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10 CFR 50.90

2130-05-20051
February 24, 2005

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. 319 – Revision to Table 3.1.1 Notes aa and bb Regarding Reactor Building Closed Cooling Water Pump and Service Water Pump Trip Conditions.

Pursuant to 10 CFR 50.90, AmerGen Energy Company, LLC (AmerGen) hereby requests changes to the Technical Specifications (TS) included in Oyster Creek Operating License No. DPR-16. These changes clarify the conditions under which the Reactor Building Closed Cooling Water (RBCCW) pumps and the Service Water (SW) pumps will trip during a Loss of Coolant Accident (LOCA). The existing notes aa and bb in Table 3.1.1, "Protective Instrumentation Requirements," state that the RBCCW pumps and SW pumps will trip during a LOCA. The proposed changes revise the notes to clarify that the RBCCW and SW pumps will trip during a LOCA only if offsite power is unavailable.

Additionally, an editorial change is being proposed to TS page 3.6-2 regarding the incorrect location of a footnote. The affected footnote was inadvertently relocated under TS Amendment 166, approved on 12/13/93.

Enclosure 1 to this letter provides the evaluation of the proposed changes and the no significant hazards consideration determination. Enclosure 2 provides the existing TS pages marked-up to show the proposed changes. The TS Bases pages are provided for information only and do not require NRC approval.

AmerGen requests approval of the proposed amendment by February 24, 2006. Once approved, the amendment will be implemented within 60 days.

AmerGen has concluded that the proposed changes present no significant hazards consideration under the standards set forth in 10 CFR 50.92.

The proposed changes to the Technical Specifications have undergone a review in accordance with Section 6.5 of the Oyster Creek Technical Specifications. No new regulatory commitments are included in this submittal.

We are notifying the State of New Jersey of this application for changes to the Technical Specifications by transmitting a copy of this letter and its attachments to the designated State Official.

If any additional information is needed, please contact Dave Robillard at (610) 765-5952.

I declare under penalty of perjury that the foregoing is true and correct.

Sincerely,

2/22/2005
Executed On

C. N. Swenson
Site Vice President
Oyster Creek Generating Station
AmerGen Energy Company, LLC

Enclosures: (1) Oyster Creek Technical Specification Change Request No. 319, Evaluation of Proposed Changes
(2) Oyster Creek Technical Specification Change Request No. 319, Markup of Proposed Technical Specification Page Changes

cc: S.J. Collins, Administrator, USNRC Region 1
P.S. Tam, USNRC Senior Project Manager, Oyster Creek
R. Summers, USNRC Senior Resident Inspector, Oyster Creek
K. Tosch, Director, Bureau of Nuclear Engineering, New Jersey Department of Environmental Protection
File No. 03044

Enclosure 1

Oyster Creek Technical Specification Change Request No. 319

Evaluation of Proposed Changes

1.0 INTRODUCTION

AmerGen Energy Company, LLC (AmerGen) is requesting a change to Technical Specifications (TS), Appendix A of the facility operating license No. DPR-16, for Oyster Creek Generating Station (OCGS). This proposed change revises Oyster Creek Technical Specifications Table 3.1.1, "Protective Instrumentation Requirements", notes aa and bb to provide clarification as to when the Service Water (SW) pumps and the Reactor Building Closed Cooling Water (RBCCW) pumps trip during a Loss of Coolant Accident (LOCA). These pumps will trip on a LOCA when offsite power is unavailable. There are no plant configuration changes associated with this proposed change.

Additionally, TS pages 3.6-1 and 3.6-2 are revised to correct the location of the footnote that applies to Specification 3.6.A.4.1. This footnote was inadvertently relocated under TS Amendment 166. This change is purely editorial in nature and has no impact on the requirements stated in Specification 3.6.A.4.1.

2.0 DESCRIPTION OF PROPOSED AMENDMENT

The proposed amendment will revise TS Table 3.1.1, note aa, which currently states that Service Water pump circuit breakers will be tripped in 10 seconds ±15% during a LOCA by relays SK7A and SK8A, and TS Table 3.1.1, note bb, which states that Reactor Building Closed Cooling Water pump circuit breakers will trip instantaneously during a LOCA. The proposed change adds the phrase "with a concurrent Loss of Offsite Power (LOOP)" to both notes.

