August 13, 1998

Mr. Michael B. Roche
Vice President and Director
GPU Nuclear, Inc.
Oyster Creek Nuclear Generating Station
P.O. Box 388
Forked River, NJ 08731

SUBJECT: OYSTER CREEK - ISSUANCE OF AMENDMENT NO. 196 , RE: DELETION OF TECHNICAL SPECIFICATION TABLE 3.5.2, LIST OF CONTAINMENT ISOLATION VALVES (TAC NO. M97307)

Dear Mr. Roche:

The Commission has issued the enclosed Amendment No. 196 to Facility Operating License No. DPR-16 for the Oyster Creek Nuclear Generating Station, in response to your application dated October 31, 1996.

The amendment deletes Table 3.5.2 which lists automatic primary containment isolation valves. In addition, this amendment clarifies the applicability of an action statement which applies to several limiting conditions for operation in Section 3.5 and deletes closure time requirements for several automatic isolation valves in Section 4.5.F.

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

Sincerely,

Original signed by:

Ronald B. Eaton, Senior Project Manager
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket No. 50-219

Enclosures: 1. Amendment No. 196 to DPR-16
2. Safety Evaluation

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DATED: August 13, 1998

AMENDMENT NO. 196 TO FACILITY OPERATING LICENSE NO. DPR-16-OYSTER CREEK

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J. Zwolinski
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T. Harris (e-mail TLH3)

M. Roche
GPU Nuclear, Inc.

cc:

Ernest L. Blake, Jr., Esquire
Shaw, Pittman, Potts & Trowbridge
2300 N Street, NW
Washington, DC 20037

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406-1415

BWR Licensing Manager
GPU Nuclear Corporation
1 Upper Pond Road
Parsippany, NJ 07054

Mayor
Lacey Township
818 West Lacey Road
Forked River, NJ 08731

Licensing Manager
Oyster Creek Nuclear Generating Station
Mail Stop: Site Emergency Bldg.
P.O. Box 388
Forked River, NJ 08731

Resident Inspector
c/o U.S. Nuclear Regulatory Commission
P.O. Box 445
Forked River, NJ 08731

Kent Tosch, Chief
New Jersey Department of
Environmental Protection
Bureau of Nuclear Engineering
CN 415
Trenton, NJ 08625

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

GPU NUCLEAR, INC.

AND

JERSEY CENTRAL POWER & LIGHT COMPANY

DOCKET NO. 50-219

OYSTER CREEK NUCLEAR GENERATING STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 196
License No. DPR-16

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

   A. The application for amendment filed by GPU Nuclear, Inc. et al., (the licensee), dated October 31, 1996, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 196 , are hereby incorporated in the license. GPU Nuclear, Inc. 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

Cecil O. Thomas

Cecil O. Thomas, Director
Project Directorate I-3
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: August 13, 1998

ATTACHMENT TO LICENSE AMENDMENT NO. 196

FACILITY OPERATING LICENSE NO. DPR-16

DOCKET NO. 50-219

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

Remove	Insert
1.0-2	1.0-2
3.5-1	3.5-1
3.5-2	3.5-2
3.5-3	3.5-3
3.5-3a	3.5-3a
3.5-4	3.5-4
3.5-8	3.5-8
3.5-9	3.5-9
3.5-10	3.5-10
3.5-11	3.5-11
3.5-12	3.5-12
3.5-13	--------
4.5-2	4.5-2
4.5-3	4.5-3
4.5-11	4.5-11
4.5-12	4.5-12

1.7 COLD SHUTDOWN CONDITION

The reactor is in the COLD SHUTDOWN CONDITION when the reactor is in the SHUTDOWN CONDITION, and (except during REACTOR VESSEL PRESSURE TESTING), the reactor coolant system is maintained at less than 212°F and vented.

1.8 PLACE IN SHUTDOWN CONDITION

Proceed with and maintain an uninterrupted normal plant shutdown operation until the SHUTDOWN CONDITION is met.

1.9 PLACE IN COLD SHUTDOWN CONDITION

Proceed with and maintain an uninterrupted normal plant shutdown operation until the COLD SHUTDOWN CONDITION is met.

1.10 PLACE IN ISOLATED CONDITION

Proceed with and maintain an uninterrupted normal isolation of the reactor from the turbine condenser system including closure of the main steam isolation valves.

1.11 REFUEL MODE

The reactor is in the REFUEL MODE when the reactor mode switch is in the REFUEL MODE position and there is fuel in the reactor vessel. In this mode the refueling platform interlocks are in operation.

1.12 REFUELING OUTAGE

For the purpose of designating frequency of testing and surveillance, a REFUELING OUTAGE shall mean a regularly scheduled REFUELING OUTAGE. Following the first REFUELING OUTAGE, successive tests or surveillances shall be performed at least once per 24 months.

