GPU NUCLEAR CORPORATION
OYSTER CREEK NUCLEAR GENERATING STATION

Facility Operating
License No. DPR-16

Technical Specification Change Request No. 242
Docket No. 50-219

Applicant submits, by this Technical Specification Change Request No. 242, 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 Michael B Roche
Michael B. Roche
Vice President and Director
Oyster Creek

Sworn and Subscribed to before me this 23 day of February, 1996.

Geraldine Levin
A Notary Public of NJ

GERALDINE E. LEVIN
NOTARY PUBLIC OF NEW JERSEY
My Commission Expires 6 · 8 · 2000

GPU Nuclear

GPU Nuclear Corporation
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Writer's Direct Dial Number:

February 23, 1996
6730-96-2049

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, for the Oyster Creek Nuclear Generating Station Operating License.

This document was filed with the United States Nuclear Regulatory Commission on February 23, 1996

Sincerely,

M. B. Roche

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, for Oyster Creek Nuclear Generating Station Operating License, filed with the U.S. Nuclear Regulatory Commission on February 23, 1996 has this day of February 23, 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 Michael B Roche
Michael B. Roche
Vice President and Director
Oyster Creek

Technical Specification
Change Request
No. 242

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

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. Footnote to definition 1.24
2. Section 3.5.A.3.b
3. Section 4.5.
4. Bases for Section 4.5.
5. 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: 1.0-5, Page 3.5-3, Entire Section 4.5 and the Bases, and Page 6-16.

III. EXTENT OF CHANGE

1. Revise the footnote to Definition 1.24 to reflect the requested frequency change for the Primary Containment Leakage Rate Testing Program.
2. Revise the existing Section 3.5.A.3.b for new Section 4.5 reference.
3. 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.
4. Revise the bases for Section 4.5 to reflect the changes in this request.
5. 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 specified 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 guidances.

The following is a description of the major changes to the Technical Specifications <u>in accordance with the new requirements</u>:

1. The Primary Containment Integrated Leak Rate Test (PCILRT) pressure is being raised from the currently approved 20 psig to a pressure of 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. <u>Except for the MSIVs</u>, 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 based on component performance.
8. Relocation of the MSIV 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. <u>Except for the Drywell Airlock barrel seal and Drywell Airlock electrical penetration</u>, 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 PCILRT 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 except for the Type C tests with a 75 months testing interval.

The following is a description of the major changes to Oyster Creek Technical Specifications <u>which are exemptions from the new requirements</u>:

1. NEI 94-01 states that acceptable performance history for Type A testing is defined as completion of two consecutive periodic Type A tests where the calculated performance leakage rate was less than 1.0 L_a before extending the Type A testing interval. It further states that in the event where previous Type A tests were performed at reduced pressure, at least one of the two consecutive previous tests shall be performed at peak accident pressure (P_a). Oyster Creek has previously performed Type A testing at reduced pressure (20 psig vs. 35 psig), as allowed by both Option A to 10 CFR 50 Appendix J and the Oyster Creek Technical Specifications. The last two Type A tests passed as found acceptance criteria with leakage less than 1.0 L_a. Oyster Creek will perform future Type A tests at P_a= 35 psig, but requests exemption from the requirement that one of the previous two Type A tests be performed at peak accident pressure in order to extend the PCILRT interval.
2. Regulatory Guide 1.163 does not endorse a maximum testing interval beyond 60 months for Type C testing, but it does allow for a 25% interval extension. This equates to an interval of 75 months. Oyster Creek is presently on a 2 year refueling cycle which would calculate to a three refueling cycle time period of approximately 72 months. GPU Nuclear is requesting that the allowed maximum interval be established at 75 months with no extension rather that 60 months with a 25% extension.
3. Regulatory Guide 1.163 does not recommend implementing a surveillance interval of greater than 30 months for Feedwater isolation valves and Containment vent and purge valves. A 30 month interval requires testing of all of the valves in this group every refueling outage. This is based on an industry wide perception of performance and/or safety significance. However, this ignores site specific valve performance data and plant specific safety significance. An Oyster Creek site specific analysis indicates that some valves may not need this frequency of testing. The requested exemption would allow certain valves with acceptable performance records to be tested every other refueling outage. The criteria to allow a 48 month (two refueling outage) interval includes: 1) at least three cycles of successful testing prior to extending the interval; 2) an engineering evaluation including risk analysis based insights of each specific valve in light of the safety significance in the Oyster Creek Safety Analysis Report; 3) using restrictive administrative limits for valves with a higher safety significance; and 4) restricting the maximum testing frequency for valves

with safety significance not explicitly described in Regulatory Guide 1.163.

4. Allow low pressure testing of the drywell airlock (10 psig) during periods when containment integrity is required. This is a previously approved Technical Specification.

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.

The major changes from the existing Oyster Creek Technical Specifications requested in accordance with the Option B requirements:

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:

   Type A Testing

   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 form 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 form 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.

The major changes from the existing Oyster Creek Technical Specifications requested by exemption to the Option B requirements:

1. Exemption to allow initiation of Option B requirements without a full pressure test during one of the last two PCILRTs.

   The original regulation to allow reduced pressure testing required that two tests be completed prior to initial plant startup. One test had to be done at full accident pressure, and the second one at the reduced pressure, but not less than 50% of the peak accident pressure. Appendix J required that the ratio of the leakage rate tests (Lm_{20} / Lm_{35}) shall be at least 70% in order to allow testing at the reduced pressure. These two tests were performed at Oyster Creek and the ratio was calculated to be 94%. These initial pre-operational tests, and all of the subsequent PCILRTs have been accomplished in full compliance with 10 CFR 50 Appendix J, and the Oyster Creek Technical Specifications.

