

March 7, 1994

Docket No. 50-219

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

Dear Mr. Barton:

SUBJECT: ISSUANCE OF AMENDMENT (TAC NO. M88449)

The Commission has issued the enclosed Amendment No. 168 to Facility Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station, in response to your application dated December 16, 1993.

The amendment revises the Technical Specifications to clarify the requirements for maintaining secondary containment integrity when one or more Reactor Building Ventilation supply and exhaust valves are declared inoperable. The Technical Specifications add a new Limiting Condition for Operation, Basis Statement and Surveillance Requirements for these isolation valves.

A copy of the related Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by:

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

Enclosures:

1. Amendment No. 168 to DPR-16
2. Safety Evaluation

cc w/enclosures:  
See next page

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NAME	SNorris	ADromerick:cn	JStolz	MZong	
DATE	2/16/94	2/17/94	2/17/94	2/23/94	1/1

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*one additional change not issued before 3/3/94*  
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UNITED STATES  
NUCLEAR REGULATORY COMMISSION

WASHINGTON, D.C. 20555-0001

March 7, 1994

Docket No. 50-219

Mr. John J. Barton  
Vice President and Director  
GPU Nuclear Corporation  
Oyster Creek Nuclear Generating Station  
Post Office Box 388  
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Sincerely,

A handwritten signature in cursive script that reads "Alexander W. Dromerick".

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

Enclosures:

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2. Safety Evaluation

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See next page

Mr. John J. Barton  
GPU Nuclear Corporation

Oyster Creek Nuclear  
Generating Station

cc:

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

GPU NUCLEAR CORPORATION

AND

JERSEY CENTRAL POWER & LIGHT COMPANY

DOCKET NO. 50-219

OYSTER CREEK NUCLEAR GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 168  
License No. DPR-16

1. The Nuclear Regulatory Commission (the Commission) has found that:
  - A. The application for amendment by GPU Nuclear Corporation, et al. (the licensee), dated December 16, 1993, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
  - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
  - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
  - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
  - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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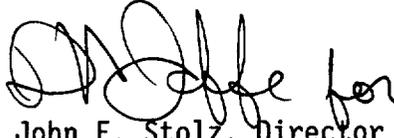
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-16 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 168, are hereby incorporated in the license. GPU Nuclear Corporation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of issuance, to be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



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

Attachment:  
Changes to the Technical  
Specifications

Date of Issuance: March 7, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 168

FACILITY OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

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

<u>Remove</u>	<u>Insert</u>
1.0-3	1.0-3
3.5-6	3.5-6
3.5-7	3.5-7
3.5-8	3.5-8
3.5-9	3.5-9
3.5-10	3.5-10
3.5-11	3.5-11
4.5-11	4.5-11

#### 1.14 SECONDARY CONTAINMENT INTEGRITY

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

- A. At least one door at each access opening is closed.
- B. The standby gas treatment system is operable.
- C. All automatic secondary containment isolation valves are operable or are secured in the closed position.

#### 1.15 (DELETED)

#### 1.16 RATED FLUX

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

#### 1.17 REACTOR THERMAL POWER-TO-WATER

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

#### 1.18 PROTECTIVE INSTRUMENTATION LOGIC DEFINITIONS

##### A. Instrument Channel

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

##### B. Trip System

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

2. Upon the accidental loss of SECONDARY CONTAINMENT INTEGRITY, restore, SECONDARY CONTAINMENT INTEGRITY within 4 hours, except as provided in specification 3.5.B.3.
3. With one or more of the automatic secondary containment isolation valves inoperable:
  - a. Maintain at least one automatic secondary containment isolation valve in each affected penetration OPERABLE.
  - b. Within 8 hours restore the inoperable automatic secondary containment isolation valve(s) to OPERABLE status or isolate each affected penetration with at least one valve secured in the closed position.
4. If Specifications 3.5.B.2 or 3.5.B.3 cannot be met:
  - a. During Power Operation:
    - (1) Have the reactor mode switch in the shutdown mode position within the following 24 hours.
    - (2) Cease all work on the reactor or its connected systems in the reactor building which could result in inadvertent releases of radioactive materials.
    - (3) Cease all operations in, above or around the Spent Fuel Storage Pool that could cause release of radioactive materials.
  - b. During refueling:
    - (1) Cease fuel handling operations or activities which could reduce the shutdown margin (excluding reactor coolant temperature changes).
    - (2) Cease all work on the reactor or its connected systems in the reactor building which could result in inadvertent releases of radioactive materials.
    - (3) Cease all operations in, above or around the Spent Fuel Storage Pool that could cause release of radioactive materials.
5. Two separate and independent standby gas treatment system circuits shall be operable when secondary containment is required except as specified by Specification 3.5.B.6.

