GPU NUCLEAR CORPORATION
OYSTER CREEK NUCLEAR GENERATING STATION

Provisional Operating
License No. DPR-16

Technical Specification
Change Request No. 106

Docket 50-219

Applicant submits by this Technical Specification Change Request No. 106 to the Oyster Creek Nuclear Generating Station Technical Specifications, the addition of a note to Section 1.12 of Appendix A allowing that the time between successive tests or surveillances shall not exceed 30 months between the cycle 9 and cycle 10 refueling outages.

BY _____
P. B. Fiedler
Vice President and
Director - Oyster Creek

STATE OF NEW JERSEY)
)
COUNTY OF MORRIS)

Sworn and subscribed to before me this 28th day of May, 1982.

_____
Notary Public

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF	)	
	)	DOCKET NO. 50-219
GPU NUCLEAR CORPORATION	)	

CERTIFICATE OF SERVICE

This is to certify that a copy of Technical Specification Change Request No. 106 for the Oyster Creek Nuclear Generating Station Technical Specifications, filed with the United States Nuclear Regulatory Commission on May 28, 1982, has this 28th day of May been served on the Mayor of Lacey Township, Ocean County, New Jersey by deposit in the United States mail, addressed as follows:

The Honorable Theodore Hutler
Mayor of Lacey Township
818 West Laey Road
Forked River, N.J. 08731

GPU NUCLEAR CORPORATION

BY _____
P. B. Fiedler
Vice President and
Director - Oyster Creek

DATED:

GPU Nuclear

GPU Nuclear
100 Interpace Parkway
Parsippany, New Jersey 07054
201 263-6500
TELEX 136-482
Writer's Direct Dial Number:

May 28, 1982

The Honorable Theodore Hutler
Mayor of Lacey Township
818 West Lacey Road
Forked River, N. J. 08731

Dear Mayor Hutler:

Enclosed is one copy of Technical Specification Change Request No. 106 for the Oyster Creek Nuclear Generating Station Operating License.

This document was filed with the United States Nuclear Regulatory Commission on May 28, 1982.

Very truly yours,

P. B. Fiedler
Vice President and
Director - Oyster Creek

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Enclosure

GPU NUCLEAR CORPORATION
OYSTER CREEK NUCLEAR GENERATING STATION
(DOCKET NO. 50-219)
PROVISIONAL OPERATING LICENSE NO. DPR-16

Appplicant hereby requests the Commission to change Appendix A to the Oyster Creek Technical Specifications as follows:

1. Section to be Changed

   a. Section 1.12 definition

2. Extent of Change

   The definition of refueling outage shall be annotated to extend the time between successive tests or surveillances from not exceeding 20 months to not exceeding 30 months.

   Surveillances included in this Change Request are:

   A) Inspection of Four Torus to Drywell Vacuum Breakers
   B) Replacement of Five Safety Valves
   C) Inspection of the Core Spray Sparger
   D) Visual Inspection of the Interior of the Torus

3. Discussion

   Attached are Safety Evaluations for each surveillance noted above:

   A. Pressure Suppression Chamber Vacuum Breakers

   The extention of the surveillance interval will not affect the operability of the vacuum breakers. The vacuum breakers are operability tested once each month and following any release of energy which would tend to increase pressure to the suppression chamber per Oyster Creek Technical Specification 4.5.I.5.a. In addition, operation of position switches, indicators and alarms are verified by the operability test. This monthly operability test will give assurance that the vacuum breakers are able to perform their safety function during the extension of the surveillance interval.

   B. Safety Valves

   The extention of the surveillance interval will not affect the safety function of the main steam safety valves. The valves will perform as required to relieve pressure to prevent the reactor pressure from exceeding the safety limit of 1375 psig. The extention of the surveillance interval will not affect the operability of the valves, the valves will open in response to system pressure. What may be affected is the pressure at which the valves will open. During operation of the plant, the setpoint of the valves will drift. This drift is caused in part by the relaxation of the spring at the high operating temperatures.