In order to transition to the Option B requirements, Oyster Creek requests an exemption from the requirement that one of the last two PCILRTs be completed at peak accident pressure. This has been determined to be technically acceptable as:

A. The two original tests in 1969 resulted in a leakage rate ratio that was significantly in excess of the required 70%. Additionally, the allowable leakage rates calculated during a PCILRT are reduced by the ratio of $(P_t / P_a)^{1/2}$, or 76%. These conservativisms could fully justify continuing testing with a reduced pressure. However, Oyster Creek will commit to performing all future PCILRTs at peak accident pressure.

B. The drywell corrosion concern at Oyster Creek has resulted in extensive inspections of the containment structure. Ultrasonic inspections to determine actual drywell wall thickness have been performed at every refueling outage, as well as every outage of sufficient duration, since 1987. These inspections have provided an additional degree of assurance that the drywell structure can fully withstand peak accident pressure.

C. Two on-line methods of determining containment leakage during operation are provided. The first is by monitoring the use of nitrogen makeup in maintaining the nitrogen atmosphere in the drywell. This is done on a daily basis. This can determine a gross leak in the Drywell and initiate an investigation into the excessive leak rate. A second method of determining gross leakage is by the performance of the periodic Torus to Drywell vacuum breaker test. This test monitors the leakage through the vacuum breakers. However, if an external leakage path exists from either the Torus or the Drywell, it will be detected during the surveillance.

Therefore, this exemption request does not involve an increase in the probability of occurrence or consequences of an accident or malfunction of equipment important to safety previously analyzed in the Safety Analysis Report as the validity of the reduced pressure testing was verified by pre-operational testing. This reduced pressure testing was approved by the NRC staff and was included in the Oyster Creek Technical Specifications. Additionally, extensive inspections since 1987 have ensured the continuing acceptability of the Drywell shell structural integrity. Finally, two independent methods for detecting gross leakage online are performed routinely as part of the surveillance testing program.

This exemption request does not create an accident or malfunction of a different type than any previously analyzed in the Safety Analysis Report as the request does not involve any changes to the design or operation of the nuclear plant. The exemption simply requests relief from the test pressure of one of the two previous surveillances and therefore cannot create any new concerns.

This exemption request does not reduce the margin of safety in the bases of the Technical Specifications as the bases of the Technical Specifications deal with offsite release limits. The previous two PCILRTs (as well as all of the PCILRTs before them) were completed in full compliance with Appendix J and the Technical Specifications. Nothing in this request changes the previous data or acceptance criteria.

2. The exemption request to allow a maximum testing interval of 75 months with no 25% extension for applicable Type C tests is acceptable as the Regulatory Guide interval including the 25% extension is also 75 months. NUREG 1493 states that the maximum testing interval could be as long as 10 years.

This exemption request does not involve an increase in the probability of occurrence or consequences of an accident or malfunction of equipment important to safety previously analyzed in the Safety Analysis Report as the testing interval extension to 75 months to any Type C component has already been determined acceptable as documented in Regulatory Guide 1.163 when the surveillance extension is included. The probability of occurrence or consequences of a previously analyzed accident or malfunction of equipment important to safety as previously analyzed in the Safety Analysis Report are not increased.

This exemption request does not create an accident or malfunction of a different type than any previously analyzed in the Safety Analysis Report as the request does not involve any changes to the design or operation of the nuclear plant. The request simply allows for the use of a 75 month interval based on performance rather than allowed interval plus allowed extensions. Scheduling alone cannot create a new accident or malfunction than previously analyzed.

This exemption request does not reduce the margin of safety as described in the bases of the Technical Specifications as the bases of the Technical Specifications deal with offsite release limits. The engineering determination that a component is capable of performing for 75 months without testing does not affect the results of a calculated offsite release. Therefore, this exemption does not reduce the margin of safety as defined in the bases of the Technical Specifications.

3. The exemption to allow feedwater isolation valve and containment vent and purge valve maximum testing frequency to be established only after three cycles of performance and testing and then be limited to a 48 month interval is acceptable as:

   This exemption request does not involve an increase in the probability of occurrence or consequences of an accident or malfunction of equipment important to safety previously analyzed in the Safety Analysis Report as the components which are allowed to extend to the 48 month limit must have completed 3 cycles of acceptable performance and test results. Therefore, only those components which have demonstrated the ability to complete more than a 48 month interval while maintaining acceptable performance characteristics will be extended. This ensures that any problem components can never go to the 48 month interval. Therefore, no increase in probability of occurrence or consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report can occur.

   This exemption request does not create an accident or malfunction of a different type than any previously analyzed in the Safety Analysis Report as the request does not involve any changes to the design or operation of the nuclear plant. A new accident or malfunction cannot be created simply by extending the interval of testing. Therefore, no new accident or malfunction is created by this exemption.

   This exemption request does not reduce the margin of safety in the bases of the Technical Specifications as the bases of the Technical Specifications deal with offsite release limits. The selection of only those valves which have demonstrated good performance for increased intervals does not affect offsite release limits.

4. The exemption request to allow the use of a 10 psig Drywell airlock pressure test during periods when containment integrity is required is an existing approved Technical Specification. Therefore, no further evaluation is required.