GPU Nuclear

GPU Nuclear Corporation
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Writer's Direct Dial Number:

June 19, 1996
6730-96-2201

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

Gentlemen:

Subject: Oyster Creek Nuclear Generating Station
Docket No. 50-219
Technical Specification Change Request No. 242, Revision 1
Implementation of 10 CFR 50 Appendix J, Option B

On February 23, 1996, GPU Nuclear docketed Technical Specification Change Request (TSCR) No. 242. During a subsequent discussion on June 6, 1996, between the Oyster Creek Nuclear Generating Station and the NRR staff, it was decided that a revision to the original request would be appropriate. Therefore, in accordance with 10 CFR 50.4(b)(1), TSCR No. 242, Revision 1 is enclosed. This revision replaces the previously submitted Technical Specification Change Request No. 242 in its entirety.

The proposed change to the Technical Specifications would allow the implementation of 10 CFR 50, Appendix J, Option B. A summary of the proposed changes is included in Enclosure 1. The changed pages are located in Enclosure 2.

This request meets the requirements for submittal as a Cost Beneficial Licensee Action in the areas of both cost and radiation exposure reduction. Oyster Creek is presently licensed to operate for approximately 13 more years. Under the existing Appendix J criteria, this would require four primary containment integrated leak rate tests (PCILRTs). Additionally, there will be seven outages requiring local leak rate testing (LLRTs). Based on the history of leak rate test results, if Option B is approved, only two more PCILRTs would be required, and the scope of each of the seven LLRTs would be decreased by approximately 40 percent. This would result in a cost savings of approximately $6.5 million from the PCILRT savings and $530,000 from the LLRT savings for a total of $7.0 million over the next thirteen years. More importantly, however, the reduction in testing requirements would result in a decrease of 8.2 person-rem to site personnel.

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This Technical Specification Change Request is requested for implementation prior to the upcoming refueling outage. As the next refueling outage is scheduled to commence on September 7, 1996, approval of this submittal is requested by July 31, 1996, to allow sufficient time for planning and scheduling. An implementation date of August 15, 1996 is requested.

The next Type A test is currently scheduled for the upcoming 16R outage. To allow for an orderly transition from the existing reduced pressure Type A leak rate testing to the full pressure Type A testing, a one time schedular extension for the next Type A test is requested. This extension does not impact the safety and health of the public and is justified by several important technical points:

1. The major contributor to the total identified leakage from Primary Containment comes from Type B and C tested components. Only a small portion of the total leakage is detectable solely through Type A testing. This is typical of the information found throughout the nuclear industry. When leakage has been identified during a Type A test, it has been appropriately replaced or repaired before the restart from the outage. A review of the Integrated Leakage Rate Test logs from 1978 to the present identified only one repeated leakpath. This occurred in 1978 and 1980, but has not occurred since. Over the last ten years, three Type A tests have been performed, and all three were within specification.

2. The Type B and C test criteria have been made much more strict. A Technical Specification Change Request in 1989 removed the limit of 12.08 scfh from each type B or C component in the test program. The program was then modified to place more restrictive alert limits on each valve or penetration, thereby identifying timely maintenance to ensure continuous compliance. The program has been successful in improving the identification of the root cause of problem valve leakage. These causes have been addressed by modifying valves, performing careful maintenance to address the cause, or replacing the valve with a different design. The program has been successful in eliminating many problem valves and in improving the overall as found performance of containment.

3. No schedular extension is requested for the Type B and C testing. All requisite tests will be performed in 16R. Therefore, the major contributors to containment leakage will be tested and maintained as appropriate.

This submittal has been evaluated under the criteria of 10 CFR 50.92 and has been determined to contain No Significant Hazards. This determination has been reviewed and concurred by onsite safety review groups.

Also enclosed is a Certificate of Service for this request certifying service to the chief executive of the township in which the facility is located, as well as the designated official of the State of New Jersey, Bureau of Nuclear Engineering.

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

Sander Levin
for Michael B. Roche
Vice President and Director
Oyster Creek

MBR/JJR
Enclosures

cc: Regional Administrator, NRC Region I
Project Manager, NRR
NRC Resident Inspector

GPU NUCLEAR CORPORATION
OYSTER CREEK NUCLEAR GENERATING STATION

Facility Operating
License No. DPR-16

Technical Specification Change Request No. 242, Revision 1
Docket No. 50-219

Applicant submits, by this Technical Specification Change Request No. 242, Revision 1, to the Oyster Creek Nuclear Generating Station Operating License, a change to pages 1.0-5, 3.5-3, 6-16, and the entire Section 4.5.

