or near rated temperature and pressure. It has been shown[9][10] that an acceptable margin with respect to flammability exists without containment inerting. Inerting the primary containment provides additional margin to that already considered acceptable. Therefore, permitting access to the drywell for the purpose of leak checking would

not reduce the margin of safety below that considered adequate and is judged prudent in terms of the added plant safety offered by the opportunity for leak inspection. The 24-hour time to provide inerting is judged to be a reasonable time to perform the operation and establish the required O_2 limit.

Secondary containment[5] is designed to minimize any ground level release of radioactive materials which might result from a serious accident. The reactor building provides secondary containment during reactor operation when the drywell is sealed and in service and provides primary containment when the reactor is shut down and the drywell is open, as during refueling. Because the secondary containment is an integral part of the overall containment system, it is required at all times that primary containment is required. Moreover, secondary containment is required during fuel handling operations and whenever work is being performed on the reactor or its connected systems in the reactor building since their operation could result in inadvertent release of radioactive material.

The standby gas treatment system[6] filters and exhausts the reactor building atmosphere to the stack during secondary containment isolation conditions, with a minimum release of radioactive materials from the reactor building to the environs.

Two separate filter trains are provided each having 100% capacity.[6] If one filter train becomes inoperable, there is no immediate threat to secondary containment and reactor operation may continue while repairs are being made. Since the test interval for this system is once/month (Specification 4.5), the time out-of-service allowance of 7 days is based on considerations presented in the bases in Specification 3.2 for a one-out-of-two system.

References

(1) FDSAR, Volume I, Section V-1
(2) FDSAR, Volume I, Section V-1.4.1
(3) FDSAR, Volume I, Section V-1.7
(4) Licensing Application, Amendment 11, Question III-25
(5) FDSAR, Volume I, Section V-2
(6) FDSAR, Volume I, Section V-2.4
(7) Licensing Application, Amendment 42
(8) Licensing Application, Amendment 32, Question 3
(9) Robbins, C. H., "Tests on a Full Scale 1/48 Segment of the Humboldt Bay Pressure Suppression Containment," GEAP-3596, November 17, 1960.
(10) Bodega Bay Preliminary Hazards Summary Report, Appendix 1, Docket 50-205, December 28, 1962.

22

4. Reactor Building to Suppression Chamber Vacuum Breakers

a. The reactor building to suppression chamber vacuum breakers and associated instrumention, including set point, shall be checked for proper operation every three months.

b. During each refueling outage each vacuum breaker shall be tested to determine that the force required to open the vacuum breaker from closed to fully open does not exceed the force specified in Specification 3.5.A.3.a. The air-operated vacuum breaker instrumentation shall be calibrated during each refueling outage.

5. Pressure Suppression Chamber - Drywell Vacuum Breakers

a. Periodic Operability Tests

22 Once each month and following any release of energy which would tend to increase pressure to the suppression chamber, each operable suppression chamber - drywell vacuum breaker shall be exercised. Operation of position switches, indicators and alarms shall be verified monthly by operation of each operable vacuum breaker.

b. Refueling Outage Tests

(1) All suppression chamber - drywell vacuum breakers shall be tested to determine the force required to open each valve from fully closed to fully open.

(2) The suppression chamber - drywell vacuum breaker position indication and alarm systems shall be calibrated and functionally tested.

(3) At least four of the suppression chamber - drywell vacuum breakers shall be inspected. If deficiencies are found, all vacuum breakers shall be inspected and deficiencies corrected such that Specification 3.5.A.4.a can be met.

(4) A drywell to suppression chamber leak rate test shall demonstrate that with an initial differential

22 pressure of not less than 1.0 psi, the differential pressure decay rate shall not exceed the equivalent of air flow through a 2-inch orifice.

J. Reactor Building and Standby Gas Treatment System

1. Preoperational secondary containment capability tests shall be conducted after isolating the reactor building and placing either standby gas treatment system filter train in operation. The tests shall be performed under an adequate number of different environmental wind conditions to enable valid extrapolation of the test results.

2. The tests shall demonstrate the capability to maintain a 1/4 inch of water vacuum under calm wind conditions with a filter train flow rate of not more than 5000 cfm.