6. With one standby gas treatment system circuit inoperable:

a. During Power Operation:

- (1) Demonstrate the operability of the other standby gas treatment system circuit within 2 hours unless significant painting, fire, or chemical release has taken place in the reactor building within the previous 12 hours. In this event, demonstration of operability shall take place within 1 hour of the expiration of the 12 hour period, and
- (2) Continue to demonstrate the operability of the standby gas treatment system circuit once per 24 hours until the inoperable standby gas treatment circuit is returned to operable status.
- (3) Restore the inoperable standby gas treatment circuit to operable status within 7 days or be subcritical with reactor coolant temperature less than 212°F within the next 36 hours.

b. During Refueling:

- (1) Demonstrate the operability of the redundant standby gas treatment system within 2 hours unless significant painting, fire, or chemical release has taken place in the reactor building within the previous 12 hours. In this event, demonstration of operability shall take place within 1 hour of the expiration of the 12 hour period, and
- (2) Continue to demonstrate the operability of the redundant standby gas treatment system once per 7 days until the inoperable system is returned to operable status.
- (3) Restore the inoperable standby gas treatment system to operable status within 30 days or cease all spent fuel handling, core alterations or operation that could reduce the shutdown margin (excluding reactor coolant temperature changes).

7. If Specifications 3.5.B.5 and 3.5.B.6 are not met, reactor shutdown shall be initiated and the reactor shall be in the cold shutdown condition within 24 hours and the condition 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 materials 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.

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

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

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

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

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

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

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

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) The 4% oxygen limit was established by analysis of the Generation and Mitigation of Combustible Gas Mixtures in Inerted BWR Mark I Containments.(12)

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<sub>2</sub> limit.

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient, while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping as a result of a seismic or other event initiating dynamic loads. It is, therefore, required that all snubbers required to protect the primary coolant system or any other safety system or component be operable whenever the systems they protect are required to be operable.

The purpose of an engineering evaluation is to determine if the components protected by the snubber were adversely affected by the inoperability of the snubber. This ensures that the protected component remains capable of meeting the designed service. A documented visual inspection will usually be sufficient to determine system operability.

Because snubber protection is required only during low probability events, a period of 72 hours is allowed for repairs or replacements.

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 shutdown 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.

When secondary containment is not maintained, the additional restrictions on operation and maintenance give assurance that the probability of inadvertent releases of radioactive material will be minimized. Maintenance will not be performed on systems which connect to the reactor vessel lower than the top of the active fuel unless the system is isolated by at least one locked closed isolation valve.

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 one 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.

Two automatic secondary containment isolation valves are installed in each Reactor Building ventilation system supply and exhaust duct penetration. Both isolation valves for each supply duct penetration are located inside the secondary containment boundary, and the two exhaust duct penetration isolation valves are located outside of the secondary containment boundary. Removal of an inboard supply or exhaust valve (closest to the boundary) is permitted only when secondary containment is not required. The outboard isolation supply or exhaust valve can be removed when secondary containment is required as long as the inboard valve is secured in the closed position.

- 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 I, Docket 50-205, December 28, 1962.
  - (11) Report H. R. Erickson, Bergen-Paterson To K. R. Goller, NRC, October 7, 1974. Subject: Hydraulic Shock Sway Arrestors.
  - (12) General Electric NEDO-22155 "Generation and Mitigation of Combustible Gas Mixtures in Inerted BWR Mark I Containment" June 1982.
  - (13) Oyster Creek Nuclear Generating Station, Mark I Containment Long-Term Program, Plant Unique Analysis Report, Suppression Chamber and Vent System, MPR-733; August, 1982.
  - (14) Oyster Creek Nuclear Generating Station, Mark I Containment Long-Term Program, Plant Unique Analysis Report, Torus Attached Piping, MPR-734; August, 1982.

e. Mechanical Snubbers Functional Test Acceptance Criteria

The mechanical snubber functional test shall verify that:

1. The force that initiated free movement of the snubber rod in either tension or compression is less than the specified maximum drag force.
2. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.
3. Snubber release rate, where required, is within the specified range in compression or tension. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

f. Snubber Service Life Monitoring

A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained as required by Specification 6.10.2.1.