By Sander Levin
for Michael B. Roche
Vice President and Director
Oyster Creek

Sworn and Subscribed to before me this 19th day of June 1996.

Betty Goodheart
A Notary Public of NJ

BETTY GOODHEART
A Notary Public of New Jersey
My Commission Expires Mar. 22, 1999

GPU Nuclear

GPU Nuclear Corporation
Post Office Box 388
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609 971-4000
Writer's Direct Dial Number

June 19, 1996

The Honorable John C. Parker
Mayor of Lacey Township
818 West Lacey Road
Forked River, NJ 08731

Dear Mayor Parker

Enclosed herewith is one copy of Technical Specification Change Request No. 242, Revision 1, for the Oyster Creek Nuclear Generating Station Operating License

This document was filed with the United States Nuclear Regulatory Commission on June 19, 1996

Sincerely,

for M. B. Roche
Vice President and Director
Oyster Creek

Enclosure
MBR/JJR

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of)
GPU Nuclear Corporation)
Docket No. 50-219)

CERTIFICATE OF SERVICE

This is to certify that a copy of Technical Specification Change Request No. 242, Revision 1, for Oyster Creek Nuclear Generating Station Operating License, filed with the U.S. Nuclear Regulatory Commission on June 19, 1996 has this day of June 19, 1996, been served on the Mayor of Lacey Township, Ocean County, New Jersey by deposit in the United States mail, addressed as follows:

The Honorable John Parker
Mayor of Lacey Township
818 West Lacey Road
Forked River, NJ 08731

By [signature]
for Michael B. Roche
Vice President and Director
Oyster Creek

GPU Nuclear

GPU Nuclear Corporation
Post Office Box 388
Route 9 South
Forked River, New Jersey 08731-0388
609 971-4000
Writer's Direct Dial Number

June 19, 1996

Mr. Kent Tosch, Director
Bureau of Nuclear Engineering
Department of Environmental Protection
CN 411
Trenton, NJ 08625

Dear Mr. Tosch:

Subject: Oyster Creek Nuclear Generating Station
Facility Operating License No. DPR-16
Technical Specification Change Request No. 242, Revision 1

Pursuant to 10 CFR 50.91(b)(1), please find enclosed a copy of the subject document which was filed with the United States Nuclear Regulatory Commission on June 19, 1996

Sincerely,

for M. B. Roche
Vice President and Director
Oyster Creek

Enclosure
MBR/JJR

Technical Specification Change Request No. 242, Revision 1

Enclosure 1

No Significant Hazards Determination

OYSTER CREEK NUCLEAR GENERATING STATION
OPERATING LICENSE NO. DPR-16
DOCKET NO. 50-219
TECHNICAL SPECIFICATION CHANGE REQUEST (TSCR) NO. 242, Revision 1

Applicant hereby requests the Commission to change Facility Operating License No. DPR-16 as discussed below, and pursuant to 10 CFR 50.91, an analysis concerning the determination of no significant hazards consideration is also presented:

I. SECTIONS TO BE CHANGED

1. Table of Contents
2. Footnote to definition 1.24
3. Definition 1.25
4. Section 3.5.A.3.b
5. Section 4.5.
6. Bases for Section 4.5.
7. Section 6.9.3.b.

II. CHANGES REQUESTED

GPU Nuclear requests that the following changed replacement pages be inserted into existing Technical Specifications (T.S.):

Pages: i, ii, iii, 1.0-5, Page 3.5-3, Entire Section 4.5 and the Bases, and Page 6-16.

III. EXTENT OF CHANGE

1. Add a new definition and rekey the Table of Contents.
2. Delete the footnote to Definition 1.24 to reflect the requested frequency change for the Primary Containment Leakage Rate Testing Program.
3. Add a new definition 1.25 to establish P_a.
4. Revise the existing Section 3.5.A.3.b for new Section 4.5 reference.
5. Replace the existing Sections 4.5.A., B., C., D., E., F., G., and I., with the new sections 4.5.A., B., C., and D. Re-letter the remaining sections and renumber the remaining pages in section 4.5.
6. Revise the bases for Section 4.5 to reflect the changes in this request.
7. Revise Section 6.9.3.b to move the reporting requirements to the Primary Containment Program.

IV. DESCRIPTION OF CHANGES

10 CFR 50, Appendix J defines the requirements for Primary Containment Leakage Rate Testing (PCLRT). The testing is divided into three sections: 1) Type A testing, which verifies the ability of the Primary Containment to meet design basis conditions; 2) Type B testing, which verifies Primary Containment penetrations; and 3) Type C testing, which tests valve leakage.