Also, the footnote on existing TS page 3.6-1 is relocated to its correct location on TS page 3.6-2.

3.0 BACKGROUND

The primary functions of the Service Water system are to provide normal cooling for the Reactor Building Closed Cooling Water heat exchangers and alternate cooling for the Turbine Building Closed Cooling Water heat exchangers. Additionally, the SW system maintains the Emergency Service Water side of the Containment Spray heat exchangers full of water.

The primary function of the Reactor Building Closed Cooling Water system is to provide cooling water to various components in the Reactor Building and the Drywell. These components include the Spent Fuel Pool Cooling heat exchangers, Reactor Water Clean-up system heat exchangers, heating and ventilation coolers, drain tank coolers, and Reactor Recirculation pump and motor coolers.

Amendment 42 of the Oyster Creek Licensing Application concluded that the Service Water System and Reactor Building Closed Cooling Water System are not safety related, and are not required for the safe shutdown of the reactor or to mitigate the consequences of postulated accidents.

The Oyster Creek Updated Final Safety Analysis Report (UFSAR) Table 8.3-1 defines the Emergency Busses Automatic Loading Schedule during Loss of Offsite Power (LOOP), LOOP and LOCA, and LOOP and LOCA with single failure of Emergency Diesel Generator conditions. The UFSAR states that the SW and RBCCW pumps are automatically loaded onto the emergency busses only on LOOP. These loads are not loaded onto the emergency busses when LOOP and LOCA occur concurrently. These loads will be manually restarted by plant operators as the need arises, and emergency bus loading permits. On a LOCA when offsite power is available, these loads remain connected to their normal power supplies, because sufficient capacity is available on the busses.

4.0 REGULATORY REQUIREMENTS & GUIDANCE

Amendment 42 to the Oyster Creek Licensing Application concluded that the SW and RBCCW pumps are not required during the accident conditions. Therefore, tripping of these pumps during LOOP with LOCA is acceptable.

Amendment 60 to the Oyster Creek TS authorized changes in the Diesel Generator Load sequence timers for the SW and RBCCW pumps. This amendment added notes aa and bb to TS Table 3.1.1 describing the time delay conditions under which the SW and RBCCW pumps will trip during a LOCA. Notes aa and bb added in Amendment 60 stated that these pumps trip during a LOCA. However, the controls for the SW and RBCCW pumps do not trip these motors on a LOCA when offsite power is available.

The footnote on TS page 3.6-1, applicable to Specification 3.6.A.4.1, was inadvertently relocated under TS Amendment 166. TS pages 3.6-1 and 3.6-2 are revised to correct the location of this footnote. This change is purely editorial in nature and has no impact on the requirements stated in Specification 3.6.A.4.1.

5.0 TECHNICAL ANALYSIS

The proposed change provides clarification to Table 3.1.1, notes aa and bb for when the SW and RBCCW pumps trip whenever a LOCA condition exists. These pumps will only trip on a LOCA when offsite power is unavailable. This clarification does not require any plant configuration change.

The SW and RBCCW systems are not required for the safe shutdown of the reactor or to mitigate the consequences of postulated accidents. Following a complete loss of station normal and auxiliary power, the Circulating Water system, the Steam, Condensate and Feedwater systems, and the Reactor Recirculation system are shutdown. Thus, very little heat load remains on the RBCCW and Turbine Building Closed Cooling Water systems. One SW pump, operated on emergency diesel generator power, can provide cooling to both of these closed cooling water systems until power is restored. Loss of function of these systems has no adverse consequences on plant releases of radioactive effluents.

The proposed changes do not affect existing controls. Following a LOCA with offsite power available, these pumps will help in removing plant heat loads. Consequently, the proposed Technical Specification changes will not adversely affect nuclear safety or safe plant operations.