Concurrent with the first inservice visual inspection and at least once per 24 months thereafter, the installation and maintenance records for each snubber shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review. If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be re-evaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This re-evaluation, replacement or reconditioning shall be indicated in the records. Service life shall not at any time affect reactor operations.

R. Secondary Containment Isolation Valves

1. Each secondary containment isolation valve shall be demonstrated operable prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator by cycling the valve through at least one complete cycle of full travel. Following maintenance, repair or replacement work on the control or power circuit for the valves, the affected component shall be tested to assure it will perform its intended function in the circuit.
2. At least once per refueling outage, all valves shall be tested for automatic closure by an isolation signal.



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 168

TO FACILITY OPERATING LICENSE NO. DPR-16

GPU NUCLEAR CORPORATION AND  
JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

1.0 INTRODUCTION

By letter dated December 16, 1993, GPU Nuclear Corporation (the licensee) requested an amendment to Facility Operating License No. DPR-16. The proposed amendment would change certain Oyster Creek Technical Specifications (TS) relating to secondary containment piping penetration isolation requirements.

2.0 DISCUSSION AND EVALUATION

2.1 Secondary Containment Description

The Oyster Creek reactor building (RB) completely encloses the reactor and its primary containment. During those reactor operational modes which require primary integrity, the RB serves as a secondary containment for control of fission products which could leak from the primary containment. During those operational modes for which primary containment integrity is impractical and accidents involving containment pressurization are unlikely (e.g., refueling), the RB serves as a primary containment.

The RB is provided with a ventilation system (RBVS) system for heating or cooling during normal (non-accident) conditions. The RBVS contains air supply and exhaust ducts which penetrate the RB walls. The supply fans are located outside of the RB and supply air to the RB via two supply duct penetrations. Exhaust air leaves the RB via a single duct penetration. Redundant automatic isolation dampers (butterfly valves) are provided in each duct at each RB penetration to automatically isolate the RB under accident conditions. Under accident conditions, the Standby Gas Treatment System (SGTS) will automatically start, exhausting the RB to the stack to provide a filtered, elevated release of the contaminated RB atmosphere. The RB is designed for a maximum inleakage rate of 200% volume/day. Failure of the butterfly valves to close tightly could result in excessive RB in-leakage exceeding the capability of the SGTS and thereby causing exfiltration or ground-level, unfiltered release of fission products during an accident.

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## 2.2 Changes to Technical Specifications

### 2.2.1 Definition of Secondary Containment Integrity

The Technical Specifications provide a high degree of assurance that the secondary containment will be operable as a fission product control system under accident conditions. TS 1.14 defines three specific requirements necessary for secondary containment integrity. The conditions to be met include:

- A. At least one door at each access opening is closed.
- B. The standby gas treatment system is operable.
- C. All reactor building ventilation system automatic isolation valves are operable or are secured in the closed position.

The proposed amendment would change the last requirement to read "All automatic secondary containment isolation valves are operable or are secured in the closed position." This change would provide consistency in terminology and does not change the scope of the TS or any specific Limiting Conditions for Operation (LCOs) or Surveillance Requirements (SRs). Substitution of the term "secondary containment isolation valves" for "reactor building isolation valves" is acceptable, since, at Oyster Creek, the secondary containment boundaries and RB boundaries coincide. The TS 1.14 change is therefore acceptable.

### 2.2.2 Completion times for repair of valves or isolation of penetration

The more significant of the changes proposed in the licensee's application are intended to permit continued operations when one or more of the butterfly isolation valves is inoperable but integrity of the associated penetration is assured by virtue of another closure device which is not subject to failure to automatically close on demand. TS 3.5.B currently requires that upon accidental loss of secondary containment integrity (i.e., conditions A, B and C above not met), the licensee must restore secondary containment integrity within 4 hours, or:

During power operation:

- (1) Have the reactor mode switch in the shutdown mode position within the following 24 hours.
- (2) Cease all work on the reactor or its connected systems in the reactor building which could result in inadvertent releases of radioactive materials.