By Final Rule "Primary Containment Leakage Testing for Water Cooled Power Reactors" documented in the Federal Register Vol. 60, No. 186, FR 60 49495, dated September 26, 1995, the USNRC approved an Option B to 10 CFR 50, Appendix J. This rule allows for the scheduling and implementation of the containment leakage rate testing program in accordance with prior performance. The Final Rule included, by reference, Regulatory Guide 1.163, Revision 0, which in turn included by reference NEI 94-01, Revision 0, and ANSI/ANS 56.8-1994. This Technical Specification Change Request is being submitted to implement the programmatic requirements defined in 10 CFR 50 Appendix J, Option B.

The requested change consists of two connected parts: 1) the relocation of PCLRT specific requirements from the plant Technical Specifications to the PCLRT Program; and 2) allowing the testing frequency to be administratively controlled based on prior containment test performance as described in the referenced guidance.

The following is a description of the changes to the Technical Specifications:

1. The Primary Containment Integrated Leak Rate Test (PCILRT) pressure is being raised from the currently approved 20 psig to 35 psig.
2. Relocation of the details of the PCLRT scheduling, performance, and acceptance criteria from the plant Technical Specifications to the PCLRT Program Description,
3. Relocation of the requirements to close Type C valves by normal operation from the Technical Specifications to the PCLRT Program Description.
4. Relocation of the requirements that testing the airlock after test gear removal is not required from the Technical Specifications to the PCLRT Program Description.

5. Relocation of the requirements that Type B seals must be tested after reclosing from the Technical Specifications to the PCLRT Program Description.
6. Relocation of the requirement for conversion of low pressure airlock testing results from the Technical Specifications to the PCLRT Program Description.
7. Except for the MSIVs, Feedwater Check Valves, and Drywell Vent and Purge Valves, relocation of the Type C maximum testing interval from the Technical Specifications to the PCLRT Program Description, and extending it from the current 24 month maximum to a maximum of 60 months based on component performance.
8. Relocation of the MSIV, Feedwater Check Valve, Drywell Vent and Purge Valve Type C testing frequency from the Technical Specifications to the PCLRT Program Description and extending it from the existing 24 months to 30 months.
9. Except for the Drywell Airlock barrel seal and Drywell Airlock electrical penetration, relocating the Type B maximum testing interval from the Technical Specifications to the PCLRT Program Description and extending the maximum interval from the current 24 months to 120 months based on component performance and function.
10. Relocation of the Drywell Airlock barrel seal and Drywell Airlock electrical penetration maximum testing interval from the Technical Specifications to the PCLRT Program Description and extending it from the current 24 months to 30 months.
11. Relocation of the Drywell Airlock testing frequency from the Technical Specifications to the PCLRT Program Description and extending the maximum interval from the current 6 months to 30 months.
12. Changing the Drywell Airlock testing frequency for an opened airlock when containment is required from the current 3 days to 7 days.
13. Relocation of the requirement to provide the NRC with a written report of the results of the PCLRT results from the Technical Specifications to the PCLRT Program Description and changing it from submitting a written report to the NRC to having the data available for onsite inspection.
14. Replacing the Type B and C test pressure from a numeric value to a technical name (i.e. change 35 psig to P_a).
15. Removing the acceptability to test MSIVs at reduced pressure (20 psig).
16. Replacing the MSIV test pressure from a numeric value to a technical name (i.e. 20 psig to P_a).
17. Allowance of the 25% surveillance interval extension for all containment leak rate testing.

V. NO SIGNIFICANT HAZARDS CONSIDERATION

GPU Nuclear has determined that this TSCR involves no significant hazards considerations as defined by NRC in 10 CFR 50.92.

1. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety as previously evaluated in the Safety Analysis Report.

   The proposed change implements Option B of 10 CFR 50, Appendix J on performance based containment leakage testing. The proposed change does not involve a change to the plant design or operation. Therefore, the proposed change does not affect any of the parameters or conditions that contribute to initiation of any of the analyzed accidents or malfunctions. The proposed change does request an allowable extension of containment testing. Therefore, a hypothetical leak could remain undetected for a greater period of time. This slight increase in risk has been determined to be insignificant as:

   <u>Type A Testing</u>

   NUREG 1493 determined that the effect of containment leakage on overall accident risk is small as risk is dominated by accident sequences that result in the failure or bypass of the containment. Industry wide PCILRTs have demonstrated that only a small fraction of the leaks discovered during testing exceeded acceptance criteria, and that the leak rate has been only marginally above the acceptable limit. Only 3% of all leaks can be detected only by PCILRT, therefore, only 3% of the theoretical leaks are affected by the extension to the Type A test interval. Experience at Oyster Creek agrees with the industry wide data in that the majority of the detected leakage from the primary containment is found through Type B and C testing. NUREG 1493 found that these observations, together with the insensitivity of reactor accident risk to the containment leakage rate, demonstrates that increasing the Type A leakage test intervals would have a minimal impact on public risk.