3. Periodic secondary containment capability of tests shall be conducted at each refueling outage prior to refueling.

4. The results of the secondary containment capability tests shall be the subject of a summary technical report which can be included in the reports specified in Section 6.6.

K. Charcoal Filter Testing

1. The charcoal filters of the standby gas treatment system shall be tested to demonstrate a halogen removal efficiency of not less than 99%. These tests shall be conducted:

   a. Whenever a filter is changed.

   b. Whenever work is performed which could affect the filter system efficiency.

   c. At each refueling outage.

   d. At intervals not to exceed six months between refueling outages.

the required closing time. 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 steam line break dose calculations.

Surveillance of the suppression chamber-reactor building vacuum breakers consists of operability checks and leakage tests (conducted as part of the containment leak - tightness tests). These vacuum breakers are normally in the closed position and open only during tests or an accident condition. As a result, a testing frequency of three months for operability is considered justified for this equipment. Inspections and calibrations are performed during the refueling outages, this frequency being based on equipment quality, experience, and engineering judgment.

The fourteen suppression chamber-drywell vacuum relief valves are designed to open to the full open position (the position that curtain area is equivalent to valve bore) with a force equivalent to a 0.5 psi differential acting on the suppression chamber face of the valve disk. This opening specification assures that the design limit of 2.0 psid between the drywell and external environment is not exceeded. Once each refueling outage each valve is tested to assure that it will open fully in response to a force less than that specified. Also it is inspected to assure that it closes freely and operates properly.

22 The containment design has been examined to establish the allowable bypass area between the drywell and suppression chamber as 10.5 in.^{2} (expressed as vacuum breaker open area). This is equivalent to one vacuum breaker disk off its seat 0.371 inch; this length corresponds to an angular displacement of 1.25°. A conservative allowance of 0.10 inch has been selected as the maximum permissible valve opening. Valve closure within this limit may be determined by light indication from two independent position detection and indication systems. Either system provides a control room alarm for a non-seated valve.

At the end of each refueling cycle, a leak rate test shall be performed to verify that significant leakage flow paths do not exist between the drywell and suppression chamber. The drywell pressure will be increased by at least 1 psi with respect to the suppression chamber pressure. The pressure transient (if any) will be monitored with a sensitive pressure gauge. If the drywell pressure cannot be increased by 1 psi over the suppression chamber pressure it would be because a significant leakage path exists; in this event the leakage source will be identified and eliminated before power operation is resumed. If the drywell pressure can be increased by 1 psi over the suppression chamber the rate of change of the suppression

chamber pressure must not exceed a rate equivalent to the rate of air flow from the drywell to the suppression chamber through a 2-inch orifice. In the event the rate of change of pressure exceeds this value then the source of leakage will be identified and eliminated before power operation is resumed.

The drywell-suppression chamber vacuum breakers are exercised monthly and immediately following termination of discharge of steam into the suppression chamber. This monitoring of valve operability is intended to assure that valve operability and position indication system performance does not degrade between refueling inspections. When a vacuum breaker valve is exercised through an opening-closing cycle, the position indicating lights are designed to function as follows:

22

Full Closed	2 Green - On
(Closed to $\leq$ 0.10" open)	2 Red - Off
Open 0.10"	2 Green - Off
(> 0.10" open to full open)	2 Red - On

During each refueling outage, four suppression chamber-drywell vacuum breakers will be inspected to assure components have not deteriorated. Since valve internals are designed for a 40-year lifetime, an inspection program which cycles through all valves in about one-tenth of the design lifetime is extremely conservative. The alarm systems for the vacuum breakers will be calibrated during each refueling outage. This frequency is based on experience and engineering judgement.

Initiating reactor building isolation and operation of the standby gas treatment system to maintain a 1/4 inch of water vacuum, tests the operation of the reactor building isolation valves, leakage tightness of the reactor building and performance of the standby gas treatment system. The filter train flow rate of 1200 cfm is consistent with the secondary containment system. Checking the initiating sensors and associated trip channels demonstrates the capability for automatic actuation. Performing these tests prior to refueling demonstrates secondary containment capability prior to extensive fuel handling operations associated with the outage. Testing prior to each refueling outage and between outages gives sufficient confidence of standby gas treatment system performance capability. A charcoal filter efficiency of 99% for halogen removal is adequate.

