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AmerGen
An Exelon/British Energy Company

10 CFR 50.90

September 11, 2001
2130-01-20167

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

Subject: Oyster Creek Generating Station
Facility License No. DPR-16
Docket No. 50-219
Technical Specification Change Request No. 294 – Cycle Specific Safety Limit MCPR

In accordance with 10 CFR 50.4(b)(1), Enclosure 1 contains Technical Specification Change Request No. 294.

The purpose of this Technical Specification Change Request is to revise the Oyster Creek Technical Specification 2.1.A to delete the cycle-specific footnote for the Safety Limit Minimum Critical Power Ratio (SLMCPR), which is no longer necessary. A mark-up of Technical Specification page 2.1-1 showing the requested change is contained in Enclosure 2. Corresponding changes to the Bases of Specification 2.1 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. 294 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. 01073

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. 294 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. 294
Docket No. 50-219

Applicant submits by this Technical Specification Change Request No. 294 to the Oyster Creek Generating Station Operating License a change to Specification 2.1.A. 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 5/31/2005

Marita Zaremba
____
Notary Public

ENCLOSURE 1

Oyster Creek Technical Specification Change Request No. 294

Safety Evaluation

And

No Significant Hazards Determination

I. Technical Specification Change Request No. 294

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

Revised Technical Specification Pages: 2.1-1, 2.1-2, and 2.1-3

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

II. Reason for Change

The proposed Technical Specification change will revise Technical Specification 2.1.A to delete the cycle-specific footnote for the Safety Limit Minimum Critical Power Ratio (SLMCPR) contained in Technical Specification 2.1.A. This footnote specifies that the Technical Specification SLMCPR value is applicable only for the current operating cycle. Thus NRC approval of the SLMCPR value must be obtained via a Technical Specification amendment each operating cycle even if the SLMCPR limit does not change. This requirement was instituted in the Oyster Creek Technical Specifications in Amendment No. 192, dated August 26, 1997, because the NRC had not approved the General Electric methodology utilized for calculating the SLMCPR limits. NRC approval of this methodology has since been issued.

The Oyster Creek cycle specific SLMCPR limit is currently calculated using NRC approved methodology described in "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-13 (GESTAR-II), and Amendment 25 to NEDE- 24011-P-A (herein after referred to as Amendment 25). The NRC evaluation approving the Amendment 25 methodology is contained in a letter from the NRC to General Electric dated March 11, 1999. Since NRC has approved this methodology, future operating cycle SLMCPR limits determined in accordance with Amendment 25 will not need prior NRC approval for each cycle unless the SLMCPR value changes. The footnote associated with Technical Specification 2.1.A is no longer necessary due to the approval of Amendment 25. Therefore, the note is being deleted.

Reference 1 listed in the Technical Specification 2.1 Bases is being updated to remove the reference to Revision 11 of NEDE-24011-P-A and incorporate reference to the latest approved version as specified in the Core Operating Limits Report (COLR).

Technical Specification 2.1 Bases contained on page 2.1-1 are being relocated to Bases page 2.1-2. This relocation of the Bases is a purely administrative change and results in repagination of pages 2.1-2 and 2.1-3.

III. Safety Evaluation Justifying Change

The proposed change involves deleting the cycle-specific footnote for the Safety Limit Minimum Critical Power Ratio (SLMCPR) contained in Oyster Creek Technical Specification 2.1.A.