Type B and C Testing

Penetrations are designed to ensure reliability of the containment isolation function. Type B penetrations use a double passive seal (e.g. o-ring, gasket) and Type C penetrations use a double isolation valve design to ensure reliability of the isolation function. Because valves perform the isolation function actively, they are more likely to fail on demand (e.g. failure to completely close on demand). To address this failure mode, Type C valves are subjected to increased design constraints and testing to ensure both acceptable leak rates and stroke times. The proposed change does not alter the installation, operation, operating environment, or testing method of these valves. Therefore, the proposed change does not introduce any new component failure modes, nor does it affect the probability of occurrence of any existing evaluated failure mode.

The failure of any single penetration barrier (isolation valve or passive seal) does not cause penetration failure. Therefore, a double failure would have to occur to cause a failure of the penetration and affect containment. Additionally, the proposed change does not change the acceptance criteria for acceptable leakage testing.

The proposed change does not alter plant design or operation, nor does it alter the allowable maximum leakage rate limit. Thus, the proposed change does not affect the probability of occurrence nor the consequences of any evaluated accident or malfunction of equipment important to safety.

2. Operation of the facility in accordance with the proposed amendment would not create the possibility of an accident or malfunction different from any accident or malfunction previously evaluated.

   The proposed change does not involve a change to the plant design or operation. As a result, the proposed change does not affect any of the parameters or conditions that could contribute to initiation of any accidents. This change only involves the reduction in Type A, B, and C test frequencies, and the Type A test pressure.

Type A Testing

The only changes proposed to the Type A testing are to frequency and test pressure. As the proposed test pressure is greater than the existing test pressure, no new type of accident or malfunction is created, and the increase in pressure provides an additional margin of safety. The increase in surveillance interval cannot introduce any new type of accident or malfunction.

The PCILRT is presently performed at 20 psig. Performance of the PCILRT at P_a (35 psig) will provide a more direct leak rate for analysis. P_a is the design pressure of the torus (the drywell design pressure is 44 psig, but the torus is non isolable from the drywell. Therefore, P_a will not create the possibility of the failure of the torus due to overpressurization. No new accident modes can be created by extending the test intervals. No safety related functions or components are altered as a result of this change. Therefore, no new accident or malfunction different from those evaluated in the Safety Analysis Report can result due to the increase in test pressure or increase in surveillance interval.

Type B and C Testing

The proposed change only deals with the frequency of performing Type B and C testing. It does not change what components are tested or the method of testing. There is no proposed change to the design or operation of the plant. Therefore, no new accident or malfunction different from those evaluated in the Safety Analysis Report can result due to the increase in test pressure or increase in surveillance interval.

3. Operation of the facility in accordance with the proposed amendment would not decrease the margin of safety as defined in the bases of the Technical Specifications.

Type A Testing

Except for the method of defining the test frequency and pressure at which the PCILRT is performed, the methods for performing the actual test are not changed. However, the proposed change can increase the probability that an increase in leakage could go undetected for an extended period of time. NUREG 1493 has determined that under several different accident scenarios, the increased risk of radioactivity release from containment is negligible with the implementation of these proposed changes.

Type B and C Testing

The proposed change only affects the frequency of Type B and C testing. The methods for performing the actual test are not changed. The design or operation of Type B and C components are not changed. The proposed change will result in a longer interval between tests of good performing Type B and C components.

The margin of safety that has the potential of being impacted by the proposed change involves the offsite dose consequences of postulated accidents which are directly related to containment leakage rate. The containment isolation system is designed to limit leakage to L_a, which is defined by the Oyster Creek Technical Specifications to be 1.0 percent by weight of the containment air at 35 psig per 24 hours. The limitation on containment leakage rate is designed to ensure the total leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure (P_a). The margin of safety for the offsite dose consequences of postulated accidents directly related to the containment leakage rate is maintained by meeting the 1.0 L_a acceptance criteria. The L_a value is not being modified by this proposed Technical Specification change request.

Therefore, the margin of safety as defined in the bases for the Technical Specification will not be reduced.