1.13 PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY means that the drywell and adsorption chamber are closed and all of the following conditions are satisfied:

A. All non-automatic primary containment isolation valves which are not required to be open for plant operation are closed.
B. At least one door in the airlock is closed and sealed.
C. All automatic primary containment isolation valves are OPERABLE or the affected penetration is isolated.
D. All blind flanges and manways are closed.

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. At any time that the nuclear system is pressurized above atmospheric or work is being done which has the potential to drain the vessel and irradiated fuel is in the vessel, the suppression pool water volume and temperature shall be maintained within the following limits.

   a. Maximum water volume - 92,000 ft^3
   b. Minimum water volume - 82,000 ft^3
   c. Maximum water temperature

      (1) During normal POWER OPERATION - 95°F

      (2) During testing which adds heat to the suppression pool, the water temperature shall not exceed 10°F above the normal POWER OPERATION limit specified in (1) above. In connection with such testing, the pool temperature must be reduced to below the normal POWER OPERATION limit specified in (1) above within 24 hours.

      (3) The reactor shall be scrammed from any operating condition if the pool temperature reaches 110°F. POWER OPERATION shall not be resumed until the pool temperature is reduced below the normal POWER OPERATION limit specified in (1) above.

      (4) During reactor isolation conditions, the reactor pressure vessel shall be depressurized to less than 180 psig at normal cooldown rates if the pool temperature reaches 120°F.

   d. If the limits of Specification 3.5.A.1.a, b or c(1) are exceeded, the reactor shall be PLACED IN the COLD SHUTDOWN CONDITION within 24 hours.

2. Maintenance and repair, including draining of the suppression pool, may be performed provided that the following conditions are satisfied:

   a. The reactor mode switch is locked in the refuel or shutdown position.

   b. (1) There is an OPERABLE flow path capable of taking suction from the condensate storage tank and transferring water to the reactor vessel, and

      (2) The fire protection system is OPERABLE.

   c. The reactor coolant system is maintained at less than 212°F and vented.

   d. At least one core spray pump, and system components necessary to deliver rated core spray flow to the reactor vessel, must remain OPERABLE to the extent that the pump and any necessary valves can be started or operated from the control room or from local control stations, and the torus is mechanically intact.

   e. (1) No work shall be performed on the reactor or its connected systems which could result in lowering the reactor water level to less than 4'8" above the TOP OF the ACTIVE FUEL and the condensate storage tank level is greater than thirty (30) feet (360,000 gallons). At least two redundant systems including core spray pumps and system components must remain OPERABLE as defined in d. above.

      or

      (2) The reactor vessel head, fuel pool gate, and separator-dryer pool gates are removed and the water level is above elevation 117 feet.

      NOTE: When filling the reactor cavity from the condensate storage tank and draining the reactor cavity to the condensate storage tank, the 30 foot limit does not apply provided there is sufficient amount of water to complete the flooding operation.

3. 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 or during REACTOR VESSEL PRESSURE TESTING.

a. With one or more of the automatic containment isolation valves inoperable:

(1) Maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 4 hours (48 hours for the traversing in-core probe system) either;

(a) Restore the inoperable valve(s) to OPERABLE status or

(b) Isolate each affected penetration by use of at least one deactivated automatic valve secured in the isolation position, or

(c) Isolate each affected penetration by use of at least one closed manual valve or blind flange.

(2) If Specification 3.5.A.3 or the provisions of Specifications 3.5.A.3.a.(1)(a), (b) or (c) can not be met, the reactor shall be PLACED IN the COLD SHUTDOWN CONDITION within 24 hours.

(3) An inoperable containment isolation valve of the shutdown cooling system may be opened with a reactor water temperature equal to or less than 350°F in order to PLACE the reactor IN the COLD SHUTDOWN CONDITION. The inoperable valve shall be returned to the OPERABLE status prior to placing the reactor in a condition where PRIMARY CONTAINMENT INTEGRITY is required.

b. If the primary containment air lock is inoperable, per Specification 4.5.C.2, restore the inoperable air lock to OPERABLE status within the 24 hours or be in at least a SHUTDOWN CONDITION within the next 12 hours and in cold shutdown within the following 24 hours.

4. <u>Reactor Building to Suppression Chamber Vacuum Breaker System</u>

a. Except as specified in Specification 3.5.A.4.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 of 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.

c. If the limits of Specification 3.5.A.4.a are exceeded, reactor shutdown shall be initiated and the reactor shall be in a COLD SHUTDOWN CONDITION within 24 hours.

5. 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.5.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.