The existing Oyster Creek Cycle 18 SLMCPR has been determined in accordance with NRC approved methodology described in "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-13 (GESTAR-II), and Amendment 25. Amendment 25 provides the methodology for determining the cycle-specific MCPR safety limits that replaced the former generic fuel type dependent values. The NRC safety evaluation approving Amendment 25 is contained in a letter from the NRC to General Electric, dated March 11, 1999 (F. Akstulewicz (NRC) to G. A. Watford (GE), "Acceptance for Referencing of Licensing Topical Reports NEDC-32601P, *Methodology and Uncertainties for Safety Limit MCPR Evaluations*; NEDC-32694P, *Power Distribution Uncertainties for Safety Limit MCPR Evaluation*; and Amendment 25 to NEDE-24011-P-A on Cycle-Specific Safety Limit MCPR," (TAC Nos. M97490, M99069 and M97491, dated March 11, 1999). General Electric has incorporated Amendment 25 into NEDE-24011-P-A-14. Amendment 25 is currently used for determining the Oyster Creek cycle specific SLMCPR limits. Since NRC has approved this methodology, future Oyster Creek SLMCPR limits determined in accordance with Amendment 25 will not need prior NRC approval for each cycle unless the value changes.

Prior to the above referenced March 11, 1999 NRC evaluation, Amendment 25 was not approved for generic use at each plant, but was approved on a cycle-by-cycle basis. Therefore, a footnote was added to Oyster Creek Technical Specification 2.1.A to specify that the approval of the SLMCPR value was applicable only for the specific cycle. As a result of the NRC approval of Amendment 25, the footnote to Technical Specification 2.1.A should be deleted. Cycle-specific SLMCPR values will continue to be developed in accordance with NRC approved methods. Technical Specification 2.1 Bases is being updated to remove reference to Revision 11 of NEDE-24011-P-A and incorporate reference to the latest approved version as specified in the COLR. This change provides a direct reference to the NRC approved methodology utilized for developing core operating limits including the SLMCPR limit. Use of these approved methods ensures that greater than 99.9% of the fuel rods in the core will not experience boiling transition if the limit is not violated, and thus satisfies the requirements of General Design Criterion 10 of Appendix A to 10 CFR Part 50 regarding acceptable fuel design limits. A change to the SLMCPR value specified in Technical Specification 2.1.A will require prior NRC approval.

The proposed change is considered administrative and 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 derivation of the cycle specific SLMCPR limit for incorporation into the Technical Specification, and its use to determine cycle specific thermal limits, has been performed using the methodology discussed in "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-13, and Amendment 25. Amendment 25 was approved by the NRC in a Safety Evaluation Report dated March 11, 1999. The footnote to Technical Specification 2.1.A is being deleted. The footnote associated with Technical Specification 2.1.A was originally included to ensure that the SLMCPR value was only applicable for the identified cycle because Amendment 25 was not yet NRC approved. Amendment 25 has subsequently been approved. Therefore, this footnote is no longer necessary. The footnote was for information only, and has no impact on the design or operation of the plant. Cycle-specific SLMCPR values will continue to be developed in accordance with NRC approved methods, which ensures that applicable regulatory requirements are met. The deletion of the footnote associated with Technical Specification 2.1.A is an administrative change.

   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 deletes the footnote contained in Technical Specification 2.1.A as the result of the NRC approval of Amendment 25 to NEDE-24011-P-A. This change does not affect the design or operation of any plant structures, systems, or components. Cycle-specific SLMCPR values will continue to be developed in accordance with NRC approved methods, which ensures that applicable regulatory requirements are met. Changes to the SLMCPR value specified in the Technical Specification will require prior NRC approval. The deletion of the footnote associated with Technical Specification 2.1.A is an administrative change.

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 deletes the footnote contained in Technical Specification 2.1.A as the result of the NRC approval of Amendment 25 to NEDE-24011-P-A. Cycle-specific SLMCPR values will continue to be developed in accordance with NRC approved methods as specified in the Technical Specifications. These methods ensure that applicable regulatory requirements are met. Changes to the SLMCPR value specified in the Technical Specifications will require prior NRC approval. The deletion of the footnote associated with Technical Specification 2.1.A is an administrative change.

   Therefore, the proposed 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 is administrative. The change does 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 amendment is administrative. The change does not affect the integrity of the reactor coolant pressure boundary or any fission product barrier. Occupational exposures are not affected by the proposed change.