- (3) Cease all operations in, above or around the Spent Fuel Storage Pool that could cause release of radioactive materials.

During refueling:

- (1) Cease fuel handling operations or activities which could reduce the shutdown margin (excluding reactor coolant temperature changes).
- (2) Cease all work on the reactor or its connected systems in the reactor building which could result in the inadvertent releases of radioactive materials.
- (3) Cease all operations in, above or around the Spent Fuel Storage Pool that could cause release of radioactive materials.

The above (currently applicable) TS do not acknowledge the use of a single deenergized or locked-closed valve as being sufficient for integrity of a secondary containment penetration. Should a valve become jammed open, the above action requirements must be followed. The amendment would revise the above requirements to insert a provision that, after expiration of the 4 hour period with one or more of the automatic secondary containment isolation valves inoperable, the licensee must:

Maintain at least one automatic secondary containment isolation valve in each affected penetration OPERABLE, and

Within 8 hours restore the inoperable automatic secondary containment isolation valves(s) to OPERABLE status or isolate each affected penetration with at least one valve secured in the closed position.

The proposed change would provide a 4-hour completion time for the case of two inoperable (open and incapable of automatic closure) valves in a penetration, and an 8-hour completion time for the case of a single inoperable valve. After expiration of these completion times the affected penetration(s) must be isolated with at least one secured valve. When isolated with a secured (i.e., deenergized) valve, the penetration is not subject to loss of leaktightness due to equipment failure or spurious signals. Thus operations can be continued until repairs can be completed.

The staff has evaluated the potential effects of the proposed TS changes using the STS (Standard Technical Specifications General Electric Plants, BWR/4, NUREG-1433, September 1992) as guidance. The STS specify an 8-hour completion time for a single inoperable secondary containment isolation valve in one or

more penetrations, and a 4-hour completion time for both secondary containment isolation valves inoperable in one or more penetrations. The STS also permit continued operations with no time limit, with penetration closure provided by a single closed, deactivated valve. Based on the STS guidance, and a determination that there are no unique or unusual features of the Oyster Creek design that would preclude use of the STS as guidance, the proposed TS changes are acceptable.

The STS also specify that when a closed and deenergized valve is used to isolate a penetration, the valve position is to be verified every 31 days. The purpose of this additional requirement is to provide added assurance that the deenergized valve is indeed closed, considering that fact that the valve's control room position indication may also be inoperable. The licensee has not explicitly included such a requirement with the proposed changes. The staff discussed this with licensee personnel, and was informed that at Oyster Creek, the RBVS secondary containment isolation valves are coincidentally tested monthly for leak tightness and proper positioning as part of the SGTS surveillance test program. The monthly SGTS surveillance test involves closure of the RBVS valves while the SGTS draws a negative pressure on the secondary containment. The monthly SGTS operability test would reveal an improperly positioned valve. Since the SGTS test interval coincides with the desired interval for verification of proper closure of isolated penetrations, the SGTS surveillance test provides a high degree of assurance of proper secondary containment isolation valve positioning. Additionally, the licensee's valve tagout administrative procedures include logging of inoperable valves, and the use of red tags at the valve and in the control room to continuously identify valves that are inoperable. The use of such controls provides added assurance that closed and deenergized (secured) valves remain in the correct position. The proposed changes are thus acceptable notwithstanding the omission of the standard 30-day verification requirement.

### 2.2.3 Changes to Bases

The proposed amendment would add a paragraph to the BASES describing permissible conditions for removal of an RBVS secondary containment isolation valve. The redundant isolation valves in the two supply duct penetrations are located inside the RB. The redundant isolation valves in the single exhaust duct penetration are located outside the RB. Due to this configuration, it is possible to remove the downstream isolation valves, if the upstream valves are closed, without loss of secondary containment integrity. Removal of an isolation valve under these conditions is consistent with the Technical Specifications operability requirements and is acceptable.

The staff has reviewed the proposed Technical Specifications changes relating to (a) the definition of secondary integrity, (b) the completion time for actions required when one or more secondary containment piping penetration isolation valves is inoperable, and (c) the removal of an RBVS isolation valve. The staff has determined that the proposed changes are acceptable.

### 3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State official was notified of the proposed issuance of the amendment. The State official had no comments.

### 4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (59 FR 4938). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

### 5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

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