The in-place testing of charcoal filters is performed using Freon-112* which is injected into the system upstream of the charcoal filters. Measurement of the Freon concentration upstream and downstream of the charcoal filters is made using a gas chromatograph. The ratio of the inlet and outlet concentrations gives an overall indication of the leak tightness

*Trade name of E.I. duPont de Nemours & Company

of the system. Although this is basically a leak test, since the filters have charcoal of known efficiency and holding capacity for elemental iodine and/or methyl iodide, the test also gives an indication of the relative efficiency of the installed system. The test procedure is an adaptation of test procedures developed at the Savannah River Laboratory which were described in the Ninth AEC Air Cleaning Conference.*

High efficiency particulate filters are installed before and after the charcoal filters to minimize potential release of particulates to the environment and to prevent clogging of the iodine filters. An efficiency of 99% is adequate to retain particulates that may be released to the reactor building following an accident. This will be demonstrated by testing with DOP as testing medium.

*D. R. Muhbaier, "In Place Nondestructive Leak Test for Iodine Absorbers," Proceedings of the Ninth AEC Air Cleaning Conference, USAEC Report CONF-660904, 1966.

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

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING

SUPPORTING AMENDMENT NO. 6 TO LICENSE NO. DPR-16

CHANGE NO. 22 TO THE TECHNICAL SPECIFICATIONS

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION UNIT NO. 1

DOCKET NO. 50-219

Introduction

In a letter dated January 22, 1973, the Regulatory staff (staff) requested specific information from Jersey Central Power & Light Company (JCPL) for our review of the Oyster Creek Nuclear Generating Station (Oyster Creek) absorption (suppression) chamber to drywell vacuum breaker system. JCPL replied in letters dated October 8, 1973 and February 26, 1974 with partial response to our request. By letter dated January 30, 1974, we: (1) requested JCPL to submit the information not yet supplied; (2) listed additional reactor operating restrictions regarding inoperable vacuum breakers; and (3) advised JCPL of additional surveillance requirements needed to verify operability of vacuum breakers.

