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

Subject: Oyster Creek Generating Station (OCGS)
Docket No. 50-219
Facility License No. DPR-16
License Amendment No. 219 Corrected Page

References:

1) License Amendment No. 219 dated January 24, 2001, "Oyster Creek Nuclear Generating Station – Issuance of Amendment Re: Revision to Ventilation Charcoal Adsorber Testing Program (TAC No. MA7804)"

2) Correspondence No. 1940-99-20026 to the USNRC dated March 7, 2000, "Technical Specification Change Request No. 273, Surveillance Frequency of Excess Flow Check Valves (EFCV)"

3) AmerGen letter No. 2130-00-20230 dated September 15, 2000, "Technical Specification Change Request (TSCR) No. 270, Response to Request for Additional Information"

4) License Amendment No. 216 dated October 25, 2000, "Oyster Creek Nuclear Generating Station – Issuance of Amendment Re: Excess Flow Check Valves (TAC No. MA8492)"

5) Correspondence No. 1940-99-20558 dated December 1, 1999, "Technical Specification Change Request No. 270, Testing Protocol for Activated Charcoal in ESF Systems"

The purpose of this letter is to provide the NRC with a corrected page for the Oyster Creek Technical Specifications for reissue of License Amendment No. 219 (Reference 1). Enclosure 1 to this letter contains revised Technical Specification page 4.5-5. A change to Specification 4.5.K on page 4.5-5 was requested in Reference 2, approved as requested in Reference 4, inadvertently changed by AmerGen Energy Company, LLC (AmerGen) in Reference 3 and reissued with the change in Reference 1.

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Oyster Creek Technical Specification Change Requests (TSCR) Nos. 270 (Reference 5) and 273 (Reference 2) were under review by the NRC at the same time. Each TSCR contained an unrelated change on page 4.5-5. In response to a request by the NRC staff regarding TSCR 270, a revision to the original requested change to Specification 4.5.H.1.a(2) was forwarded by Reference 3. That letter contained two versions of page 4.5-5. Enclosure 1 of Reference 3 was to be issued with the license amendment approving TSCR 270 if TSCR 270 was approved before TSCR 273 and Enclosure 2 was to be issued if TSCR 273 was approved before TSCR 270. An error was introduced in the Enclosure 2 version of page 4.5-5 regarding Specification 4.5.K. A proposed change to Specification 4.5.K was under review by the NRC staff in response to TSCR 273. The error involved a rewording of the first sentence of 4.5.K that did not alter the requirement. The error resulted from an incorrect version of TSCR 273 residing in an electronic database and in the hardcopy working file that were used in preparation of Reference 3.

As a result of the change to Specification 4.5.K introduced by Reference 3 as discussed above, License Amendment No. 219 altered the wording of Specification 4.5.K from the wording contained in License Amendment No. 216. As previously mentioned, the altered wording does not change the requirement of Specification 4.5.K. Technical Specification page 4.5-5 contained in Enclosure 1 to this letter corrects the error introduced by License Amendment No. 219.

Enclosure 2 to this letter contains complete Section 4.5 bases for reissue with License Amendment No. 219. It reflects all updates to the bases from prior amendments. Some bases updates were also inadvertently omitted when Reference 1 was issued as these bases pages were submitted with Reference 5 and not updated to reflect subsequent changes at the time License Amendment No. 219 was issued. The bases have been corrected to include changes due to License Amendment Nos. 210 and 211 on page 4.5-12 and License Amendment No. 216 on page 4.5-15.

The prior Oyster Creek process of submitting replacement pages with TSCRs is a significant contributor to the potential for introducing errors into the Technical Specifications as happened with License Amendment No. 219. The process has been changed. Currently, replacement pages are not forwarded with license change applications (TSCRs). They will be provided to the NRC staff when issuance of an amendment is imminent. This will help to ensure that the pages to be issued reflect any changes made subsequent to the submittal of the TSCR.

We regret and apologize for any inconvenience this has caused the NRC.

