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August 28, 1996

U. S. Nuclear Regulatory Commission
Attn.: Document Control Desk
Washington, DC 20555

Dear Sir:

Subject: Oyster Creek Nuclear Generating Station
Docket No. 50-219
Technical Specification Change Request No. 245, Rev 2
Corrected Pages

On July 17, 1996, GPU Nuclear submitted Technical Specification Change Request (TSCR) No. 245, Revision 2, for the Oyster Creek Nuclear Generating Station. During the staff review of that submittal, several typographical errors were identified. Attached are the corrected pages. Please remove the following pages from TSCR 245, Revision 2, and insert these corrected pages: ii; iii, 1.0-5; 3.5-3; 4.5-2; 4.5-3; 4.5-4; 4.5-8; 4.5-10; 4.5-12; and 4.5-13. Please note that the vertical bars in the right hand margin indicate changes requested by TSCR 245, Revision 2, and not the typographic corrections

As the corrected pages restore the submitted pages to the previously approved wording, they do not constitute a change to the Oyster Creek Nuclear Generating Station Technical Specifications. Therefore, no Significant Safety Hazards consideration is required.

If any additional information or assistance is required, please contact Mr. John Rogers of my staff at 609.971.4893.

Arthur H Rone for

Michael B. Roche
Vice President and Director
Oyster Creek



JJB/JJR

A001/1

cc: Oyster Creek NRC Project Manager
Administrator, Region I
Senior Resident Inspector

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*Issued by NRC Order dated 10-24-80

B. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

## 1.24 SURVEILLANCE REQUIREMENTS

Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within the safety limits, and that the limiting conditions of operation will be met. Each surveillance requirement shall be performed within the specified time interval with a maximum allowable extension not to exceed 25% of the surveillance interval. [1]

Surveillance requirements for systems and components are applicable only during the modes of operation for which the system or components are required to be operable, unless otherwise stated in the specification.

This definition establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance, e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are performed at each refueling outage and are specified with a fuel cycle length surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for the surveillance that are not performed during refueling outages. The limitation of this definition is based on engineering judgement and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

## 1.25 APPENDIX J TEST PRESSURE

For the purpose of conducting leak rate tests to meet 10 CFR 50 Appendix J, $P_a$ = 35 psig.

## 1.26 FRACTION OF LIMITING POWER DENSITY (FLPD)

The fraction of limiting power density is the ratio of the linear heat generation rate (LHGR) existing at a given location to the design LHGR for that bundle type.

## 1.27 MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD)

The maximum fraction of limiting power density is the highest value existing in the core of the fraction of limiting power density (FLPD).

[1] For the 10 CFR 50 Appendix J Type A test, the 25% shall not exceed 15 months.

(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) 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 breaker 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.

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 containment isolation valves specified in Table 3.5.2 shall be tested for automatic closure by an isolation signal during each refueling outage. The following valves are required to close in the time specified below:

| | |
|---|---|
| Main steam line isolation valves | $\geq$ 3 sec & $\leq$ 10 sec |
| Isolation condenser isolation valves | $\leq$ 60 sec |
| Cleanup system isolation valves | $\leq$ 60 sec |
| Cleanup auxiliary pumps system isolation valves | $\leq$ 60 sec |
| Shutdown system isolation valves | $\leq$ 60 sec |

2. Each containment isolation valve shown in Table 3.5.2 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 specified isolating time. Following maintenance, repair or replacement work on the control or power circuit for the valves shown in Table 3.5.2, 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.

(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.5.a can be met.

(4) A drywell to suppression chamber leak rate test (interval not to exceed 20 months) shall demonstrate that with an initial differential 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.

G. Reactor Building

1. Secondary containment capability tests shall be conducted after isolating the reactor building and placing either Standby Gas Treatment System filter train in operation.

2. The tests shall be performed at least once per operating cycle (interval not to exceed 20 months) and shall demonstrate the capability to maintain a ¼ inch of water vacuum under calm wind conditions with a Standby Gas Treatment System Filter train flow rate of not more than 4000 cfm.

3. A secondary containment capability test shall be conducted at each refueling outage prior to refueling.

4. The results of the secondary containment capability tests shall be in the subject of a summary technical report which can be included in the reports specified in Section 6.

H. Standby Gas Treatment System

1. The capability of each Standby Gas Treatment System circuit shall be demonstrated by:

a. At least once per 18 months, after every 720 hours of operation, and following significant painting, fire, or chemical release in the reactor building during operation of the Standby Gas Treatment System by verifying that:

(1) The charcoal absorbers remove $\geq$99% of a halogenated hydrocarbon refrigerant test gas and the HEPA filters remove $\geq$99% of the DOP in a cold DOP test when tested in accordance with ANSI N510-1975.

In addition to the regular sample, snubbers which failed a previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed (if it is repaired and installed in another position) and the replacement snubber shall be retested. The results from testing of these snubbers are not included for determining additional sampling requirements.

For any snubber that fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated. If caused by manufacturer or design deficiency, actions shall be taken to ensure that all snubbers of the same design are not subject to the same defect.

d. Hydraulic Snubbers Functional Test Acceptance Criteria

The hydraulic snubber functional test shall verify that:

1. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

2. Snubber bleed, or release rate, where required, is within the specified range in compression or tension. For snubbers specifically required to not displace under continuous load, the ability of the snubbers to withstand load without displacement shall be verified.

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.

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

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

Although the dose calculations suggest that the allowable test leak rate could be allowed to increase to about 2.0%/day before the guideline thyroid dose limit given in 10 CFR 100 would be exceeded, establishing the limit of 1.0%/day provides an adequate margin of safety to assure the health and safety of the general public. It is further considered that the allowable leak rate should not deviate significantly from the containment design value to take advantage of the design leak-tightness capability of the structure over its service lifetime. Additional margin to maintain the containment in the "as-built" condition is achieved by establishing the allowable operational leak rate. The operational limit is derived by multiplying the allowable test leak rate by 0.75 thereby providing a 25% margin to allow for leakage deterioration which may occur during the period between leak rate tests.

A Primary Containment Leakage Rate Testing Program has been established to implement the requirements of 10 CFR 50, Appendix J, Option B. Guidance for implementation of Option B is contained in NRC Regulatory Guide 1.163,"Performance Based Containment Leak Test Program", Revision 0, dated September 1995. Additional guidance for NRC Regulatory Guide 1.163 is contained in Nuclear Energy Institute (NEI) 94-01, "Industry Guideline for Implementing Performance Based Option of 10 CFR 50, Appendix J", Revision 0, dated July 26, 1995, and ANSI/ANS 56.8-1994, "Containment System Leakage Testing Requirements".

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. 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 is based on the value 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 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

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:

| | |
|---|---|
| Full Closed<br>(Closed to 0.10" open) | 2 Green - On<br>2 Red - Off |
| Open 0.10"<br>(0.10" open to full open) | 2 Green - Off<br>2 Red - Off |

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 1/10th 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. Checking the initiating sensors and associated trip channels demonstrates the capability for automatic actuation. Performing the reactor building in leakage test prior to refueling demonstrates secondary containment capability prior to extensive fuel handling operations associated with the outage. Verifying the efficiency and operation of charcoal filters once per 18 months 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 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 Cleaning Conference.**

High efficiency particulate filters are installed before and after the charcoal filters to minimize potential releases 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 at testing medium.

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

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