AmerGen
An Exelon/British Energy Company

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10 CFR 50.90

September 11, 2001
2130-01-20174

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

Subject: Oyster Creek Generating Station
Facility Operating License No. DPR-16
Docket No. 50-219
Technical Specification Change Request No. 298 - Refueling Interlocks

In accordance with 10 CFR 50.4(b)(1), enclosed is Technical Specification Change Request No. 298.

The purpose of this Technical Specification Change Request is to revise Oyster Creek Technical Specification 3.9 to incorporate compensatory provisions, which permit fuel-handling operations without the refueling interlocks operable. The proposed change is described in Enclosure 1 and is similar to the change previously approved by NRC for the Vermont Yankee Nuclear Power Station in Amendment No. 200, issued April 20, 2001. A mark-up of Technical Specification (TS) page 3.9-1 showing the requested change is contained in Enclosure 2. Corresponding changes to the Bases of Specification 3.9 are also included in Enclosure 2. Replacement TS pages reflecting the requested change will be provided to the NRC prior to the issuance of the license amendment.

Using the standards in 10 CFR 50.92, AmerGen Energy Company, LLC (AmerGen) 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). Pursuant to 10 CFR 50.91(b)(1), a copy of this Technical Specification Change Request is provided to the designated official of the State of New Jersey, Bureau of Nuclear Engineering, as well as the Chief Executive of the township in which the facility is located.

This proposed change to the Technical Specifications has undergone a safety review in accordance with Section 6.5 of the Oyster Creek Technical Specifications. No new regulatory commitments are established by this submittal.

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NRC approval of this change is requested by September 10, 2002. If any additional information is needed, please contact David J. Distel at (610) 765-5517.

Very truly yours,

Ron J. DeGregorio
Vice President - Oyster Creek

RJD/djd

Enclosures: (1) Oyster Creek Technical Specification Change Request No. 298 Safety Evaluation and No Significant Hazards Consideration
(2) Affected Oyster Creek Technical Specification Pages

c: H. J. Miller, Administrator, USNRC Region I
H. N. Pastis, USNRC Senior Project Manager, Oyster Creek
L. A. Dudes, USNRC Senior Resident Inspector, Oyster Creek
File No. 01075

United States of America
Nuclear Regulatory Commission

In the Matter of)

AmerGen Energy Company, LLC) Docket No. 50-219

Certificate of Service

This is to certify that a copy of Technical Specification Change Request No. 298 for the Oyster Creek Generating Station Operating License, filed with the U.S. Nuclear Regulatory Commission on September 11, 2001, has this 11th day of September 2001 been served on the State of New Jersey Bureau of Nuclear Engineering, as well as the Chief Executive of the township in which the facility is located, by deposit in the United States mail, addressed as follows:

The Honorable Ronald Sterling
Mayor of Lacey Township
818 West Lacey Road
Forked River, NJ 08731

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

By: ______________________
Ron J. DeGregorio
Vice President – Oyster Creek

Oyster Creek Generating Station

Facility Operating License
No. DPR-16

Technical Specification Change
Request No. 298
Docket No. 50-219

Applicant submits by this Technical Specification Change Request No. 298 to the Oyster Creek Generating Station Operating License a change to Specification 3.9. All statements contained in this submittal have been reviewed, and all such statements made and matters set forth therein are true and correct to the best of my knowledge.

By: ______________________
Ron J. DeGregorio
Vice President – Oyster Creek

Sworn to and subscribed before me this 11th day of September 2001.

MARITA ZAREMBA
NOTARY PUBLIC OF NEW JERSEY
Commission Expires 5/31/2006

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Notary Public

ENCLOSURE 1

Oyster Creek Technical Specification Change Request No. 298

Safety Evaluation

And

No Significant Hazards Determination

I. Technical Specification Change Request No. 298

AmerGen Energy Company, LLC (AmerGen) requests that the following changed replacement pages be inserted into the existing Technical Specifications:

Revised Technical Specification Pages: 3.9-1, 3.9-2, and 3.9-3

Marked up pages showing the requested changes are provided in Enclosure 2.

II. Reason for Change

The purpose of this Technical Specification Change Request is to revise Oyster Creek Technical Specification 3.9.C to incorporate compensatory provisions to permit fuel-handling operations to be performed with the head off the reactor vessel and the refueling interlocks inoperable. The proposed change would require verification that all control rods are fully inserted and control rod withdrawal is disabled prior to fuel movement with the refueling interlocks inoperable. This change provides enhanced operational flexibility while moving fuel to and from the reactor vessel.

Technical Specification 3.9 Bases is also revised to reflect the above Technical Specification change. Technical Specification 3.9 Bases contained on page 3.9-2 are being relocated to Bases page 3.9-3. This relocation of the Bases is a purely administrative change.

