GPU NUCLEAR

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

March 7, 2000
1940-99-20026

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

Gentlemen,

Subject: Oyster Creek Nuclear Generating Station, (OCNGS)
Docket No. 50-219
Technical Specification Change Request No. 273
Surveillance Frequency of Excess Flow Check Valves (EFCV)

In accordance with 10 CFR 50.4(b)(1), enclosed is Oyster Creek Technical Specification Change Request (TSCR) No. 273. The purpose of this TSCR is to modify the surveillance requirements from once per refueling interval for each EFCV to testing a representative sample of EFCVs once per 24 months. The use of the phrase once per 24 months is consistent with the Oyster Creek definition of refueling outage as applied to testing and surveillance. The request to reduce the requirements is based on the extremely small occurrence of failure and the minimal impact such a failure would have.

The submittal includes two versions of page 4.5-5. In response to GL 99-02, TSCR 270 proposed a change to Specification 4.5.H.1.a.(2) to reflect a new test protocol for activated charcoal used in engineered safety systems. TSCR 270 was submitted December 1, 1999 and was under review on the date of this submittal. If TSCR 270 is approved prior to TSCR 273, the version of page 4.5-5 that reflects the use of ASTM D 3803-1989 should be used in the subsequent Amendment. If TSCR 270 is still under review when TSCR 273 is approved, the other version of page 4.5-5 should be included in the subsequent Amendment. In addition, a change to the Bases of Specification 4.5 has been included for information purposes and is not part of the change request.

Using the standards in 10 CFR 50.92, GPU Nuclear, Inc. has concluded that these proposed changes do 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.

A001

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

Sincerely,

Sander Levin
Acting Director, Oyster Creek

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

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March 7, 2000
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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
TSCR No. 273 Surveillance Frequency of Excess Flow Check Valves

Enclosed is one copy of the Technical Specification Change Request No. 273 for the Oyster Creek Nuclear Generating Station Operating License.

This document was filed with the U.S. Nuclear Regulatory Commission on March 7, 2000.

Very truly yours,

Sander Levin

Sander Levin
Acting Director, Oyster Creek

Enclosure

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March 7, 2000
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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
TSCR No. 273 Surveillance Frequency of Excess Flow Check Valves

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This document was filed with the U.S. Nuclear Regulatory Commission on March 7, 2000.

Very truly yours,

Sander Levin

Sander Levin
Acting Director, Oyster Creek

Enclosure

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

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. 273 for the Oyster Creek Nuclear Generating Station Technical Specifications, filed with the U.S. Nuclear Regulatory Commission on March 7, 2000 , has this day of March 7, 2000, 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
Acting Director,
Oyster Creek

OYSTER CREEK NUCLEAR GENERATING STATION

OPERATING LICENSE
NO. DPR-16

TECHNICAL SPECIFICATION
CHANGE REQUEST NO. 273
DOCKET NO. 50-219

Applicant submits by this Technical Specification Change Request No. 273 to the Oyster Creek Nuclear Generating Station Technical Specifications, modified pages 4.5-5, 4.5-6 and 4.5-15.

By: Sander Levin

Sander Levin
Acting Director, Oyster Creek

Sworn to and Subscribed before me this 7th day of March, 2000

George W. Busch
Notary Public of State of New Jersey

GEORGE W. BUSCH
NOTARY PUBLIC OF NEW JERSEY
My Commission Expires Aug. 8, 2000

I. TECHNICAL SPECIFICATION CHANGE REQUEST (TSCR) NUMBER 273

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

Replace existing pages 4.5-5, 4.5-6 and 4.5-15 with the attached replacement pages 4.5-5, 4.5-6 and 4.5-15.

II. REASON FOR CHANGE

Excess flow check valves (EFCV) are utilized in the Oyster Creek containment to limit the release of fluid in the event of an instrument line break. These valves are tested during each refueling outage in accordance with Technical Specification 4.5.K. There are 60 EFCVs at Oyster Creek and testing them requires many man-hours to complete and can be a critical path activity during an outage.

The BWR Owners' Group has developed a strategy of reducing the EFCV testing frequency that is contained in Topical Report B21-00658-01, "Excess Flow Check Valve Testing Relaxation,"dated November 1998. The NRC reviewed the Topical Report in conjunction with the lead plant submittal from the Duane Arnold Energy Center (DAEC) and issued an Amendment to DAEC in December 1999. The EFCVs at Oyster Creek are typical of those described in the Topical Report and have an exceptional record of reliability. Therefore, GPU Nuclear believes that the EFCVs may safely be tested at a reduced frequency. However, since the testing frequency is currently specified in the Technical Specifications, prior NRC approval is required.

III. SAFETY EVALUATION JUSTIFYING CHANGE

Excess flow check valves limit the reactor coolant release following the failure of an instrument line, valve or component on an instrument line. The valves isolate at a given flow and are periodically functionally tested to ensure proper isolation with resulting minimal flow.

Oyster Creek is equipped with 60 EFCVs, each of which is associated with an instrument line. The current technical specification concerns the ability of the valves to isolate for a given flow and also confirms that the valves are open, when required, to ensure proper communication between the instruments and the process fluid. This change will not impact the latter function. Thus, there is no concern associated with the instrument functionality associated with this change. What is affected by this change is the verification of the valve isolation function, which limits the reactor coolant release following the failure of an instrument line.

The approach used to evaluate this issue is consistent with the BWROG Topical Report. The valves have been shown to be highly reliable when considering the historical data associated with the testing of the isolation function. In addition, the radiological consequences have been assessed to ensure the severity of failure is not increased.