The drift is in the downward direction. Recent testing of 16 main steam safety valves from Nine Mile Point (same type of valve as used at Oyster Creek) at Wyle Labs indicates that 4 valves were within proper setpoint tolerance and 12 valves were out of tolerance. The valves which were out of tolerance had "as-found" setpoints 20-50 psi below required setpoints. Experience with similar type valves at TMI-1 indicate that when valves are out of setpoint tolerance, the setpoint has drifted downward.

C. Core Spray Sparger Inspection

A Repair Proposal and Safety Evaluation was presented to the NRC on April 1, 1980. One part of this evaluation looked at the adequacy of a core spray sparger with cracks for continued operation.

The conclusions based on the indepth analysis were in part:

1. Extensive additional cracking is not likely in the next cycle of operation.

2. Existing cracks and new cracks, should there be any, are not likely to propagate or open very much during normal plant operation.

3. The only significant operational loads that can be postulated are those associated with initiation of the core spray system and even under these loads, significant crack initiation or growth is not expected.

4. For reasonable assumptions on the existing crack size and propagation, analyses demonstrates that the design nominal flow and distribution characteristics of the core spray system can be maintained within acceptable limits. Margin remains for new cracks.

5. The clamp assemblies at all significant crack locations provide additional structural capability to the sparger and minimize the chances for geometry changes significant to the hydraulic performance.

6. Crack flow is unlikely to significantly affect the spray distribution.

Based on the analysis, there is a high probability that additional cracking if any during the requested surveillance extension time will not be of significance to affect the intended function of the core spray system.

D. Inspection of the Interior of the Torus

Section 4.5.P.2 of the Oyster Creek Technical Specifications require that a visual inspection of the suppression chamber interior, including water line regions be made at each major refueling outage.

Past inspections resulted in finding loose baffles in the early seventies which were removed at that time. The remaining baffles, although not loose were removed in 1977. In addition, general corrosion

was noted on the shell and corrective action was taken to remove the corrosion and build-up the area with welding. The corrosion problem was not considered significant.

Based on past inspection history of not finding any significant problems, the requested surveillance extension will not create a condition adverse to the health and safety of the public.

1.8 PLACE IN SHUTDOWN CONDITION

Proceed with and maintain an uninterrupted normal plant shutdown operation until the shutdown condition is met.

1.9 PLACE IN COLD SHUTDOWN CONDITION

Proceed with and maintain an uninterrupted normal plant shutdown operation until the cold shutdown condition is met.

1.10 PLACE IN ISOLATED CONDITION

Proceed with and maintain an uninterrupted normal isolation of the reactor from the turbine condenser system including closure of the main steam isolation valves.

1.11 REFUEL MODE

The reactor is in the refuel mode when the reactor mode switch is in the refuel mode position and there is fuel in the reactor vessel. In this mode the refueling platform interlocks are in operation.

1.12 REFUELING OUTAGE

For the purpose of designating frequency of testing and surveillance, a refueling outage shall mean a regularly scheduled refueling outage; however, where such outages occur within 8 months of the end of the previous refueling outage, the test or surveillance need not be performed unitl the next regularly scheduled outage. Following the first refueling outage, the time between successive tests or surveillance shall not exceed 20 months.*

1.13 PRIMARY CONTAINMENT INTEGRITY

Primary containment integrity means that the drywell and adsorption chamber are closed and all of the following conditions are satisfied:

A. All non-automatic primary containment isolation valves which are not required to be open for plant operation are closed.

B. At least one door in the airlock is closed and sealed.

C. All automatic containment isolation valves specified in Table 3.5.2 are operable or are secured in the closed position.

D. All blind flanges and manways are closed.

1.14 SECONDARY CONTAINMENT INTEGRITY

Secondary containment integrity means that the reactor buidling is closed and the following conditions are met:

* The time between successive tests or surveillances shall not exceed 30 months prior to the cycle 10 refueling outage only.