Should you have any questions or require any additional information please contact Mr. George B. Rombold at 610-765-5516.

Very truly yours,

Ron J. DeGregorio
Vice President
Oyster Creek

Enclosures: Enclosure 1: Replacement Technical Specification Page 4.5-15
Enclosure 2: Replacement Section 4.5 Bases Pages 4.5-10 – 4.5-15

c: H. J. Miller, Administrator, USNRC Region I
L. A. Dudes, USNRC Senior Resident Inspector, Oyster Creek
H. N. Pastis, USNRC Senior Project Manager, Oyster Creek
File No. 99158

Enclosure 1

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Technical Specification Page 4.5-5

(2) Results of laboratory carbon sample analysis show ≥95% radioactive methyl iodide removal efficiency when tested in accordance with ASTM D 3803-1989 (30°C, 95% relative humidity, at least 45.72 feet per minute charcoal bed face velocity).

b. At least once per 18 months by demonstrating:

(1) That the pressure drop across a HEPA filter is equal to or less than the maximum allowable pressure drop indicated in Figure 4.5.1.

(2) The inlet heater is capable of at least 10.9 KW input.

(3) Operation with a total flow within 10% of design flow.

c. At least once per 30 days on a STAGGERED TEST BASIS by operating each circuit for a minimum of 10 hours.

d. Anytime the HEPA filter bank or the charcoal absorbers have been partially or completely replaced, the test per 4.5.H.1.a (as applicable) will be performed prior to returning the system to OPERABLE STATUS.

e. Automatic initiation of each circuit every 18 months.

I. Inerting Surveillance

When an inert atmosphere is required in the primary containment, the oxygen concentration in the primary containment shall be checked at least weekly.

J. Drywell Coating Surveillance

Carbon steel test panels coated with Firebar D shall be placed inside the drywell near the reactor core midplane level. They shall be removed for visual observation and weight loss measurements during the first, second, fourth and eighth refueling outages.

K. Instrument Line Flow Check Valves Surveillance

The capability of a representative sample of instrument line flow check valves to isolate shall be tested at least once per 24 months. In addition, each time an instrument line is returned to service after any condition which could have produced a pressure flow disturbance in that line, the open position of the flow check valve in that line shall be verified. Such conditions include:

Enclosure 2

Oyster Creek Generating Station
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Technical Specification Bases Pages 4.5-10 – 4.5-15

4.5 CONTAINMENT SYSTEM

Bases:

In the event of a loss-of-coolant accident, the peak drywell pressure would be 38 psig, which would rapidly reduce to 20 psig within 100 seconds following the pipe break. The total time the pressure would be above 35 psig is calculated to be about 7 seconds. Following the pipe break, absorption chamber pressure rises to 20 psig within 8 seconds, equalizes with drywell pressure at 25 psig within 60 seconds and thereafter rapidly decays with the drywell pressure decay [1]

The design pressures of the drywe11 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". 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 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 breaker 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 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 every 3 months 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 (Closed to 0.10" open)	2 Green - On 1 Red - Off
Open 0. 10 " (0.10" open to full open)	2 Green - Off 2 Red – On

During each refueling outage, four suppression chamber-drywell vacuum breakers will be inspected to assure components have not deteriorated. Since valve internals are designed for a 40-year lifetime, an inspection program which cycles through all valves in about 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 halogenated hydrocarbon refrigerant, which is injected into the system upstream of the charcoal filters. Measurement of the refrigerant 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.

The 95% methyl iodide removal efficiency is based on the formula in GL 99-02 for allowable penetration [(100% - 90% credited in DBA analysis) divided by a safety factor of 2]. If the allowable penetration is $\leq$5%, the required removal efficiency is $\geq$95%. If laboratory tests for the adsorber material in one circuit of the Standby Gas Treatment System are unacceptable, all adsorber material in that circuit shall be replaced with adsorbent qualified according to Regulatory Guide 1.52. Any HEPA filters found defective shall be replaced with those qualified with Regulatory Position C.3.d of Regulatory Guide 1.52.