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. 294

Affected Technical Specification Pages

SECTION 2

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMIT - FUEL CLADDING INTEGRITY

Applicability: Applies to the interrelated variables associated with fuel thermal behavior.

Objective: To establish limits on the important thermal hydraulic variables to assure the integrity of the fuel cladding.

Specifications:

A. When the reactor pressure is greater than or equal to 800 psia and the core flow is greater than or equal to 10% of rated, the existence of a minimum CRITICAL POWER RATIO (MCPR) less than 1.09~~*~~ shall constitute violation of the fuel cladding integrity safety limit. DELETE ASTERISK

B. When the reactor pressure is less than 800 psia or the core flow is less than 10% of rated, the core thermal power shall not exceed 25% of rated thermal power.

C. In the event that reactor parameters exceed the limiting safety system settings in Specification 2.3 and a reactor scram is not initiated by the associated protective instrumentation, the reactor shall be brought to, and remain in, the COLD SHUTDOWN CONDITION until an analysis is performed to determine whether the safety limit established in Specification 2.1.A and 2.1.B was exceeded.

D. During all modes of reactor operation with irradiated fuel in the reactor vessel, the water level shall not be less than 4'8" above the TOP OF ACTIVE FUEL.

RELOCATE TO PAGE 2.1-2

Bases:

The fuel cladding integrity safety limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not necessarily result in damage to BWR fuel rods, the critical power at which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedure used to calculate the

~~*Applicable for cycle 18 only.~~

critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity safety limit is defined as the CRITICAL POWER RATIO in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties.

The Safety Limit MCPR is determined using the General Electric Thermal Analysis Basis, GETAB[1], which is a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the General Electric Critical Quality (X) - Boiling Length (L), GEXL, correlation.

The use of the GEXL correlation is not valid for the critical power calculations at pressures below 800 psia or core flows less than 10% of rated. Therefore, the fuel cladding integrity safety limit is protected by limiting the core thermal power.

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low power and all flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and all flows will always be greater than 4.56 psi. Analyses show that with a flow of 28×10^3 lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, bundle flow with a 4.56 psi driving head will be greater than 28×10^3 lbs/hr irrespective of total core flow and independent of bundle power for the range of bundle powers of concern. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors this corresponds to a core thermal power of more than 50%. Thus, a core thermal power limit of 25% for reactor pressures below 800 psi or core flow less than 10% is conservative.

Plant safety analyses have shown that the scrams caused by exceeding any safety setting will assure that the Safety Limit of Specification 2.1.A or 2.1.B will not be exceeded. Scram times are checked periodically to assure the insertion times are adequate. The thermal power transient resulting when a scram is accomplished other than by the expected scram signal (e.g., scram from neutron flux following closure of the main turbine stop valves) does not necessarily cause fuel damage. Specification 2.1.C requires that appropriate analysis be performed to verify that backup protective instrumentation has prevented exceeding the fuel cladding integrity safety limit prior to resumption of POWER OPERATION. The concept of not approaching a Safety Limit provided scram signals are OPERABLE is supported by the extensive plant safety analysis.

If reactor water level should drop below the TOP OF ACTIVE FUEL, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. With a water level above the TOP OF ACTIVE FUEL, adequate cooling is maintained and the decay heat can easily be accommodated. It should be noted that during power generation there is no clearly defined water level inside the shroud and what actually exists is a mixture level. This mixture begins within the active fuel region and extends up through the moisture separators. For the purpose of this specification water level is defined to include mixture level during power operations.

The lowest point at which the water level can presently be monitored is 4'8" above the TOP OF ACTIVE FUEL. Although the lowest reactor water level limit which ensures adequate core cooling is the TOP OF ACTIVE FUEL, the safety limit has been conservatively established at 4'8" above the TOP OF ACTIVE FUEL.

REFERENCES

(1) NEDE-24011-P-A~~-11,~~ General Electric Standard Application for Reactor Fuel ~~and US Supplement NEDE-24011-P-A-11-US.~~ (latest approved version as specified in the COLR)