III. Safety Evaluation Justifying Change

The proposed change involves the refueling interlocks and their intended functions, which is to restrict fuel handling operations such that there is assurance that inadvertent criticality does not occur. When the reactor mode switch is in the refuel position, the refueling interlocks prevent the refueling platform from being moved over the core if a control rod is withdrawn and fuel is on the hoist. Therefore, the refueling interlocks prevent criticality during fuel handling operations by preventing the loading of fuel into the core with any control rod withdrawn, or by preventing withdrawal of a control rod from the core during fuel handling operation over the core by inserting a control rod block. This proposed change will permit fuel handling operations in the refuel mode with the head off the reactor vessel and the refueling interlocks inoperable provided all control rods are verified to be fully inserted and control rod withdrawal has been disabled prior to fuel movement. With all control rods inserted and control rod withdrawal disabled, Technical Specification 3.2.A is met and inadvertent criticality due to fuel handling cannot occur.

The proposed change will continue to ensure against inadvertent criticality via the refuel interlocks or through appropriate alternative actions and provides an equivalent level of assurance that fuel will not be loaded into a core cell with a control rod withdrawn. Prior to fuel movement with the refueling interlocks inoperable, administrative controls will require verification that all control rods are fully inserted and control rod withdrawal is disabled. Therefore, the proposed change does not adversely affect nuclear safety or safe plant operations.

IV. No Significant Hazards Determination

AmerGen has determined that this Technical Specification Change Request poses no significant hazards considerations as defined by 10 CFR 50.92.

1. Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change involves refueling interlock operability requirements during refueling operations. The only design basis accident described in the Oyster Creek Updated Final Safety Analysis Report (UFSAR) for cold shutdown or refueling conditions is a postulated fuel handling (dropped bundle) accident. The refueling interlocks are not involved in the mitigation or prevention of a fuel handling accident as previously evaluated. The proposed change does not effect the safety function of the refueling interlocks since alternative specified actions provide an equivalent level of protection against inadvertent criticality during fuel handling operations.

   Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant equipment or to the status of the reactor core during refueling. The Technical Specifications will ensure either through the refueling interlocks or the proposed alternative, that all control rods remain fully inserted and cannot be withdrawn as control rod movement is disabled. This will ensure that fuel is not loaded into the core when a control rod is withdrawn.

Therefore, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

   The proposed change will continue to ensure against inadvertent criticality during fuel handling operations. This is achieved by physical interlocks or by Technical Specification restrictions on fuel handling operations which will prevent fuel from being loaded into a core cell void of a control rod. This is accomplished by preventing fuel from being loaded into the vessel when a control rod is withdrawn and by blocking control rod withdrawal whenever fuel is being loaded into the reactor vessel.

   Therefore, this change does not involve a significant reduction in a margin of safety.

V. Information Supporting an Environmental Assessment

10 CFR 51.22 (c)(9) provides criteria for and identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration, (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (3) result in a significant increase in individual or cumulative occupational radiation exposure.

AmerGen has reviewed this license amendment and has determined that it meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22 (c)(9). Pursuant to 10 CFR 51.22 (c), no environmental impact statement or environmental assessment needs to be prepared in connection with the issuance of the proposed license amendment. The basis for this determination is as follows:

1. The proposed license amendment does not involve a significant hazards consideration as described in Item IV of this evaluation.

2. The proposed license amendment will not result in a significant change in the types or increase in the amounts of any effluents that may be released offsite. The proposed amendment continues to ensure the prevention of inadvertent criticality

during fuel handling operations. The changes do not modify the reactor coolant pressure boundary, containment integrity, nor make any physical changes to the facility design, material, or construction standards.

3. The proposed license amendment will not result in a significant increase in individual or cumulative occupational radiation exposure. The consequences of any design basis accident are not affected by this change. The proposed changes do not affect the integrity of the reactor coolant pressure boundary or any fission product barrier. Occupational exposures are not affected by the proposed changes.

VI. Conclusion

The proposed change has been reviewed in accordance with Section 6.5 of the Oyster Creek Technical Specifications, and it has been concluded that this change requires NRC approval. As discussed above, using the standards in 10 CFR 50.92, AmerGen has determined that there are no significant hazards involved with the proposed change.

AmerGen requests that the amendment authorizing this change be effective immediately upon issuance and implemented within 30 days of issuance.

ENCLOSURE 2

Oyster Creek Technical Specification Change Request No. 298

Affected Technical Specification Pages

Fuel Handling operations with the head off the reactor vessel can be performed with the refueling interlocks inoperable provided all the following specifications are satisfied:
1. All control rods are verified to be fully inserted.
2. Control rod withdrawal has been disabled.

3.9 REFUELING

Applicability: Applies to fuel handling operations during refueling.

Objective: To assure that criticality does not occur during refueling.