TS pages 3.6-1 and 3.6-2 are revised to correct the location of the existing footnote on TS page 3.6-1, applicable to Specification 3.6.A.4.1. This change is purely editorial in nature and has no impact on the requirements stated in Specification 3.6.A.4.1.

6.0 REGULATORY ANALYSIS

The technical analysis, provided in section 5, and the regulatory requirements, provided in section 4, to add clarification to the TS Table 3.1.1 and to describe the plant configuration as to when the SW and RBCCW pumps trip meets the criteria of 10 CFR 50.36 for inclusion as a condition for operation.

The footnote being relocated from TS page 3.6-1 to page 3.6-2 is applicable to Specification 3.6.A.4.1, which addresses requirements related to sampling of reactor coolant to verify that the reactor coolant specific activity is within specified limits. The Limiting Condition for Operation associated with reactor coolant specific activity meets the criteria of 10 CFR 50.36 for inclusion as a condition for operation.

7.0 NO SIGNIFICANT HAZARDS CONSIDERATION

AmerGen has determined that this License Amendment Request poses no significant hazards considerations as defined by 10 CFR 50.92.

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed revision to Technical Specification (TS) Table 3.1.1 to clarify the tripping of the Service Water (SW) and Reactor Building Closed Cooling Water (RBCCW) pumps documents the as-built controls for these loads. Amendment No. 42 to the Oyster Creek Licensing Application concluded that these pumps are not required to perform any functions related to safe plant shutdown. During a loss of coolant accident (LOCA) condition, with offsite power available, the plant electrical busses have enough capacity and capability to supply the SW and RBCCW pumps. This proposed change is an administrative change only, and is being made to align the Oyster Creek Technical Specifications with the design of the plant. No physical changes are being made to the plant. Also, the footnote on TS page 3.6-1 would be relocated to TS page 3.6-2 to appear on the same TS page as the Specification to which it applies. The proposed changes do not alter the physical design or operational procedures associated with any plant structure, system, or component.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed revision to Technical Specification Table 3.1.1 to clarify the tripping of the SW and RBCCW pumps documents as-built controls for these loads. These pumps provide cooling to various non-safety related plant equipment. Following a LOCA condition, with offsite power available, these pumps will help in removing plant heat loads. This clarification that the SW and RBCCW pumps do not trip during a LOCA, with offsite power available, does not affect the Emergency Diesel Generator time delayed loading sequence. The relocation of the footnote applicable to Specification 3.6.A.4.1 is editorial in nature and has no impact on any accident previously evaluated. Accordingly, the proposed changes do not introduce any new accident initiators, nor do they reduce or adversely affect the capabilities of any plant structure or system in the performance of their safety function.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The proposed revision to Technical Specification Table 3.1.1 to clarify the tripping of the SW and RBCCW pumps documents as-built controls for these loads. The NRC Safety Evaluation Report (SER) for Amendment 42 to the Oyster Creek Licensing Application concluded that it is acceptable to automatically trip the SW and RBCCW pumps during a loss of coolant accident. The NRC SER for Technical Specification Amendment 60 concluded that the immediate tripping of the RBCCW pump and the time delayed tripping of the SW pumps during a LOCA was also acceptable. The clarification that the SW and RBCCW pumps do not trip during a loss of coolant accident when offsite power is available does not reduce any margin of safety because these pumps are not required to mitigate the consequences of any postulated accident. The relocation of the footnote applicable to Specification 3.6.A.4.1 is editorial in nature and has no impact on any accident margin of safety.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above, AmerGen concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92, and accordingly, a finding of "no significant hazards consideration" is justified.

8.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to the installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluents that may be released offsite, or (iii) significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(c), no environmental impact statement or environmental assessment need be prepared in connection with the proposed license amendment.

9.0 PRECEDENT

The proposed changes to provide clarification to Technical Specification Table 3.1.1 for the conditions under which the RBCCW and SW pumps will trip on a LOCA and to relocate a footnote to its correct location are unique to OCGS.

