GPU NUCLEAR

GPU Nuclear, Inc.
U.S. Route #9 South
Post Office Box 388
Forked River, NJ 08731-0388
Tel 609-971-4000

December 1, 1999
1940-99-20558

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 25555

Gentlemen,

Subject: Oyster Creek Nuclear Generating Station, (OCNGS)
Docket No. 50-219
Technical Specification Change Request No. 270
Testing Protocol for Activated Charcoal in ESF Systems

In accordance with 10 CFR 50.4(b)(1), enclosed is an Oyster Creek Technical Specification Change Request (TSCR) No. 270. The purpose of this TSCR is to revise the standard by which GPU Nuclear tests charcoal used in ESF systems to ASTM D3803-1989. In the case of Oyster Creek, the acceptance criteria for absorption efficiency is also being increased. These changes are being made in accordance with Generic Letter (GL) 99-02. The first test under the new standard, to be conducted by NCS Corporation, will be performed in December, 1999.

Using the standards in 10 CFR 50.92, GPU Nuclear, Inc. has concluded that the proposed change does not constitute a significant hazards consideration, as described in the enclosed analysis performed in accordance with 10 CFR 50.91(a)(1). Also enclosed is a Certificate of Service for this request, certifying service to the chief executives of the township and county in which the facilities are located, as well as the designated official of the state of New Jersey, Bureau of Nuclear Engineering.

If additional information is required, please contact Dennis Kelly of my staff at (609) 971-4246.

Sincerely,

Sander Levin

Sander Levin
Site Director, Oyster Creek

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

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December 1, 1999
1940-99-20558

Mr. Kent Tosch, Director
Bureau of Nuclear Engineering
Department of Environmental Protection
CN 415
Trenton, NJ 08628

Dear Mr. Tosch:

Subject: Oyster Creek Nuclear Generating Station
Operating License No. DPR-16
Technical Specification Change Request No. 270

Enclosed is one copy of the Technical Specification Change Request No. 270 for the Oyster Creek Nuclear Generating Station Operating License.

This document was filed with the U.S. Nuclear Regulatory Commission on December 1, 1999.

Very truly yours,

Sander Levin
Site Director, Oyster Creek

SL/DPK
Enclosure

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December 1, 1999
1940-99-20558

The Honorable William J. Boehm
Mayor of Lacey Township
818 West Lacey Road
Forked River, NJ 08731

Dear Mayor:

Subject: Oyster Creek Nuclear Generating Station
Operating License No. DPR-16
Technical Specification Change Request No. 270

Enclosed is one copy of the Technical Specification Change Request No. 270 for the Oyster Creek Nuclear Generating Station Operating License.

This document was filed with the U.S. Nuclear Regulatory Commission on December 1, 1999.

Very truly yours,

Sander Levin
Site Director, Oyster Creek

SL/DPK
Enclosure

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

GPU NUCLEAR

GPU Nuclear, Inc.
U.S. Route #9 South
Post Office Box 388
Forked River, NJ 08731-0388
Tel 609-971-4000

IN THE MATTER OF)
GPU NUCLEAR, INC.)

DOCKET NO. 50-219

CERTIFICATE OF SERVICE

This is to certify that a copy of Technical Specification Change Request No. 270 for the Oyster Creek Nuclear Generating Station Technical Specifications, filed with the U.S. Nuclear Regulatory Commission on December 1, 1999, has this day of December 1, 1999, been served on the Mayor of Lacey Township, Ocean County, New Jersey by deposit in the U.S. mail, addressed as follows:

The Honorable William J. Boehm
Mayor of Lacey Township
818 West Lacey Road
Forked River, NJ 08731

By: Sander Levin
Sander Levin
Site Director, Oyster Creek

OYSTER CREEK NUCLEAR GENERATING STATION

OPERATING LICENSE
NO. DPR-16

TECHNICAL SPECIFICATION
CHANGE REQUEST NO. 270
DOCKET NO. 50-219

Applicant submits by this Technical Specification Change Request No. 270 to the Oyster Creek Nuclear Generating Station Technical Specifications, modified pages 4.5-4, 4.5-5, 4.5-9, 4.5-10, 4.5-11, 4.5-12, 4.5-13, 4.5-14,and 4.5-15.

By: Sander Levin
Sander Levin
Site Director, Oyster Creek

Sworn to and Subscribed before me this day of 1st day of December, 1999.

Diana M. DeBlasio
Notary Public

DIANA M. DeBLASIO
NOTARY PUBLIC OF NEW JERSEY
My Commission Expires 6/13/2001

I. TECHNICAL SPECIFICATION CHANGE REQUEST (TSCR) NUMBER 270

GPU Nuclear requests that the following replacement pages be inserted into the existing Technical Specifications:

Replace existing pages 4.5-4, 4.5-5 and 4.5-9 through 4.5-15 with the attached replacement pages 4.5-4, 4.5-5 and 4.5-9 through 4.5-15.