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

The snubber inspection frequency is based upon the number of unacceptable snubbers found during the previous inspection, the total population or category size for each snubber type, and the previous inspection interval. A snubber is considered unacceptable if it fails to satisfy the acceptance criteria of the visual inspection. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, that decision must be made and documented before any inspection and used as the basis upon which to determine the next inspection interval for that category.

If continued operation cannot be justified with an unacceptable snubber, the snubber shall be declared inoperable and the applicable action requirements met. To determine the next surveillance interval, the snubber may be reclassified as acceptable if it can be demonstrated that the snubber is operable in its as-found condition by the performance of a functional test and if it satisfies the acceptance criteria for functional testing.

The next visual inspection interval may be twice, the same, or reduced by as much as two-thirds of the previous inspection interval. This interval depends on the number of unacceptable snubbers found in proportion to the size of the population or category for each type of snubber included in the previous inspection. Table 4.5-1 establishes the length of the next visual inspection interval.

To further increase the assurance of snubber reliability, functional tests should be performed once each refueling cycle. These tests will include stroking of the snubbers to verify proper piston movement, lock-up and bleed. Ten percent represents an adequate sample for such tests. Observed failures of these samples require testing of additional units.

After the containment oxygen concentration has been reduced to meet the specification initially, the containment atmosphere is maintained above atmospheric pressure by the primary containment inerting system. This system supplies nitrogen makeup to the containment so that the very slight leakage from the containment is replaced by nitrogen, further reducing the oxygen concentration. In addition, the oxygen concentration is continuously recorded and high oxygen concentration is annunciated. Therefore, a weekly check of oxygen concentration is adequate. This system also provides the capability for determining if there is gross leakage from the containment.

The drywell exterior was coated with Firebar D prior to concrete pouring during construction. The Firebar D separated the drywell steel plate from the concrete. After installation, the drywell liner was heated and expanded to compress the Firebar D to supply a gap between the steel drywell and the concrete. The gap prevents contact of the drywell wall with the concrete, which might cause excessive local stresses during drywell expansion in a loss-of-coolant accident.

The surveillance program is being conducted to demonstrate that the Firebar D will maintain its integrity and not deteriorate throughout plant life. The surveillance frequency is adequate to detect any deterioration tendency of the material. [8]

The operability of the instrument line flow check valves is demonstrated to assure isolation capability for excess flow and to assure the operability of the instrument sensor when required. The representative sample consists of an approximately equal number of EFCVs such that each EFCV is tested at least every 10 years (nominal). In addition, the EFCVs in the sample are representative of the various plant configurations, models, sizes and operating environments. This ensures that any potentially common problem with a specific type or application of EFCV is detected at the earliest possible time. The nominal 10 year interval is based on other performance-based testing programs, such as Inservice Testing (snubbers) and Option B to 10 CFR 50, Appendix J. EFCV test failures will be evaluated to determine if additional testing in that test interval is warranted to ensure overall reliability is maintained. Operating experience has demonstrated that these components are highly reliable and that failures to isolate are very infrequent. Therefore, testing of a representative sample was concluded to be acceptable from a reliability standpoint.[9]

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally changes very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be continually monitored and also observed during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual examination following any event where potentially high loadings could occur provides assurance that no significant damage was encountered. Particular attention should be focused on structural discontinuities in the vicinity of the relief valve discharge since these are expected to be the points of highest stress.

References

(1) Licensing Application, Amendment 32, Question 3
(2) FDSAR, Volume I, Section V-1.1
(3) GE-NE 770-07-1090, "Oyster Creek LOCA Drywell Pressure Response," February 1991
(4) Deleted
(5) FDSAR, Volume I, Sections V-1.5 and V-1.6
(6) FDSAR, Volume I, Sections V-1.6 and XIII-3.4
(7) FDSAR, Volume I, Section XIII-2
(8) Licensing Application, Amendment 11, Question III-18
(9) GE BWROG B21-00658-01, "Excess Flow Check Valve Testing Relaxation," dated November 1998.