Specification:

A. Fuel shall not be loaded into a reactor core cell unless the control rod in that core cell is fully inserted.

B. During core alterations the reactor mode switch shall be locked in the REFUEL position.

C. The refueling interlocks shall be operable with the fuel grapple hoist loaded switch set at $\leq$485 lb. during the fuel handling operations with the head off the reactor vessel. If the frame-mounted auxiliary hoist, the trolley-mounted auxiliary hoist or the service platform hoist is to be used for handling fuel with the head off the reactor vessel the load limit switch on the hoist to be used shall be set at $\leq$400 lb.

D. During core alterations the source range monitor nearest the alteration shall be operable.

E. Removal of one control rod or rod drive mechanism may be performed provided that all the following specifications are satisfied.

1. The reactor mode switch is locked in the refuel position.

2. At least two (2) sources range monitor (SRM) channels shall be operable and inserted to the normal operation level. One of the operable SRM channel detectors shall be located in the core quadrant where the control rod is being removed and one shall be located in an adjacent quadrant.

F. Removal of any number of control rods or rod drive mechanisms may be performed provided all the following specifications are satisfied:

1. The reactor mode switch is locked in the refuel position and all refueling interlocks are operable as required in Specification 3.9.C. The refueling interlocks associated with the control rods being withdrawn may be bypassed as required after the fuel assemblies have been removed from the core cell surrounding the control rods as specified in 4, below.

2. At least two (2) source range monitor (SRM) channels shall be operable and inserted to the normal operation level. One of the operable SRM channel detectors shall be located in the core quadrant where a control rod is

being removed and one shall be located in an adjacent quadrant.

3. All other control rods are fully inserted with the exception of one rod which may be partially withdrawn not more than two notches to perform refueling interlock surveillance.

4. The four fuel assemblies are removed from the core cell surrounding each control rod or rod drive mechanism to be removed.

5. The SHUTDOWN MARGIN requirements of Specification 3.2.A are met.

6. An evaluation will be conducted for each refuel/reload to ensure that actual core criticality of the proposed order of defueling and refueling is bounded by previous analysis performed to support such defueling and refueling activities, otherwise a new analysis shall be performed.

   The new analysis must show that sufficient conservatism exists for the proposed order of defueling and refueling before such operation shall be allowed to proceed.

G. With any of the above requirements not met, cease core alterations or control rod removal as appropriate, and initiate action to satisfy the above requirements.

Basis: During refueling operations, the reactivity potential of the core is being altered. It is necessary to require certain interlocks and restrict certain refueling procedures such that there is assurance that inadvertent criticality does not occur.

START BASIS SECTION ON NEXT PAGE

Addition of large amounts of reactivity to the core is prevented by operating procedures, which are in turn backed up by refueling interlocks (1) on rod withdrawal and movement of the refueling platform. When the mode switch is in the "Refuel" position, interlocks prevent the refueling platform from being moved over the core if a control rod is withdrawn and fuel is on a hoist. Likewise, if the refueling platform is over the core with fuel on a hoist control rod motion is blocked by the interlocks. With the mode switch in the refuel position only one control rod can be withdrawn (1,2).

The one rod withdrawal interlock may be bypassed in order to allow multiple control rod removal for repair, modifications, or core unloading. The requirements for simultaneous removal of more than one Control rod are more stringent than the requirements for removal of a single control rod, since in the latter case Specification 3.2.A assures that the core will remain subcritical.

Fuel handling is normally conducted with the fuel grapple hoist. The total load on this hoist when the interlock is required consists of the weight of the fuel grapple and the fuel assembly. This total is approximately 773 lbs. in the extended position in comparison to the load limit of 485 lbs. Provisions have also been made to allow fuel handling with either of the three auxiliary hoists and still maintain the refueling interlocks. The 400 lb load trip setting on these hoists is adequate to trip the interlock when one of the more than 600 lb. fuel bundles is being handled.

The source range monitors provide neutron flux monitoring capabilities with the reactor is in the refueling and shutdown modes (3). Specification 3.9.D assures that the neutron flux is monitored as close as possible to the location where fuel or controls are being moved. Specifications 3.9.E and F require the operability of at least two source range monitors when control rods are to be removed.

REFERENCES:

(1) FDSAR, Volume I, Section VII-7.2.5
(2) FDSAR, Volume I, Section XIII-2.2
(3) FDSAR, Volume I, Section VII-4.2.2 and VII-4.3.1

The refueling interlocks may be inoperable provided that all 137 control rods are verified to be fully inserted and control rod withdrawal has been disabled prior to commencing or recommencing fuel handling operations with the head off the reactor vessel. This will ensure that all control rods remain fully inserted during fuel handling operations with the head off the reactor vessel. Therefore, Specification 3.2.A is met and the core will remain subcritical during fuel handling operations.