10.0 REFERENCES

1) Oyster Creek Nuclear Generating Station Licensing Application, Amendment No. 42
2) Oyster Creek Nuclear Generating Station Technical Specification, Amendment No. 60
3) Oyster Creek Nuclear Generating Station Technical Specification, Amendment No. 166

Enclosure 2

Oyster Creek Technical Specification Change Request No. 319

Proposed Technical Specification Changes (Mark-up)

The pages included in this enclosure are:

PAGES

3.1-7
3.1-18
3.6.1
3.6.2

The low-low-low water level trip point is set at 4'8" above the top of the active fuel and will prevent spurious operation of the automatic relief system. The trip point established will initiate the automatic depressurization system in time to provide adequate core cooling.

Specification 3.1.B.1 defines the minimum number of APRM channel inputs required to permit accurate average core power monitoring. Specifications 3.1.B.2 and 3.1.C.1 further define the distribution of the OPERABLE chambers to provide monitoring of local power changes that might be caused by a single rod withdrawal. Any nearby, OPERABLE LPRM chamber can provide the required input for average core monitoring. A Travelling Incore Probe or Probes can be used temporarily to provide APRM input(s) until LPRM replacement is possible. Since APRM rod block protection is not required below 61% of rated power, as discussed in Section 2.3, Limiting Safety System Settings, operation may continue below 61% as long as Specification 3.1.B.1 and the requirements of Table 3.1.1 are met. In order to maintain reliability of core monitoring in that quadrant where an APRM is inoperable, it is permitted to remove the OPERABLE APRM from service for calibration and/or test provided that the same core protection is maintained by alternate means.

In the rare event that Travelling In-core Probes (TIPs) are used to meet the requirements 3.1.B or 3.1.C, the licensee may perform an analysis of substitute LPRM inputs to the APRM system using spare (non-APRM input) LPRM detectors and change the APRM system as permitted by 10 CFR 50.59.

Under assumed loss-of-coolant accident conditions and certain loss of offsite power conditions with no assumed loss-of-coolant accident, it is inadvisable to allow the simultaneous starting of emergency core cooling and heavy load auxiliary systems in order to minimize the voltage drop across the emergency buses and to protect against a potential diesel generator overload. The diesel generator load sequence time delay relays provide this protective function and are set accordingly. The repetitive accuracy rating of the timer mechanism as well as parametric analyses to evaluate the maximum acceptable tolerances for the diesel loading sequence timers were considered in the establishment of the appropriate load sequencing.

Manual actuation can be accomplished by the operator and is considered appropriate only when the automatic load sequencing has been completed. This will prevent simultaneous starting of heavy load auxiliary systems and protect against the potential for diesel generator overload.

Also, the Reactor Building Closed Cooling Water and Service Water pump circuit breakers will trip whenever a loss-of-coolant accident condition exists. This is justified by Amendment 42 of the Licensing Application which determined that these pumps were not required during this accident condition.

with a concurrent loss of offsite power

TABLE 3.1.1 (CONT'D)
Sheet 10 of 13

o. There are two time delay relays associated with each of two pumps. One timer per pump is for sequence starting (SK1A, SK2A) and one timer per pump is for tripping the pump circuit breaker (SK7A, SK8A).

p. Two time delay relays per pump must be OPERABLE.

q. Manual initiation of affected component can be accomplished after the automatic load sequencing is completed.

r. Time delay starts after closing of containment spray pump circuit breaker.

s. These functions not required to be OPERABLE with the reactor temperature less than 212°F and the vessel head removed or vented or during REACTOR VESSEL PRESSURE TESTING.

t. These functions may be inoperable or bypassed when corresponding portions in the same core spray system logic train are inoperable per Specification 3.4.A.

u. These functions not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required to be maintained.

v. These functions not required to be OPERABLE when the ADS is not required to be OPERABLE.

w. These functions must be OPERABLE only when irradiated fuel is in the fuel pool or reactor vessel and SECONDARY CONTAINMENT INTEGRITY is required per Specification 3.5.B.

y. Deleted

z. The bypass function to permit scram reset in the SHUTDOWN or REFUEL MODE with control rod block must be OPERABLE in this mode.

aa. Pump circuit breakers will be tripped in 10 seconds ± 15% during a LOCA by relays SK7A and SK8A.

bb. Pump circuit breakers will trip instantaneously during a LOCA.

with a concurrent Loss of Offsite Power (LOOP)

cc. Only applicable during STARTUP MODE while OPERATING in IRM range 10.