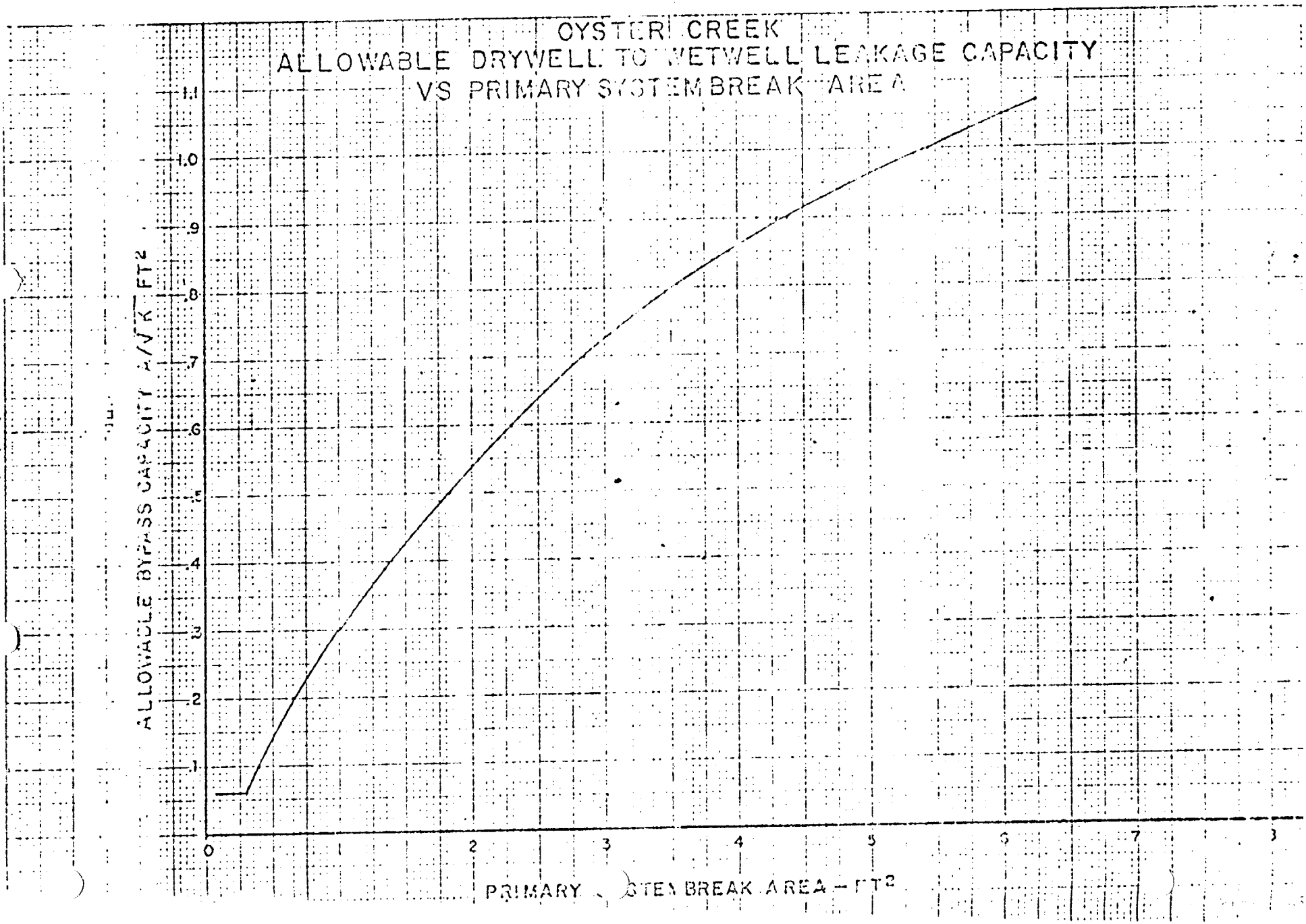
In a letter dated May 28, 1974, JCPL submitted the remainder of the requested information and "Technical Specification Change Request No. 24" for the Oyster Creek suppression chamber to drywell vacuum breaker system. This requested change provides limiting conditions for operation, surveillance requirements, and the basis for the vacuum breaker system modification as described in the JCPL letter dated May 28, 1974.

Evaluation

The JCPL's "Technical Specification Change Request No. 24" for the Oyster Creek plant, submitted by letter dated May 28, 1974, proposed changes to the limiting conditions for operation, surveillance requirements, and bases for the suppression chamber to drywell vacuum breaker system technical specifications which reflect system modifications. The system modifications include an increase in the force available to close the vacuum breakers and the installation of a position indication and alarm subsystem. These modifications provide reasonable assurance that each vacuum breaker will remain in a normally closed position following exercising or transient conditions and will alert the operators to non-standard conditions of the vacuum breakers.

A small break in the primary system concurrent with an excessive leakage path between the drywell and suppression chamber could result in an internal pressure that would exceed the structural capability of the primary containment. The results of the analysis for bypass leakage capacity vs. primary system break area shown in Figure 1 were prepared by General Electric Company (GE), and presented in JCPL's letter dated May 28, 1974. For break areas of less than about 0.3 sq. ft., the pressure differential between the drywell and suppression chamber stabilizes at a value equivalent to the hydrostatic head (4 ft.) of water in the downcomer, i.e., 1.8 psi. Because the break is small, this situation may prevail for a relatively long time and thereby maximizes the steam bypassed into the suppression chamber air space. Since small breaks in the primary system have the potential for yielding the most severe pressurization, their associated bypass areas establish the limiting condition. From Figure 1, it can be seen that the maximum bypass leakage capacity which may be tolerated is $\frac{A}{\sqrt{K}}$ = 0.060 sq. ft., i.e., the limiting region of the curve; this value is equivalent to a vacuum breaker bypass area of about 10.5 sq. in. and a plate orifice flow area of about 0.1 sq. ft. (4.2 in. diameter). Based on the 10.5 sq. in. bypass opening developed above, the maximum amount that any one vacuum breaker could be off its seat is 0.371 inch. This arc length corresponds to an angular displacement of 1.25°.