      (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.

   d. If Specifications 3.5.A.5 (a), (b) or (c) can not be met, the reactor shall be PLACED IN the COLD SHUTDOWN CONDITION within 24 hours.

6. After completion of the startup test program and demonstration of plant electrical output, the primary containment atmosphere shall be reduced to less than 4.0% 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.

7. Deleted.

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.^{3}, indicate that the corner room ambient temperature would not exceed the core spray and containment spray pump motor operating temperature limits and, therefore, would not adversely affect the long-term core cooling capability. The maximum water volume limit allows for an operating range without significantly affecting accident analyses with respect to free air volume in the absorption chamber. For example, the containment capability(8) with a maximum water volume of 92,000 ft.^{3} is reduced by not more than 5.5% metal-water reaction below the capability with 82,000 ft.^{3}.

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

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

The function of the primary containment isolation valves (PCIVs), in combination with other accident mitigation systems, is to limit fission product release during and following postulated Design Basis Accidents (DBAs) to within limits. Primary containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a DBA.

The OPERABILITY requirements for PCIVs help ensure that an adequate primary containment boundary is maintained during and after an accident by minimizing potential paths to the environment. Therefore, the OPERABILITY requirements provide assurance that primary containment function assumed in the safety analyses will be maintained. These isolation devices are either passive or active (automatic). Manual valves, deactivated automatic valves secured in their closed position (including check valves with flow through the valve secured), blind flanges, and closed systems are considered passive devices. Check valves, or other automatic valves designed to close without operator action following an accident, are considered active devices. Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analyses. One of these barriers may be a closed system.

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-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.

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_2 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.

b. If the airlock is opened during a period when Primary Containment is not required, it need not be tested while Primary Containment is not required, but must be tested at P_a prior to returning the reactor to an operating mode requiring PRIMARY CONTAINMENT INTEGRITY.

D. Primary Containment Leakage Rates shall be limited to:

1. The maximum allowable Primary Containment leakage rate is 1.0 L_a. The maximum allowable Primary Containment leakage rate to allow for plant startup following a type A test is 0.75 L_a. The leakage rate acceptance criteria for the Primary Containment Leakage Rate Testing Program for Type B and Type C tests is $\leq$0.60 L_a at P_a.

2. The leakage rate acceptance criteria for an MSIV shall be 0.05(0.75) L_a at P_a.

3. The leakage rate acceptance criteria for the drywell airlock shall be $\leq$0.05 L_a when measured or adjusted to P_a.

E. Continuous Leak Rate Monitor

1. When the primary containment is inerted, the containment shall be continuously monitored for gross leakage by review of the inerting system makeup requirements.

2. This monitoring system may be taken out of service for the purpose of maintenance or testing but shall be returned to service as soon as practical.

F. Functional Test of Valves

1. All automatic primary containment isolation valves shall be tested for automatic closure by an isolation signal during each REFUELING OUTAGE and the isolation time determined to be within its limit. The following valves are required to close in the time specified below:

   Main steam line isolation valves: $\geq$ 3 seconds and $\leq$ 10 seconds

2. Each automatic primary 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 and verifying the isolation time limit is met. 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.

3. Quarterly, during periods of sustained POWER OPERATION, each main steam isolation valve shall be closed (one at a time) and its closure time verified to be within the limits of Specification 4.5.F.1 above. Such testing shall be conducted with reactor power not greater than 50% of rated power.

4. Reactor Building to Suppression Chamber Vacuum Breakers

    a. The reactor building to suppression chamber vacuum breakers and associated instrumentation, including setpoint, 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.4.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

        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.

The Primary Containment Leakage Rate Testing Program conforms with this guidance.

The maximum allowable leakage rate for the primary containment (L_a) is 1.0 percent by weight of the containment air per 24 hours at the design basis LOCA maximum peak containment pressure (P_a). As discussed below, P_a for the purpose of containment leak rate testing is 35 psig.

The penetration and air purge piping leakage test frequency, along with the containment leak rate tests, is adequate to allow detection of leakage trends. Whenever a double gasketed penetration (primary containment head equipment hatches and the absorption chamber access hatch) is broken and remade, the space between the gaskets is pressurized to determine that the seals are performing properly. The test pressure of 35 psig is consistent with the accident analyses and the maximum preoperational leak rate test pressure.

Monitoring the nitrogen makeup requirements of the inerting system provides a method of observing leak rate trends and would detect gross leaks in a very short time. This equipment must be periodically removed from service for test and maintenance, but this out-of-service time will be kept to a practical minimum.