3.6 Radioactive Effluents

Applicability: Applies to the radioactive effluents of the facility.

Objective: To assure that radioactive material is not released to the environment in an uncontrolled manner and to assure that the radioactive concentrations of any material released is kept as low as is reasonably achievable and, in any event, within the limits of 10 CFR part 20.1301 and 40 CFR Part 190.10(a).

Specification:

3.6.A. Reactor Coolant Radioactivity

The specific activity of the primary coolant except during REFUEL MODE shall be limited to: Less than or equal to 0.2 microcuries per gram DOSE EQUIVALENT (D.E.) I-131

Limiting Condition for Operation

1. Whenever an isotopic analysis shows reactor coolant activity exceeds 0.2 uCi/gram DOSE EQUIVALENT (D.E.) I-131, operation may continue for up to 48 hours. Additional analyses shall be done at least once per 4 hours until the specific activity of the primary coolant is restored to within its limit. The provisions of Specification 3.0.C.3 are applicable.

2. If the reactor coolant activity is greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or greater than 4.0 microcuries per gram D.E. I-131, be in at least SHUTDOWN CONDITION within 12 hours. The provisions of Specification 3.0.C.3 are applicable.

3. Annual Reporting Requirement

   The results of specific activity analyses in which the reactor coolant exceeded the limits of Specification 3.6.A shall be reported on an annual basis. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded until after the radioiodine activity is reduced to less than the limit; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after radioiodine activity was reduced to less than the limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded until after the radioiodine activity is reduce to less than the limit; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and, (5) The time duration when specific activity of the primary coolant exceeded the radioiodine limit.

Move to page 3.6-2

* If there are consecutive thermal power changes by more than 15% per hour, take sample and analyze at least one sample between 2 and 6 hours following the change and at least once per four hours thereafter, until the specific activity of the primary coolant is restored to within limits.

4. With the reactor mode switch in Run or Startup position, with:

    1. Thermal power changed by more than 15% of rated thermal power in one hour*, or

    2. The off-gas level, at the SJAE, increased by more than 10,000 microcuries per second in one hour during steady state operation at release rates less than 75,000 microcuries per second, or

    3. The off-gas level, at the SJAE, increased by more than 15% in one hour during steady state operation at release rates greater than 75,000 microcuries per second,

    take sample and analyze at least one sample, between 2 and 6 hours following the change in thermal power or off-gas level and at least once per four hours thereafter, until the specific activity of the primary coolant is restored to within limits.

3.6.B <u>Liquid Radwaste Treatment</u> - RELOCATED TO THE ODCM

3.6.C <u>Radioactive Liquid Storage</u>

Applicability: Applies at all times to specified outdoor tanks used to store radioactive liquids.

1. The quantity of radioactive material, excluding tritium, noble gases, and radionuclides having half-lives shorter than three days, contained in any of the following outdoor tanks shall not exceed 10.0 curies:

    a. Waste Surge Tank, HP-T-3

    b. Condensate Storage Tank

2. In the event the quantity of radioactive material in any of the tanks named exceeds 10.0 curies, begin treatment as soon as reasonably achievable, continue it until the total quantity of radioactive material in the tank is 10 curies or less, and describe the reason for exceeding the limit in the next Annual Effluent Release Report.

3. Specifications 3.0.A and 3.0.B do not apply.

3.6.D <u>Condenser Offgas Treatment</u> - RELOCATED TO THE ODCM

3.6.E <u>Main Condenser Offgas Radioactivity</u>

1. The gross radioactivity in noble gases discharged from the main condenser air ejector shall not exceed 0.21/E Ci/sec after the holdup line where E is the average gamma energy (Mev per atomic transformation).

2. In the event Specification 3.6.E.1 is exceeded, reduce the discharge rate below the limit within 72 hours or be in at least SHUTDOWN CONDITION within the following 12 hours.

INSERT FROM PAGE 3.6-1