JCPL proposes to provide each vacuum breaker with a two channel indication and alarm system that will alert the operator if one or more vacuum breakers are open more than 0.1 inch. If such a condition should exist, the open valve(s) would be immediately locked closed and considered inoperable. This action can be taken because the valves are mounted outside the suppression chamber and are physically accessible. The major components of each channel include limit switches (two per valve), time delay relays (one per valve), indicator lamps (two per valve), and a control room annunicator (one per channel). Each limit switch is mounted so that it responds to rotational movement of the valve shaft. The limit switch can detect a differential travel of 0.01 in.; however, the overall alignment of the system will be such that the alarm will be initiated if a valve disc opens 0.10 in. or more. Each actuated switch energizes a time delay relay which is set to minimize false alarms that may occur from vibrations. The relays are connected to red and green indicating lamps that light when the vacuum breakers are open or closed, respectively. In series with the time delay relays are relays whose contacts are connected in the control room alarm system to provide annunication if one or more valves are open at least 0.10 in. The system is designed to meet single failure criteria and has been developed using paragraphs 4.2, 4.3, 4.6 and 4.18 of IEEE 279-1971 as references. This system provides an acceptable means for annunicating a vacuum breaker opening greater than allowable.



Conclusion

We have concluded, based on the reasons discussed above, that: (1) because the change does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the change does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

John I. Riesland
Operating Reactors Branch #3
Directorate of Licensing

George Lear
George Lear, Chief
Operating Reactors Branch #3
Directorate of Licensing

Date: SEP 27 1974

The suppression chamber drywell vacuum breaker limiting condition for operation has been established as an opening not greater than 0.10 in. This condition requires that a faulty vacuum breaker be considered inoperable and immediately locked closed. The allowable minimum number of operable vacuum breakers is established from data obtained during pressure suppression tests performed by GE.1/ In Amendment 15 of the Oyster Creek FDSAR, Section II, the licensee states that an area of 2920 sq. in. for flow of non-condensible gases from the suppression chamber to the drywell is the minimum required area. To conform with this requirement, at least (minimum) 12 of the 14 vacuum breakers (18" diameter) must be operable. The reactor must be shutdown if more than two vacuum breakers become inoperable and are secured in the closed position. The technical specifications will be revised to include this limitation. Inasmuch as the 2920 sq. in. area for flow of non-condensable gases from the suppression chamber to the drywell is maintained under the proposed change, there is no decrease in the margin of safety.

Tests to verify acceptable leakage from the drywell to the suppression chamber should be performed during each refueling outage. This surveillance requirement will be included in the technical specifications. The allowable bypass area from the drywell to suppression chamber, as stated above, is that area equivalent to an orifice with a diameter of approximately 4.2 in. JCPL has specified that the maximum allowable leakage from the drywell to the suppression chamber, during tests, should be equivalent to flow through a 2-inch diameter orifice with an initial differential pressure of not less than 1 psi. This is an acceptable test limit for leakage and will be included in the technical specifications.

A fifteen day period is specified by JCPL for correction of malfunctions in one of the two positions alarm systems for one or more vacuum breakers. This period is acceptable provided that one position alarm system for each operable drywell to suppression chamber vacuum breaker is functional and that all vacuum breakers are physically verified to be closed immediately and daily thereafter. This surveillance requirement will be included in the technical specifications. We will also include in the technical specifications a requirement that monthly tests of each operable vacuum breaker be performed to verify operability of the position indication and alarm system and that calibration of the alarm system for each vacuum breaker be performed during each refueling outage.

JCPL did not propose technical specifications for the reactor building to suppression chamber vacuum breakers. A standard technical specification is being used for other plants with Mark I containment structures. This will be added to the Oyster Creek technical specifications.

1/ GEAP-3596, "Tests of Full Scale 1/48 Segment of the Humboldt Bay Pressure Suppression Containment," C. H. Robbins, November 17, 1960.