The GE Topical Report evaluated the reliability of EFCVs installed at Oyster Creek and other plants. Oyster Creek and three other facilities have installed Chemquip excess flow check valves. The evaluation was based on information covering a 10 year period. In that time frame, Oyster Creek had zero (0) failures and there were a total of two (2) failures at all four plants. Chemquip EFCVs were shown by the BWROG to have a failure rate of 1.78E-7, which was the lowest of the valve manufacturers included in the evaluation.

Although the Topical Report requested data for ten years, Oyster Creek records demonstrate that there has never been a failure of an EFCV to isolate in the thirty year history of Oyster Creek. The valves have a very simple design with the internals consisting of only a spring and a poppet. The EFCVs are located in stagnant lines and are only cycled for testing. These conditions minimize wear and the buildup of corrosion. The data clearly shows that the EFCVs installed at the OCNGS are highly reliable.

This change will not alter the physical design of the plant. The current OC accident analysis does not take credit for the restriction provided by EFCVs. However, the EFCVs, by design, actually restrict flow when they are not shut. This effect is provided by a 3/8 inch restriction on the valve inlet that would still be effective if the majority of the EFCV body failed to maintain a pressure boundary. Additionally, a 1/4 inch restriction is built into the valve outlet. This design provides a flow restriction benefit even if the EFCV internals fail to seal shut.

The radiological consequences of an instrument line break have been evaluated at Oyster Creek. That evaluation does not take credit for the excess flow check valve when assessing the radiological consequences of the accident. The analysis was submitted to the NRC and reviewed in SEP Topic XV-16 "Failure of Small Lines Carrying Primary Coolant Outside Containment – Oyster Creek". The analysis was approved in NUREG 1382 "Safety Evaluation Report related to the full term operating license for Oyster Creek Nuclear Generating Station". The NRC review resulted in a change to the reactor coolant activity making Oyster Creek consistent with the BWR Standard Technical Specifications. The Oyster Creek Full Term Operating License states the following:

Under SEP Topic XV-16, the-staff reviewed the radiological consequences of failure of small lines carrying primary coolant outside the containment. The staff concluded that reactor coolant activity should be maintained within the limits imposed in the BWR Standard Technical Specifications (NUREG-0123). This will ensure that the radiological consequences of an event that results in release of reactor coolant to the environment will be low.

The proposed Technical Specification change does not alter the plant design in any manner. Furthermore, the instrument line break analysis assumptions also remain unchanged. Therefore, there is no impact on the current procedures or accident analysis. As a result, there is no operational concern associated with this change to the Technical Specifications.

An analysis has been performed which indicates that following implementation of this Technical Specification change, the Oyster Creek Nuclear Generating Station will remain in compliance with the RG 1.11 and meet General Design Criteria 55 and 56 for instrument lines penetrating or connected to primary reactor containment.

The phrasing of Specification 4.5.K has been revised from "every period between refueling outages" to "once per 24 months". This change is consistent with definition 1.16 "Refueling Outage" in the Oyster Creek Technical Specifications.

The surveillance interval proposed for the EFCV remains per refueling outage for a representative sample of valves. The representative sample consists of an approximately equal number of EFCVs, such that each EFCV is tested at least every 10 years (nominal). The nominal 10 year interval value is performance-based, justified by the successful history of the testing these valves have had. It is also consistent with other performance-based testing programs, such as Inservice Testing (snubbers) and Option B to 10 CFR 50, Appendix J.

These changes involve no increase in the amounts, and no change in the types, of any effluents that may be released offsite. Since there is no increase in individual or cumulative occupational radiation exposure, this change involves no significant hazards consideration. Accordingly, this change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with this activity.

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 the GE Topical Report "Excess Flow Check Valve Testing Relaxation" which was reviewed and approved by the NRC. The EFCVs installed at Oyster Creek are extremely reliable and the surveillance interval remains the same for a representative sample of the valves. 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 proportion of EFCVs to be tested in each refueling interval. 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, and supported by a GE Topical Report. The NRC has reviewed the Topical Report and approved a lead plant submittal for this issue. 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.

(2) Results of laboratory carbon sample analysis show ≥95% radioactive methyl iodine removal efficiency when tested in accordance with ASTM D 3803-1989 (30°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 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:

(2) Results of laboratory carbon sample analysis show $\geq$90% radioactive methyl iodine removal efficiency when tested in accordance with ASTM D 3803-79 (30°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 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:

Leakage at instrument fittings and valves
Venting an unisolated instrument or instrument line
Flushing or draining an instrument
Installation of a new instrument or instrument line

L. Suppression Chamber Surveillance

1. At least once per day the suppression chamber water level and temperature and pressure suppression system pressure shall be checked.

2. A visual inspection of the suppression chamber interior, including water line regions, shall be made at each major refueling outage.

3. Whenever heat from relief valve operation is being added to the suppression pool, the pool temperature shall be continually monitored and also observed until the heat addition is terminated.

4. Whenever operation of a relief valve is indicated and the suppression pool temperature reaches 160°F or above while the reactor primary coolant system pressure is greater than 180 psig, an external visual examination of the suppression chamber shall be made before resuming normal power operation.

M. Shock Suppressors (Snubbers)

As used in this specification, "type of snubber" shall mean snubbers of the same design and manufacturer, irrespective of capacity.

1. Each snubber shall be demonstrated OPERABLE by performance of the following inspection program:

   a. Visual Inspections

      Snubbers are categorized as inaccessible or accessible during reactor operation. Each of the categories (inaccessible and accessible) may be inspected independently according to the schedule determined by Table 4.5-1. The visual inspection interval for each type of snubber shall be determined based upon the criteria provided in Table 4.5-1.

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. The representative sample consists of an approximately equal number of EFCV's, such that each EFCV is tested at least every 10 years (nominal). 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