Automatic primary containment isolation valves are provided to maintain PRIMARY CONTAINMENT INTEGRITY following the design basis loss-of-coolant accident. Closure times for the automatic primary containment isolation valves are not critical because it is on the order of minutes before significant fission product release to the containment atmosphere for the design basis loss of coolant accident. These valves are highly reliable, see infrequent service and most of them are normally in the closed position. Therefore, a test during each REFUELING OUTAGE is sufficient.

Large lines connecting to the reactor coolant system, whose failure could result in uncovering the reactor core, are supplied with automatic isolation valves (except containment cooling). Closure times restrict coolant loss from the circumferential rupture of any of these lines outside primary containment to less than that for a main steam line break (the design basis accident for outside containment line breaks). The minimum time for main steam isolation valve (MSIV) closure of 3 seconds is based on the transient analysis that shows the pressure peak 76 psig below the lowest safety valve setting. The maximum time for MSIV closure of 10 seconds is based on the value assumed for the main steam line break dose calculations and restricts coolant loss to prevent uncovering the reactor core. Since the main steam line isolation valves are normally in the open position, more frequent testing is specified. Per ASME Boiler and Pressure Vessel Code, Section XI, the quarterly full closure test will ensure OPERABILITY and provide assurance that the valves maintain the required closing time.

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 judgement.

The 14 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.

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 196

TO FACILITY OPERATING LICENSE NO. DPR-16

GPU NUCLEAR, INC. AND

JERSEY CENTRAL POWER & LIGHT COMPANY

OYSTER CREEK NUCLEAR GENERATING STATION

DOCKET NO. 50-219

1.0 INTRODUCTION

By letter dated October 31, 1996, GPU Nuclear, Inc. (the licensee) submitted a request for changes to the Oyster Creek Nuclear Generating Station Technical Specifications (TSs). The amendment deletes Table 3.5.2 which lists automatic primary containment isolation valves (CIVs). In addition, this amendment clarifies the applicability of an action statement which applies to several limiting conditions for operation in Section 3.5 and deletes closure time requirements for several automatic isolation valves in Section 4.5.F.

2.0 EVALUATION

The Commission's regulations specify those requirements which must be included in TS. The component list is not required to be included in TSs under 10 CFR 50.36 because the CIVs are not safety limits, limiting safety system settings, limiting control settings, limiting conditions for operation, or surveillance requirements, design features, or administrative controls. Deletion of Table 3.5.2 is consistent with the staff's Generic Letter (GL) 91-08, "Removal of Component Lists from Technical Specifications," dated May 6, 1991. The staff finds therefore, this change is acceptable.

In addition to removing Table 3.5.2, the table of CIVs, from the TSs, the licensee has also proposed related changes to TSs 1.13, 3.5.A.3.a., 4.5.F.1., and 4.5.F.2. Currently these TSs refer specifically to the table being deleted, so instead they will be revised to refer to the automatic CIVs as a group. This is in accordance with the provisions of Generic Letter (GL) 91-08, with one difference. GL 91-08 states that reference should be made to all CIVs rather than all automatic CIVs, once the list of CIVs is removed from the TS. In general, TS lists of CIVs at most plants would contain most or all of the CIVs, so it would be appropriate, when removing the list, to instead refer to all CIVs. However, the CIV table at Oyster Creek lists only automatic CIVs; thus, it is appropriate to limit the references in the revised TSs to only the automatic CIVs.

TS 4.5.F.1. currently lists maximum closure times for several automatic CIVs. In accordance with GL 91-08 and the staff's Improved Standard Technical Specifications, the licensee has proposed to delete the closure times for all but the four main steam isolation valves (MSIVs). The requirement to test the closure times of the currently listed valves and other CIVs and demonstrate they are within their limits is not being eliminated from the TS; only the list of valves and the specification of the numerical values of the time limits are being deleted. The proper closure time limits will continue to be denoted in plant procedures. The closure times for the MSIVs are being retained in the revised TS because, unlike the other valves, the closure times of the MSIVs have a direct influence on the consequences of several postulated accident sequences. The staff finds these changes to be acceptable because they are consistent with the provisions of GL 91-08.

Current TS 3.5.A.7. is an action statement which requires that a shutdown be initiated and the plant put in cold shutdown within 24 hours if TS 3.5.A.1.a, b, c(1), or 3.5.A.2. through 3.5.A.5. cannot be met. As an editorial change, the licensee has proposed to delete the current TS and instead move the action requirements to each of the individual TSs. Presenting the information in this way will make the requirements easier to understand without changing the actual requirements. Therefore, the staff finds this editorial clarification to be acceptable.

The licensee has also proposed changes to the associated Bases sections which are consistent with the revised TSs; the staff also finds these changes to be 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 (61 FR 66707). 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.

Principal Contributor: J. Pulsipher

Date: August 13, 1998