II. REASON FOR CHANGE

On June 3, 1999 the NRC issued Generic Letter (GL) 99-02 "Laboratory Testing of Nuclear-Grade Activated Charcoal". The GL established four categories of plants relative to testing protocols and the plant Technical Specifications (TS). Oyster Creek is in category 2, "plants in compliance with their TS that test in accordance with a test protocol other than ASTM D3803-1989". Category 2 plants were requested to submit a Technical Specification change requiring testing in accordance with that protocol or justify an alternative testing method. Oyster Creek currently tests to the ASTM D3803-1979 and, therefore, must change the protocol and modify the Technical Specifications. Oyster Creek has elected to adopt ASTM D3803-1989.

The GL provides a formula for determining allowable penetration which includes a safety factor of 2. The application of this formula will change the acceptance criteria for absorption efficiency. In addition to changing the protocol and the required efficiency, the Bases pages were separated from the specification pages and, consequently, pagination was affected. The term methyl iodine was corrected to methyl iodide and a minor change to the Bases was made on page 4.5-12. The Bases change corrects a misstatement and was previously submitted as part of TSCR 267.

III. SAFETY EVALUATION JUSTIFYING CHANGE

The Standby Gas Treatment System (SGTS) consists of two separate filter trains, each having 100% capacity. Either of the two filter trains is considered as an installed spare, with the remaining one capable of the required flow capacity. Each SGTS train is composed of various elements to filter and remove radioactive iodines and particulates that may be present in the Reactor Building atmosphere during and after an accident. A charcoal filter is provided to remove radioiodines from the Reactor Building exhaust air during an accident condition. The radioiodines in the exhaust air are expected to be 90 percent in the "normal" forms of molecular iodine and iodine salts, and 10 percent in the form of methyl iodide (CH_3I), or other organic salts.

The NRC reviewed the testing of nuclear-grade activated charcoal to standards other than American Society for Testing and Materials (ASTM) D3803-1989, "Standard Test Method for Nuclear-Grade Activated Carbon". The staff determined that testing to any other standard does not provide assurance for complying with the current licensing basis as it relates to the dose limits in General Design Criterion (GDC) 19 of 10 CFR 50.

Oyster Creek has modified its contract with Nuclear Containment Systems, Inc. (NCS) to periodically test charcoal to the requirements of ASTM D3803-1989 rather than ASTM D3803-1979. The first test under the new criteria will be performed in December, 1999. The test will be conducted at 30° C and 95% relative humidity (RH). A radioactive methyl iodide removal rate ≥ 90% is currently required. Applying the formula in GL 99-02 will increase the required removal rate to ≥ 95%. The temperature and RH are typical for the test, and the 90% removal rate used as a factor in the formula, is consistent with the Oyster Creek licensing basis. Actual test results of the methyl iodide removal rate are routinely greater than 99%. If laboratory tests for the adsorber material in one train of the Standby Gas Treatment System are unacceptable, all adsorber material in that circuit is replaced.

IV. NO SIGNIFICANT HAZARDS DETERMINATION

GPU Nuclear has determined that this TSCR poses no significant hazard as defined by 10 CFR 50.92.

1. The proposed change is in accordance with NRC guidance in GL 99-02 which states that the new testing protocol is more accurate and demanding than older tests. The acceptance criteria for charcoal efficiency has been made more stringent and there is no change to an operating parameter of any system, component or structure. Therefore, the probability of occurrence or the consequences of an accident previously evaluated in the SAR will not increase as a result of this change.

2. The proposed change revises the testing standard for activated charcoal efficiency to a more conservative methodology while increasing the acceptance criteria through the application of a safety factor. There is no change to an operating parameter of any system, component or structure. Therefore, the proposed activity does not create the possibility for an accident or malfunction of a different type than any previously identified in the SAR.

3. The proposed change does not involve a reduction in the margin of safety. The change is primarily administrative, adheres to NRC guidance, and is more conservative than the previously employed standard. The change does not modify an operating parameter of any system, component or structure. Therefore, there is no reduction in the margin of safety.

V. IMPLEMENTATION

GPU Nuclear requests that this amendment be effective upon issuance.

(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 **shall be performed once every 24 months** to 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.

(2) Results of laboratory carbon sample analysis show ≥95% radioactive methyl iodide removal efficiency when tested in accordance with ASTM D 3803-1989 (30 degrees C, 95% relative humidity).

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 each in strument line flow check valve to isolate shall be tested at least once in every period between refueling outages. Each time an instrument line is returned to service after any condition which could have

f. Snubber Service Life Monitoring

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

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

N. Secondary Containment Isolation Valves

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

2. At least once per refueling outage, all valves shall be tested for automatic closure by an isolation signal.

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 drywell 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 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". 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 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 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 the 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 (Closed to 0.10" open)	2 Green - On 2 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 ≤ 5%, the required removal efficiency is ≥ 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.

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* D.R. Muhàbier, "In Place Nondestructive Leak 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 are demonstrated to assure isolation capability for excess flow and to assure the operability of the instrument sensor when required.

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